11 RADIOACTIVE WASTE MANAGEMENT

This chapter of the safety evaluation report (SER) documents the review by the U.S. Nuclear Regulatory Commission (NRC) staff (hereinafter referred to as the staff) of Chapter 11, "Radioactive Waste Management," of the NuScale Power, LLC (NuScale) (hereinafter referred to as the applicant), Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 2 of the SDAA, dated April 9, 2025 (Agencywide Documents Access and Management System Accession No. ML25099A237). The precise parameter values, as reviewed by the staff in this SER, are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this SER to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

This chapter contains the results of the NRC staff's review of the NuScale design-basis and realistic radioactive source terms, radioactive waste management system (RWMS), and process and effluent radiation monitoring instrumentation and sampling system (PERMISS).

The RWMS consists of the liquid radioactive waste system (LRWS), gaseous radioactive waste system (GRWS), and solid radioactive waste system (SRWS). The RWMS and PERMISS include the instrumentation used to monitor and control releases of radioactive effluents and wastes. These systems are designed for normal operations, including refueling outages, routine maintenance, and anticipated operational occurrences (AOOs). As operational events, AOOs include unplanned releases of radioactive materials associated with equipment failures, operator errors, and administrative errors, with radiological consequences that are not considered accident conditions.

11.1 Source Terms

11.1.1 Introduction

The NuScale Power Plant, which is designed to operate with up to six NuScale Power Modules, will generate radioactive materials during normal operations, including AOOs. These materials include fission, activation, and corrosion products, present both in primary and, to lesser extents, the secondary coolant. Radioactivity released in the primary coolant is developed from the reactor core fission product inventory using industry parameters of fission product escape rate coefficients, coolant cleanup rate, and demineralizer effectiveness. Radioactivity in the secondary coolant is developed from various removal mechanisms and primary-to-secondary leakage from the reactor coolant system (RCS) through steam generator tube defects. The radioactivity generated in the primary and secondary coolant comprises two radioactive source terms: a design-basis coolant source term and a realistic coolant source term.

The design-basis coolant source term is used to represent conditions for characterizing the radionuclide inventory and concentrations considered in the design of the LRWS, GRWS, and SRWS to collect, hold, and process the associated types and quantities of radioactivity and to describe how the process and effluent radiation monitoring system controls and monitors effluent releases. This source term serves as the basis for the FSAR Section 12.2, "Radiation Sources," source terms; conducting shielding analyses of structures, systems, and components

(SSCs); establishing radiation zones; and evaluating certain potential occupational radiation exposures to plant workers. The design-basis coolant source term also provides the radionuclide inventory and concentrations for the initial conditions used in design-basis accident consequence calculations. In the NuScale design, the design-basis coolant source term is scaled by a factor of 10 times the realistic coolant source term, except for the water activation and corrosion and activation products, as described later in this section.

The realistic source coolant term is used to represent conditions for characterizing the average radionuclide inventory and concentrations under normal operating conditions, including AOOs. This source term serves as the basis for evaluating the impacts of normal expected liquid and gaseous effluent releases to the environment and assessing doses to members of the public. In the NuScale design, several of the design parameters are outside the applicability range of the NRC pressurized-water reactor (PWR) Gaseous and Liquid Effluent (GALE)86 code, based on the guidance in NUREG-0017, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors,” issued April 1985 (ML112720411), for large light-water reactors (LWRs). Therefore, NuScale developed a realistic coolant source term using an alternate methodology to the GALE86 code. SER Section 11.1.4 contains the NRC staff’s review of this methodology.

Additional sources of radioactivity in coolant source terms, such as tritium (H-3), carbon (C)-14, argon (Ar)-41, and nitrogen (N)-16, are produced by neutron activation of constituents within the RCS. These water activation products occur independently of failed fuel and are handled differently in the NuScale design. CRUD (a term originating in the late 1950s and used to describe radioactive deposits) is produced from neutron activation of nonradioactive corrosion and wear products that are circulated in the primary coolant. These corrosion and wear product concentrations are also produced independently of failed fuel and are determined using the guidance in American National Standards Institute (ANSI)/American Nuclear Society (ANS)-18.1-1999, “Radioactive Source Term for Normal Operation of Light Water Reactors.”

For the DCA, the staff reviewed and approved the methodology described in NuScale technical report TR-123242, Revision 1, “Effluent Release (GALE Replacement) Methodology and Results.” TR-123242, Revision 1 is also used for SDAA. Under updated audit plan issued on July 29, 2024 (ML24211A089), the staff reviewed nondocketed information, including calculation packages and Microsoft Excel© spreadsheets in native format to verify the changes made to the SDAA source term information. This information helped the NRC staff understand the methods, models, parameters, and assumptions for developing the realistic coolant source term using fundamental first-principle calculations. The staff verified that the effluent release methodology remained essentially the same as in the DCA, but the numerical values and results were updated to account for revised source terms due to design changes (such as reactor power level changes) in the SDAA, as expected. SER Section 11.1.4 includes the staff’s evaluation of NuScale TR-123242.

11.1.2 Summary of Application

FSAR Section 11.1: The applicant described the design-basis and realistic coolant source terms in FSAR Section 11.1, “Source Terms.” This section provides information on the sources of radioactive material produced within the reactor core, primary and secondary coolant systems, and downstream processing through the LRWS and GRWS as liquid and gaseous wastes, respectively. FSAR Table 11.1-1, “Maximum Core Isotopic Inventory,” shows the bounding fuel isotopic inventory calculated using the industry standard ORIGEN code for the

NuScale design. This section also identifies the applicant's alternate methodology in NuScale TR-123242, Revision 1, used in developing the realistic coolant source terms.

Fission products, water activation products, and CRUD, which make up the coolant source terms in the NuScale design, are derived from fundamental first-principle calculations and nuclear power plant empirical and operating data found in ANSI/ANS 18.1-1999; NUREG-0017, Revision 1 (April 1985); and Electric Power Research Institute (EPRI) TR-1009903, "Tritium Management Model" (2005). FSAR Table 11.1-2, "Parameters Used to Calculate Coolant Source Terms," and Table 11.1-3, "Specific Parameters for Crud," show the parameters and values used to calculate the coolant source terms from the bounding fuel isotopic inventory.

Because the water activation products and CRUD are independent of failed fuel, the design-basis and realistic coolant source term concentrations are assumed to be the same radionuclide concentrations. The same methodology is applied to calculate the design-basis coolant source terms except for scaling the conservative realistic failed fuel fraction (RFFF) by a factor of 10 to obtain the design-basis failed fuel fraction described in SER Section 11.1.4. FSAR Table 11.1-4, "Primary Coolant Design Basis Source Term," and Table 11.1-5, "Secondary Coolant Design Basis Source Term," show the design-basis coolant source terms. FSAR Table 11.1-6, "Primary Coolant Realistic Source Term," and Table 11.1-7, "Secondary Coolant Realistic Source Term," show the realistic coolant source terms.

Technical Specifications (TS): The following TS are associated with the source terms and are found in SDAA Part 4, Volume 1:

- TS 3.4.8, "RCS Specific Activity"
- TS 5.5.1, "Offsite Dose Calculation Manual"
- TS 5.5.2, "Radioactive Effluent Control Program"
- TS 5.5.6, "Explosive Gas and Storage Tank Radioactivity Monitoring Program"

Technical Reports: NuScale TR-123242, Revision 1, is associated with this area of review.

Topical Reports: No topical reports are associated with this area of review.

11.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for the source term area of review, associated acceptance criteria, and review interfaces with other sections of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), appear in the SRP and in "Design Specific Review Standard for NuScale SMR [Small Modular Reactor] Design" (DSRS), Section 11.1, "Source Terms," issued June 2016 (ML15355A333). The following summarize the regulatory requirements:

- Title 10 of the *Code of Federal Regulations* (CFR) Part 20, "Standards for Protection Against Radiation," as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents released to unrestricted areas

- Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents considered in the context of numerical guides for design objectives and limiting conditions for operation to meet the criterion of as low as is reasonably achievable (ALARA) for radioactive material in LWR effluents

- General Design Criterion (GDC) 60, “Control of releases of radioactive materials to the environment,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, as it relates to determining the operational source term used in calculations associated with potential radioactivity in effluents released to unrestricted areas, such that a nuclear power unit design shall include means to suitably control the release of radioactive materials in gaseous and liquid effluents provided during normal reactor operation, including AOOs

- 10 CFR 52.137(a)(5), as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits established in 10 CFR Part 20

The following documents contain the regulatory positions and guidance for meeting the relevant requirements identified above:

- Regulatory Guide (RG) 1.112, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” issued March 2007 (ML070320241), as it relates to the method of calculating the release of radioactive materials in effluents from nuclear power plants

- ANSI/ANS-18.1-1999, as it relates to the methodology for determining the source term for normal reactor operations, including AOOs

- NUREG-0017, Revision 1, as it relates to PWRs, including (1) the volumes and concentrations of radioactive material given for normal operation and AOOs for each source of liquid and gaseous waste, (2) decontamination factors for in-plant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers, and evaporators, and (3) building mixing efficiency for containment internal cleanup

11.1.4 Technical Evaluation

Information needed to review the RWMS includes the types and quantities of radioactivity that are put into these systems for treating liquid and gaseous wastes. This includes consideration of parameters used to determine the amount of radioactive material from fission products released to the reactor coolant and the concentrations of all nonfission product radioactive isotopes in the reactor coolant. The source term analysis also determines bounding values of parameters to be used in evaluating RWMS capacities and effluent monitoring systems and in analyzing the consequences of certain postulated accidents. Industry experience and guidance are also principal determinants of expected values for source term parameters.

The NRC staff evaluated the information in FSAR Revision 1, Section 11.1, and NuScale TR-123242, Revision 1, against the applicable NRC regulations and guidance in DSRS Section 11.1.

The NRC staff found that the applicant provided NuScale TR-123242 as a replacement methodology for the GALE86 code that has been used for LWR design reviews. In this replacement methodology, the applicant provided information about the production, transport, and release of radionuclides. For the production of radionuclides, the applicant considered water activation, corrosion product activation or CRUD, and fission products. Where applicable, the applicant used information from ANSI/ANS-18.1-1999 and NUREG-0017, Revision 1, which support and provide the basis for the NRC's GALE86 code.

For the water activation pathways, the applicant described pathways for H-3, C-14, Ar-41, and N-16. The applicant documented the neutron reactions considered for each coolant activation pathway in its technical report. To determine the CRUD isotopes, the applicant referenced ANSI/ANS-18.1-1999, since it uses values that are representative of operating reactors with adjustment factors for differing plant parameters. For the fission product pathway, the applicant used the Standardized Computer Analyses Licensing Evaluation (SCALE), Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion) TRITON, and Oak Ridge Isotope GENeration (ORIGEN) computer codes to determine the isotopic distribution.

In addition to providing the source term data, the applicant described the plant parameters necessary to calculate coolant source terms in FSAR Table 11.1-2. The staff performed audits with NuScale to verify the information contained in FSAR Table 11.1-2 and to confirm that the information is acceptable to calculate subsequent coolant source terms in the NuScale US460 design. The staff's review of this information involved understanding the impacts of parameters such as primary coolant mass, primary coolant density, and escape rate coefficients in the applicant's spreadsheet calculations. The NRC staff reviewed the details of the calculations in the DCA and documented its findings in the NRC audit report of the NuScale effluent release (GALE replacement) methodology (ML18330A232). The NRC staff used the DCA audit to understand and verify the various source terms for water activation, CRUD, and fission product radionuclides that were included in the NuScale effluent release (GALE replacement) methodology and FSAR source term tables. In the DCA review, the staff also ensured that FSAR tables included those radionuclides that could have an impact on effluent release doses and were calculated in applicant spreadsheets. The staff audited the same calculations (ML24211A089) in the SDAA to verify that the changes to the source term in the SDAA were acceptable, based on increases to power level, cycle length, and burnup time. As discussed above, the staff found the information in the NuScale effluent release (GALE replacement) methodology (TR-123242) and related information in the SDAA FSAR, including FSAR Table 11.1-2, to be acceptable.

The applicant proposed the use of a new fuel failure rate for determining the amount of fission products in the reactor coolant for normal operations. NuScale TR-123242, Revision 1, discusses the selection of the 0.0066 percent RFFF in determining the amount of fission products in the realistic coolant source terms. The staff's review of the NuScale technical report finds that the RFFF information was determined based on a review of an EPRI database of fuel failure rates seen in the operating fleet. As documented in the NuScale DCA audit report dated April 30, 2018 (ML18103A198), the NRC staff contacted EPRI for questions on industry fuel failure data that the applicant used for the DCA. Based on information provided by EPRI, the NRC staff found that the source of compiled fuel failure data for U.S. nuclear power plants is from the EPRI Fuel Reliability Database (FRED). Using ERPI FRED data obtained during the audit, the applicant presented the NRC staff with an updated maximum RFFF value of 0.0066 percent or 66 parts per million (from 2007–2016 for U.S. PWRs). For the DCA, the staff determined that the proposed RFFF was acceptable given the history of fuel failure events and the selection of the highest fuel failure rate in the agreed-upon 2007 to 2016 timeframe.

For the SDAA review, the NRC staff finds that the RFFF used in the DCA analysis is also applicable to the current design; however, the source term for the SDA is different from the DCA due to other differences between the DCA and the SDA, such as the higher power level of each unit in the SDA and the reduced number of units in the SDA.

The NRC staff determined that the coolant source terms in the FSAR include technetium-99, a long-lived and environmentally mobile radionuclide produced in the fuel, which can escape as a fission product into the RCS for environmental release. The NRC staff verified the incorporation of this radionuclide among others specifically referenced in Branch Technical Position (BTP) 11-6, Revision 4, “Postulated Radioactive Releases Due to Liquid-Containing Tank Failures,” issued January 2016 (ML15027A401). Radionuclides such as H-3, C-14, strontium-90, iodine-129, cesium-137, bromine-84, rubidium-88, yttrium-91m, tellurium-129, tellurium-131, and cerium-143 for the environmental transport analysis were also verified, and the staff understands that these radionuclides will be used to perform the BTP 11-6 analysis referenced in Combined Operating License (COL) Item 11.2-2.

The NRC staff determined that the alternate methodology to calculate the H-3 production rate in the RCS appropriately estimates the H-3 liquid and gaseous effluent releases and offsite doses expected during normal operations. The NuScale application contains information on the generation of tritium and discusses the varying concentrations of tritium in FSAR Table 11.1-8, “Tritium Concentrations versus Primary Coolant Recycling Modes,” to show estimates for tritium based on different modes of operation. The applicant evaluated the H-3 concentration based on three recycling modes to maximize the H-3 concentration provided in FSAR Section 11.2, “Liquid Waste Management System,” FSAR Section 11.3, “Gaseous Waste Management System,” and NuScale TR-123242. Based on the staff’s assessment of the tritium concentrations in FSAR Table 11.1-8, “Tritium Concentration versus Primary Coolant Recycling Modes,” the staff finds that the applicant has appropriately calculated tritium concentrations for subsequent doses analysis in FSAR Sections 11.2, 11.3, and 12.2.

11.1.5 Combined License Information Items

This section contains no COL information items.

11.1.6 Conclusion

The NRC staff has determined that the NuScale design meets the applicable requirements cited in the SRP and DSRS Section 11.1, and that the applicant’s alternate method described in NuScale TR-123242 for calculating liquid and gaseous effluent releases during normal operations, including AOOs, is acceptable. The NRC staff further determined that these coolant source terms processed by the LRWS and GRWS, as discussed in SER Sections 11.2 and 11.3, respectively, will meet the applicable requirements of 10 CFR Part 20; 10 CFR Part 50, Appendix A, GDC 60; 10 CFR Part 50, Appendix I; and 10 CFR 52.137(a)(5). The NRC staff finds that the normal operation source terms found in Section 11.1 of the NuScale design, are acceptable, and their use in calculating doses associated with normal operation, including AOOs, will meet the applicable regulatory requirements in 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”

This conclusion is based on the following:

- The NuScale alternate method for developing normal liquid and gaseous effluent source terms is acceptable and, when used to determine estimated effluents, meets the regulatory requirements under 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," Table 2, Columns 1 and 2, for effluent concentration limits; dose limits for members of the public in 10 CFR 20.1301, "Dose limits for individual members of the public," and 10 CFR 20.1302, "Compliance with dose limits for individual members of the public"; and 10 CFR Part 50, Appendix I, design objectives. The applicant provided sufficient information to justify the GALE replacement parameters using the methods and guidance in Revision 1 of RG 1.112, Revision 1 of NUREG-0017, and ANSI/ANS 18.1-1999 in calculating liquid and gaseous effluent source terms for normal operations, including AOOs.

- The applicant has described an operation source term used in calculations associated with potentially radioactive effluents to unrestricted areas that supports subsequent dose calculations in determining doses to members of the public in support of GDC 60.

- In FSAR Chapter 11 and FSAR Section 12.2, the applicant identifies the kinds and quantities of radioactive material expected during normal operation, which addresses the source term requirements of 10 CFR 52.137(a)(5).

11.2 Liquid Waste Management System

11.2.1 Introduction

The LRWS is designed to collect, hold, and process liquid radioactive waste generated from normal operations and AOOs. After processing and sampling, liquids may be recycled or discharged. An operator located in the waste management control room operates the LRWS in a batch mode.

The LRWS receives radioactive fluids from the chemical and volume control system, the SRWS, the containment evacuation system (CES), the reactor component cooling water system, the pool cooling and cleanup water system (PCWS), the boron addition system, and the radioactive waste drain system (RWDS). The facility design reduces liquid effluent discharges from the LRWS to the environment by adequately processing liquid wastes and monitoring releases. The design uses a single point of discharge for liquid effluents to the environment through the LRWS discharge header, which is sent to the utility water system (UWS) discharge basin where it is diluted, monitored, and released.

11.2.2 Summary of Application

SDAA Part 8: The information associated with this section is found in SDAA Part 8, Section 3.12.

SDAA Part 8, Section 3.12, provides the inspections, tests, analyses, and acceptance criteria (ITAAC) associated with the design of the radioactive waste building (RWB). The ITAAC verify that the entire below-grade portions of the RWB and the above-grade portions of the building used for storage and processing of radioactive waste are built to the RW-IIa criteria in RG 1.143, Revision 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," issued November 2001 (ML013100305). This is based on the radionuclide contents of SSCs in the

RWB and as discussed in FSAR Sections 11.2, 11.3, and 11.4, “Solid Waste Management System,” and FSAR Chapter 12, “Radiation Protection”.

SDAA Part 8, Section 3.14, provides the ITAAC associated with the design of components. The ITAAC verify that components that process gaseous radioactive waste, including components associated with the LRWS degasifier that are classified RW-IIa in accordance with RG 1.143, are constructed to RW-IIa design criteria. This conforms to the information and conclusions made for RWMS categories for SSCs in the RWB and as discussed in FSAR Sections 11.2 and 11.3.

FSAR Section 11.2: The applicant described the seismic classification of the RWB and the reactor building (RXB) in FSAR Section 3.2, “Classification of Structures, Systems, and Components,” and FSAR Table 3.2-1, “Seismic Classification of Build Structures.”

FSAR Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” provides the top-level design features that include the seismic design of the RWB.

In FSAR Section 11.2, the applicant described the design of the LRWS and its functions in controlling, collecting, processing, storing, and disposing of liquid radioactive waste generated through normal operation, including AOOs. The LRWS, located in the RXB and RWB, is a non-safety-related system. The RXB is a seismic Category I structure, but several areas of the building are classified as seismic Category II, and the portions of the below-grade elevations of the RWB where the liquid waste management system components are housed is a seismic Category RW-IIa structure that meets the guidance of RG 1.143 for an RW-IIa safety classification (evaluated in SER Section 3.2.1). Failure of the LRWS does not adversely affect any safety-related system or component, and the system performs no safety function related to the safe shutdown of the plant. The quality assurance (QA) program (evaluated in SER Chapter 17) ensures the LRWS equipment and installation are in accordance with the codes and standards described in RG 1.143, Revision 2, Tables 1 through 4.

FSAR Figures 11.2-1a, “Liquid Radioactive Waste System Diagram,” through 11.2-1j, “Liquid Radioactive Waste System Diagram,” depict the process flow for the LRWS. The inputs to the LRWS are segregated as follows:

- low-conductivity waste (LCW) subsystem: coolant-grade boron and hydrogen-containing wastes with high radioactivity concentrations
- high-conductivity waste (HCW) subsystem: RXB and RWB floor drains through the RWDS, balance-of-plant drain system for chemical wastes, PCWS pool surge control tank overflow, RWDS RCCW system drain tanks, and RWDS RXB chemical drain tanks
- chemical waste subsystem: waste collected by the RWDS and manually introduced into the LRWS after sampling and analysis (FSAR Section 9.3.3, “Equipment and Floor Drain Systems”)
- detergent waste subsystem: detergent wastes from hand decontamination processes and personnel decontamination showers

FSAR Section 11.2.2.2, “High Conductivity Waste Subsystem,” states that the liquid wastes from the various sources are temporarily stored in collection tanks located in the RWB. System equipment and components are located in stainless-steel-lined, shielded cubicles as necessary to contain leaks and for radiation shielding. Other equipment areas, located outside of steel-lined cubicles, have concrete surfaces that are sealed with a coating, infused with epoxy, or equivalent to minimize absorption of contaminates. The system operates on a batch basis using skid-based processing equipment that includes filters, ion exchangers, and reverse osmosis components. After processing, the liquid is routed to sample tanks to monitor its quality before recycling or release. If the quality is not acceptable, the water is returned to a collection tank for further treatment.

The LRWS design in FSAR Table 11.2-1, “Major Component Design Parameters,” includes the following nominal tank volumes:

- two 60,567-liter (16,000-gallon) LCW collection tanks
- two 60,567-liter (16,000-gallon) HCW collection tanks
- two 60,567-liter (16,000-gallon) LCW sample tanks
- two 60,567-liter (16,000-gallon) HCW sample tanks
- one 1,893-liter (500-gallon) detergent waste collection tank
- one 37,854-liter (10,000-gallon) demineralized water break tank

The LRWS is operated and monitored from the waste management control room. Parameters such as tank levels and processing flow rates are indicated, alarmed, or both, to provide information on operational and equipment performance. FSAR Table 11.2-2, “Off-Normal Operation and Anticipated Operational Occurrence Consequences,” summarizes off-normal events along with the associated indications, system responses, and corrective actions.

The LRWS is designed to control leakage and facilitate access, operation, inspection, testing, and maintenance to maintain radiation exposures to operating and maintenance personnel ALARA and to minimize contamination of the facility.

The applicant, in FSAR Section 11.2.2, “System Description,” stated that the LRWS design includes the following maintenance considerations:

- location of redundant permanent plant equipment in separate shielded cubicles

- clean-in-place provisions to reduce the radiation source term before maintenance

- redundant components allowing uninterrupted waste operation and flexibility in maintenance scheduling

The COL licensee will develop administrative procedures governing the operation of all subsystems, control the treatment of various process and waste streams, and prevent accidental discharges into the environment.

In assessing the radiological impacts from radioactive liquid effluent releases, the FSAR provides the text and tables in FSAR Section 11.2 to present information supporting the development of the liquid source term and compliance with the effluent concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, and 10 CFR 20.1301(e), insofar as it requires meeting the U.S. Environmental Protection Agency (EPA) environmental radiation protection standards of 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” and the numerical design objectives of 10 CFR Part 50, Appendix I. The results show that expected annual liquid effluents released during normal operation, including AOOs, in unrestricted areas and doses to members of the public comply with the NRC regulations and conform to NRC guidance. As discussed in SER Section 11.2, the results also demonstrate compliance with the ALARA requirements of 10 CFR Part 50, Appendix I, and the SRP acceptance criteria in BTP 11-6 for the postulated failure of a liquid tank containing radioactivity.

Technical Specifications: TS 5.5.1 and 5.5.2 are associated with the LRWS and are found in SDAA Part 4, Volume 1. The TS also include the following reports: 5.5.6, "Explosive Gas and Storage Tank Radioactivity Monitoring"; 5.6.1, "Annual Radiological Environmental Operating Report"; and 5.6.2, "Radioactive Effluent Release Report."

Technical Reports: NuScale TR-123242, Revision 1 is associated with this area of review.

Topical Reports: No topical reports are associated with this area of review.

11.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for the LRWS area of review, associated acceptance criteria, and review interfaces with other SRP sections appear in SRP Section 11.2, "Liquid Waste Management System." The following summarizes the regulatory requirements:

- 10 CFR 20.1301, as it relates to dose limits for individual members of the public
- 10 CFR 20.1302, as it relates to limits on doses to members of the public and liquid effluent concentrations and doses in unrestricted areas
- 10 CFR 20.1406, "Minimization of contamination," as it relates to facility design and operational procedures for minimizing facility contamination and the generation of radioactive waste
- 10 CFR Part 20, Appendix B, Table 2, Column 2, as it relates to the liquid effluent concentration limits for release to the environment
- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors," as it relates to the inclusion of sufficient design information in demonstrating compliance with the design objectives for equipment necessary to control releases of radioactive effluents to the environment
- 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors," as it relates to TS requiring that operating procedures be developed for radiological monitoring and sampling equipment as part of the administrative controls and surveillance of effluent controls in meeting the ALARA criterion and 10 CFR 20.1301
- 10 CFR Part 50, Appendix I, Sections II.A and II.D, as they relate to numerical guidelines and design objectives and limiting conditions for operation in meeting dose criteria and the ALARA criterion of Appendix I
- GDC 60, as it relates to the design of LRWS to control releases of liquid radioactive effluents
- GDC 61, "Fuel storage and handling and radioactivity control," as it relates to the design of the LRWS in ensuring adequate safety under normal operations and postulated accident conditions

- GDC 64, “Monitoring radioactivity releases,” as it relates to the design of the LRWS to monitor for radioactivity that may be released from normal operations, including AOOs, and from postulated accidents

- 40 CFR Part 190 (EPA’s generally applicable environmental radiation standards), as implemented under 10 CFR 20.1301(e) as it relates to controlling doses within the EPA’s generally applicable environmental radiation standards

- 10 CFR 52.47(b)(1), which requires that applications for design certifications (DCs) contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC and provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations

- 10 CFR 52.137(a)(5), as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits established in 10 CFR Part 20

The following documents contain the regulatory positions and guidance for meeting the relevant requirements of the regulations identified above:

- RG 1.109, Revision 1, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,” issued October 1977 (ML003740384), as it relates to demonstrating compliance with the numerical guidelines for dose design objectives and the ALARA criterion of 10 CFR Part 50, Appendix I

- RG 1.110, Revision 1, “Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors,” issued October 2013 (ML13241A052), as it relates to performing a cost-benefit analysis (CBA) for reducing cumulative doses to populations by using available technology

- RG 1.112, Revision 1, as it relates to the acceptable methods for calculating annual average releases of radioactivity in effluents

- RG 1.113, Revision 1, “Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I,” issued April 1977 (ML003740390), as it relates to the use of acceptable methods for estimating aquatic dispersion and transport of liquid effluents in demonstrating compliance with dose objectives in 10 CFR Part 50, Appendix I

- RG 1.143, Revision 2, as it relates to the seismic design and quality group classification of components used in the LRWS and the structures housing this system, as well as provisions used to control leakage

- RG 1.206, Revision 1, “Applications for Nuclear Power Plants,” issued October 2018 (ML18131A181), as it relates to the minimum information requirements specified in 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” to be submitted in a COL application

- RG 1.33, Revision 2, “Quality Assurance Program Requirements (Operation),” issued February 1978 (ML003739995), as it relates to QA for operating the LRWS provisions for sampling and monitoring radioactive materials in process and effluent streams and controlling radioactive effluent releases to the environment

- RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” issued June 2008 (ML080500187), as it relates to minimizing both the contamination of equipment, plant facilities, and the environment and the generation of radioactive waste during plant operation

- BTP 11-6, Revision 4 (January 2016), as it relates to the assessment of radiological impacts associated with the assumed failure of an LRWS tank

- NUREG-0017, Revision 1, as it relates to the methodology for calculating gaseous and liquid effluent releases

- NUREG-1301, “Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors,” issued April 1991 (ML091050061), as it relates to guidance for calculating doses from PWR plants

- NUREG/CR-4013, “LADTAP II—Technical Reference and User Guide,” issued April 1986 (ML14098A069), as it relates to the methodology for calculating liquid effluent doses

- Nuclear Energy Institute (NEI) 08-08A, Revision 0, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination,” issued October 2009, and RG 4.21, as they relate to acceptable levels of detail and content needed to demonstrate compliance with the programmatic elements of 10 CFR 20.1406

- Generic Letter (GL) 89-01, “Implementation of Programmatic and Procedural Controls for Radiological Effluent Technical Specifications,” Supplement No. 1, dated November 14, 1990, as it relates to an operational program that addresses the development of a site-specific radiological environmental monitoring program

- Inspection and Enforcement (IE) Bulletin No. 80-10, “Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment,” dated May 6, 1980, as it relates to methods and procedures used in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled releases of radioactivity

- ANSI/ANS-18.1-1999, as it relates to the methodology for determining the source term for normal reactor operations, including AOOs

11.2.4 Technical Evaluation

The NRC staff evaluated the information in FSAR, Section 11.2, against the applicable NRC regulations and guidance in SRP Section 11.2 and in DSRS Section 11.2, “Liquid Waste Management System,” issued June 2016 (ML15355A334).

11.2.4.1 Design Considerations

11.2.4.1.1 General Design Criteria 60, 61, and 64 and 10 CFR 50.34a

The applicant must meet the requirements of GDC 60 and 61 with respect to controlling releases of radioactive materials to the environment. GDC 60 requires that the nuclear power unit design include provisions to handle radioactive wastes produced during normal reactor operations, including AOOs. GDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. GDC 64 requires that the LRWS be designed to monitor radiation levels and radioactivity in effluents and radioactive leakages and spills during routine operations, including AOOs.

The NRC staff considered the ability of the proposed liquid radwaste treatment management system to meet the demands of the plant resulting from AOOs and has concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant. The applicant has met the requirements of GDC 60 and 61 with respect to controlling releases of radioactive materials to the environment using automatic control features for terminating liquid effluent discharges or diverting process flows to systems for storage and further processing, as needed. GDC 64 is met because the design includes a liquid effluent radiation monitor to check the effluent discharge from the UWS basin during normal operations and AOOs.

In assessing compliance with GDC 60 and 61, the NRC staff reviewed the QA provisions and guidance in RG 1.143, Revision 2, specified by the applicant in the FSAR. The applicant stated that the LRWS will conform to RG 1.143, Revision 2. FSAR Table 11.2-1 also identifies the seismic category, quality group, and safety class for components of the LRWS. The QA program is in agreement with ANSI/ANS-55.6, “Liquid Radioactive Waste Processing Systems for Light Water Reactor Plants,” consistent with RG 1.143, Revision 2, Regulatory Position C.7. In determining the design for radwaste systems, the applicant provided FSAR Table 11.2-1, which includes the radwaste system component classifications consistent with the guidance specified in RG 1.143, Revision 2. Components are classified using the calculated design-basis quantities of radioactive material within each component. Classifications are based on the values for A_1 and A_2 in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” Appendix A, “Determination of A_1 and A_2,” consistent with RG 1.143, Revision 2.

Conformance to the guidance in RG 1.143, Revision 2, demonstrates that the assigned safety classifications (i.e., RW-IIa, RW-IIb, and RW-IIc) of SSCs for the LRWS, GRWS, and SRWS comply with GDC 61, as they relate to natural phenomena and human-induced hazards. The applicant defined the boundaries of the radwaste systems classifications as those components up to and including the system isolation devices. This definition is consistent with system boundaries reviewed in the past, and the NRC staff finds this boundary definition acceptable.

As indicated above, the FSAR Section 12.2 tables provide source terms for various component inventories and radwaste safety classifications for each radwaste SSC in FSAR Table 11.2-1. The NRC staff reviewed the information, performed a confirmatory analysis using the source term inventories, and verified the safety classifications of many of the radwaste SSCs. Based on this review, the staff finds that the applicant has correctly determined the radwaste safety classification for radwaste system components.

The staff's review finds that the applicant has followed the guidance contained in RG 1.143 by defining component source terms and correctly determined the radwaste safety classification. As a result, the staff finds that the applicant conformed to GDC 61 as it relates to the design of SSCs for the LRWS, GRWS, and SRWS.

For compliance with 10 CFR 50.34a, the applicant must provide sufficient information to demonstrate that the design objectives of equipment necessary to treat and control releases of radioactive effluents into the environment have been met. The requirements of 10 CFR 50.34a are met by describing the process used to treat and handle liquid waste and by identifying the system boundaries. The applicant described the LRWS in FSAR Section 11.2.2 for the planned pathways for a release to the environment. SER Section 11.5 discusses radiation monitoring to limit or control releases to the environment. The applicant also provided FSAR Tables 11.2-5, "Estimated Annual Releases to Liquid Radioactive Waste System Discharge," and 11.2-7, "Liquid Effluent Dose Results for 10 CFR 50 Appendix I," to summarize the releases to the environment that meet the requirements to quantify each of the principal radionuclides expected to be released annually to unrestricted areas in liquid effluents produced during normal reactor operations under 10 CFR 50.34a(e)(2). The section below discusses the staff's evaluation of the effluent release results.

11.2.4.1.2 10 CFR Part 50, Appendix I, Liquid Effluent Doses

The NRC staff reviewed FSAR Section 11.2.3, Radioactive Effluent Releases," to verify compliance with 10 CFR Part 50, Appendix I, Sections II.A and II.D. The applicant calculated the liquid effluent release doses using the NRC-endorsed LADTAP II computer code. Using the information provided by the applicant from FSAR Table 11.2-6, "LADTAP II Inputs," and LADTAP II input and output files, the NRC staff performed a confirmatory analysis of the liquid effluent release doses in FSAR Table 11.2-7.

The facility design reduces liquid radioactive effluent discharges from the LRWS to the environment by processing liquid wastes and then monitoring any liquid radioactive effluent releases. The LRWS design uses a single point of discharge for liquid effluents to the environment through the LRWS discharge header, which is sent to the UWS discharge basin where it is diluted, monitored, and released.

The UWS provides the single point liquid effluent release to the environment. The requirements of GDC 60 are met by controlling the release of liquid effluent above predetermined limits to the environment through the single point liquid effluent release point and by providing the ability to isolate the source of the liquid effluent. GDC 64 is met because the design provides a liquid effluent radiation monitor to check the effluent discharge from the UWS basin during normal operations and AOOs. The NRC staff performed confirmatory calculations for LRWS releases using parameters contained in FSAR Section 11.2 and Table 11.2-6.

In FSAR Table 11.2-6, the applicant summarized the liquid inputs used for the LADTAP II code. This table includes a pointer to describe the use of FSAR Table 11.2-5 for the liquid effluent source term in curies per year (Ci/yr) for the analysis. In SER Section 11.1, the NRC staff confirmed this source term and the results produced by NuScale's effluent release methodology. The NRC staff also confirmed the use of the RG 1.109 default values since this design does not have site-specific values to reference for the calculation.

In performing these confirmatory dose calculations, the NRC staff reviewed the applicant input and output files and FSAR tables. These values were found to agree with RG 1.109, Table E5, "Recommended Values for UAP to Be Used for the Maximum Exposed Individual in Lieu of

Site-Specific Data.” The applicant provided information to support the input and output files for the LADTAP II code. During the review of the liquid effluent release input, the NRC staff found that there were several discharge and dilution flow rates provided in FSAR Section 11.2. The distinction between these values was pertinent to the verification of the calculation of the liquid effluent release doses. These discharge and dilution flow rates are used to calculate effluent discharge concentrations and dose rate values. Different values were used for different purposes. The “liquid effluent discharge flow rate” value of 605 gpm (1.35 cfs) in FSAR Table 11.2-6 is the minimum discharge flow rate from the plant to meet 10 CFR Part 20, Appendix B, effluent release concentration limits (note that this value is less than the UWS dilution factor of 700 gpm (1.56 cfs) specified in FSAR Table 11.2-4, “Liquid Effluent Release Calculation Inputs,” which is used to calculate the discharge concentrations provided in FSAR Table 11.2-9, “Liquid Release Concentrations Compared to 10 CFR 20 Appendix B Limits.”). The “Off-site minimum dilution flow rate (river plus liquid effluent discharge)” value of 141 cfs (approximately 63,285 gpm) in FSAR Table 11.2-6 is the amount of total dilution flow, when including an assumed discharge to a river (or other body of water), which is used in calculating the doses provided in FSAR Table 11.2-7 for meeting the 10 CFR 50, Appendix I criteria. While NuScale used the 605 gpm (1.35 cfs) value in their LADTAP calculations, the calculation results were adjusted to account for the 141 cfs minimum dilution flow rate value, in order to calculate the values provided in FSAR Table 11.2-7.

To confirm the results provided by the applicant in FSAR Table 11.2-7, the NRC staff used the LADTAP II inputs reported in FSAR Table 11.2-6. The results of the NRC staff’s confirmatory calculation verified the estimated total body, organ, and age group doses. In FSAR Section 11.2.3.1, Radioactive Releases,” and FSAR Table 11.2-7, the applicant reported a total body dose and a maximum organ dose. SER Table 11.2-1 compares the applicant’s and the NRC staff’s results against the effluent dose design objectives in 10 CFR Part 50, Appendix I.

Table 11.2-1 Comparison of the Applicant and NRC Estimated Annual Individual Doses from Liquid Effluent Releases in mSv/yr (mrem/yr)

Pathway	Applicant Results	NRC Staff Results	Design Objective
Total Body	0.029 (2.9)	0.0268(2.68)	0.03 (3)
Max Organ Exposed Organ Age Group	0.045 (4.5) Liver Child	0.0362 (3.62) Liver Child	0.1 (10)

The NRC staff’s review determined that the applicant had appropriately calculated the liquid effluent release doses and that the results are within the ALARA design objectives in 10 CFR Part 50, Appendix I. The NRC staff’s confirmatory calculations verified that the applicant’s approach is acceptable.

The NRC staff’s determination is based on the use of non-site-specific data for the analyses. Presently, the applicant uses conservative estimates in its LADTAP II analysis to show the bounding results for liquid releases. An applicant that references the NuScale Power Plant SDA will perform a site-specific evaluation using the site-specific parameters as described in COL Item 11.2-1. Following COL Item 11.2-1, the COL applicant is to calculate doses to members of the public using site-specific parameters and compare the doses resulting from the liquid effluents with the numerical design objectives of Appendix I to 10 CFR Part 50, 10 CFR 20.1302, and 40 CFR Part 190.

11.2.4.1.3 Site-Specific Cost-Benefit Analysis

FSAR Section 11.2.3.4, "Site-Specific Cost-Benefit Analysis," provides COL Item 11.2-4. COL Item 11.2-4 specifies that the COL applicant will provide the site-specific cost-benefit analysis (CBA) to demonstrate compliance with the requirements of 10 CFR 50.34a and 10 CFR Part 50, Appendix I, Sections II.A and II.D. The CBA is to be performed using the guidance of RG 1.110. RG 1.110 describes an acceptable method of performing a CBA to demonstrate that the LRWS design includes all items of reasonably demonstrated technology for reducing to ALARA levels cumulative population doses from releases of radioactive materials from each reactor.

Because the CBA requires site-specific information, which is outside the scope of the FSAR, the NRC staff finds the inclusion of COL Item 11.2-4 acceptable.

11.2.4.1.4 10 CFR Part 20, Appendix B, Effluent Concentration Limits

FSAR Table 11.2-9, "Liquid Release Concentrations Compared to 10 CFR 20 Appendix B Limits," shows that the sum of the fractions (i.e., "unity rule" calculation) determined from summing the ratio of the assumed discharge concentration and respective effluent concentration limit for each radionuclide released in liquid effluent met the unity rule calculation specified in Note 4 of 10 CFR Part 20. The NRC staff notes that the liquid effluent site concentrations appear to be below the limits in 10 CFR Part 20, Appendix B, Table 2, Column 2, and the unity calculation described in Note 4.

The NRC staff performed a confirmatory analysis of the results presented in FSAR Table 11.2-9 and verified that the unity rule calculation was less than one for all radionuclides identified in FSAR Table 11.2-9. The results of the NRC staff's confirmatory analysis determined the information contained in FSAR Table 11.2-9 is acceptable and the applicant has demonstrated compliance with the 10 CFR Part 20, Appendix B, effluent concentration limits.

In accordance with COL Item 11.2-3, a COL applicant that references the NuScale Power Plant US460 standard design will perform a site-specific evaluation using the site-specific source term and dilution flow for liquid effluent releases and confirm that the discharge concentrations do not exceed the limits specified by 10 CFR Part 20, Appendix B, Table 2. It is appropriate for the COL applicant to perform a site-specific analysis to ensure that discharge concentrations do not exceed limits. As a result, the staff finds the COL item to be acceptable.

11.2.4.1.5 10 CFR 20.1301(e) Compliance with 40 CFR Part 190

Using the applicant's LADTAP II code input and output files and response, the NRC staff reviewed the doses from liquid effluent releases to members of the public in unrestricted areas to evaluate compliance with 10 CFR Part 50, Appendix I; 10 CFR 20.1302; and 40 CFR Part 190. Input values pertaining to environmental characteristics (i.e., the hydrologic model, water type, dilution factors, irrigation rates, usage and consumption factors, and exposure pathways) rely on site-specific information addressed by the COL applicant in COL Item 11.2-1.

COL Item 11.2-1 states that an applicant that references the NuScale Power Plant US460 standard design will calculate doses to members of the public using the site-specific parameters; compare those liquid effluent doses to the numerical design objectives of 10 CFR Part 50, Appendix I; and comply with the requirements of 10 CFR 20.1302 and 40 CFR Part 190. Consistent with COL Item 11.2-1, the COL applicant will be required to use the effluent doses related to liquid and gaseous effluents and any direct radiation from each site to determine the actual site’s 40 CFR Part 190 direct dose.

11.2.4.1.6 Minimization of Contamination, 10 CFR 20.1406

The NRC staff reviewed the information in FSAR Section 11.2 and FSAR Section 12.3.6, “Minimization of Contamination and Radioactive Waste Generation,” against the criteria in 10 CFR 20.1406 for minimizing contamination. In 10 CFR 20.1406, the NRC requires, in part, that design features and procedures 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. In FSAR Section 12.3.6, the applicant described the use of RG 4.21 in meeting the requirements of 10 CFR 20.1406. RG 4.21 describes an acceptable method for implementing a program that will minimize radioactive waste generation and contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste.

The tables in FSAR Section 12.3, “Radiation Protection Design Features,” give information on how the design of the LRWS includes provisions to minimize the contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste. Information in the tables in FSAR Section 12.3 includes pressure monitoring to check for leakage; ensuring sufficient capacity for tanks; avoiding embedded piping and buried piping and using double walled piping with leak detection for buried piping; allowing for the collection of radioactive waste from drains; and allowing for those wastes to be sent to the LRWS for processing and disposal.

FSAR Section 12.3.1.1.2, “Valves” indicates that double isolation valves are used at the interface between contaminated and non-contaminated systems to prevent cross contamination and that valves are designed to fail to the safe position. In addition, the NuScale design includes various radiation monitors that would identify unexpected leaks to non-radioactive systems, such as radiation monitoring of the reactor component cooling water system. These design features minimize contamination in accordance with 10 CFR 20.1406 and satisfy IE Bulletin No. 80-10 in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases to the environment.

FSAR Section 12.3.6 and SER Section 12.3 provide additional information on compliance with 10 CFR 20.1406, including programmatic requirements to be addressed by a COL applicant.

11.2.4.1.7 Mobile or Temporary Equipment

The NuScale LRWS is designed with permanently installed equipment. The LRWS does not include mobile or temporary equipment. The design includes sufficient radwaste processing and holdup capabilities at design-basis waste generation rates to address the liquid radioactive waste holdup capacity requirement of GDC 60 and the guidance in NuScale DSRS Section 11.2, without the need for mobile or temporary equipment. The holdup capabilities of the permanently installed equipment are sufficient to allow adequate retention of radioactive materials for delayed treatment if processing equipment would be temporarily unavailable for maintenance.

11.2.4.1.8 Radioactive Effluent Releases Caused by Failure of Radioactive Liquid Tank, *Branch Technical Position 11-6*

BTP 11-6 provides acceptable approaches in addressing radioactive effluent releases caused by failure of an outside radioactive liquid tank.

FSAR Section 11.2.3.2, "Compliance with Branch Technical Position 11-6," states that the only outdoor tank expected to contain radioactive liquids is the PCWS pool surge control storage (PSCS) tank. This tank has a corresponding containment tank with sufficient volume to store the contents of the PCWS PSCS tank plus the contents of the related piping. The applicant provided the source term information for this tank in FSAR Table 12.2-9, "Pool Cooling and Cleanup System Component Source Terms - Radionuclide Content," with the water mass and volume information given in FSAR Table 12.2-8, "Pool Cooling and Cleanup System Component Source Term Inputs and Assumptions."

The NRC staff's review of this information confirms that the limiting radiological consequence from a failure of an active component could occur from the PCWS PSCS tank containing radioactive material. Thus, the applicant provided COL Item 11.2-2 in FSAR Section 11.2.3.2 for the COL applicant to perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the PSCS storage tank in accordance with BTP 11-6. The NRC staff finds the inclusion of this COL item appropriate, given that the applicant has identified the source terms and volume of the tank to allow a COL applicant to calculate the transport of radionuclides based on site-specific parameters. The NRC staff finds the information in FSAR Section 11.2.3.2 to be acceptable to provide an adequate level of safety during normal reactor operation, including AOOs, because using the information in the FSAR and the approach described in COL Item 11.2-2, a COL applicant will perform an analysis in accordance with BTP 11-6 and ensure that the requirements referenced in BTP 11-6 and related acceptance criteria are met.

The staff also notes that TS 5.5.6, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," requires a surveillance program to ensure that, for all outdoor liquid tanks with radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have overflows and surrounding area drains connected to the LRWS, the quantity of radioactivity they contain is less than the quantity that would result in concentrations less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. Based on this, the staff finds that the liquid content in tanks is adequately controlled so that an accidental release or failure of a liquid radioactive tank will not result in exceeding the criteria in 10 CFR Part 20, Appendix B, Table 2, Column 2.

11.2.4.1.9 Technical Specifications

From the review of the US460 generic TS in SDAA Part 4, the NRC staff determined that no TS are directly associated with liquid waste storage and processing. However, SDAA Part 4, TS 5.5.1, requires an established, implemented, and maintained offsite dose calculation manual (ODCM). TS 5.5.2 has provisions for a radioactive effluent control program that conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining ALARA the doses to members of the public from radioactive effluents.

In SDAA Part 4, TS 5.6.1 and TS 5.6.2 specify annual reporting requirements in describing the results of the radiological monitoring program and summarize the quantities of radioactive liquid effluents released to the environment. As stated in TS 5.5.1 for COL licensee-initiated changes to the ODCM, there must be sufficient information to support the change(s), together with the appropriate analyses or evaluations justifying the change(s), and there must be a determination that the change(s) do not adversely impact the accuracy and reliability of effluent or dose calculations and that the levels of radioactive effluent control comply with the requirements of 10 CFR 20.1302; 40 CFR Part 190; 10 CFR 50.36a; and 10 CFR Part 50, Appendix I. The TS also require the radioactive effluent controls program, which is described in the ODCM, to include instrumentation to monitor and control liquid effluent discharges; meet limits on effluent concentrations released to unrestricted areas; monitor, sample, and analyze liquid effluents before and during releases; set limits on annual and quarterly dose commitments to a member of the public; and assess cumulative doses from radioactive liquid effluents.

FSAR Section 11.5.1.4, "Offsite Dose Calculation Manual and Radiological Environmental Monitoring Program," specifies that the ODCM and radiological environmental monitoring program (REMP) are developed and implemented in accordance with the recommendations and guidance of NEI 07-09A, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," issued March 2009. The NRC staff finds the proposed TS requirements acceptable because the applicant described those programs that will address the administrative programs on radioactive effluent controls and monitoring and will follow the approved guidance contained in NEI 07-09A for the development of the ODCM and REMP.

11.2.5 Combined License Information Items

SER Table 11.2-2 lists the COL item number, descriptions, and FSAR section for the LRWS COL items.

Table 11.2-2 NuScale COL Items for FSAR Section 11.2

COL Item No.	COL Item Description	FSAR Section No.
11.2-1	An applicant that references the NuScale Power Plant US 460 standard design will calculate doses to members of the public using the site-specific parameters, compare those liquid effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.2.3.1
11.2-2	An applicant that references the NuScale Power Plant US 460 standard design will perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the PSCS storage tank in accordance with NRC Branch Technical Position 11-6.	11.2.3.2
11.2-3	An applicant that references the NuScale Power Plant US 460 standard design will perform a site-specific evaluation using the site-specific source term and dilution flow for liquid effluent releases and confirm that the discharge concentrations do not exceed the limits specified by 10 CFR 20, Appendix B, Table 2.	11.2.3.3
11.2-4	An applicant that references the NuScale Power Plant US 460 standard design will perform a cost-benefit analysis as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory	11.2.3.4

COL Item No.	COL Item Description	FSAR Section No.
	requirements. This cost-benefit analysis is to be performed using the guidance of RG 1.110.	

The NRC staff finds that the list adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items for inclusion in the FSAR for the LRWS.

11.2.6 Conclusion

The NRC staff has determined that the NuScale design meets the applicable requirements discussed in SER Section 11.2. The NRC staff concludes that the LRWS, as a shared system, includes the equipment necessary to collect, process, handle, store, and dispose of liquid radioactive wastes generated as a result of normal operations, including AOOs. The applicant provided sufficient design information to demonstrate that the LRWS design is adequate to meet the requirements of 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 20.1406; values in 10 CFR Part 20, Appendix B, Table 2; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR Part 50, Appendix A, GDC 60, 61, and 64; 10 CFR Part 50, Appendix I; 40 CFR Part 190, 10 CFR 52.47(b)(1); 10 CFR 52.137(a)(5), and SRP and DSRS Section 11.2 acceptance criteria. The NRC staff based this conclusion on the following:

- The NuScale design demonstrates compliance with 10 CFR 50.34a, as it relates to the inclusion of sufficient design information and system design features that are necessary for collecting, storing, processing, controlling, and monitoring the safe discharges of liquid wastes. The design conforms to the guidelines of SRP and DSRS Section 11.2.

- The NuScale design meets the requirements of 10 CFR Part 50, Appendix A, GDC 60, with respect to controlling releases of liquid effluents by monitoring LRWS discharges through a single discharge line. An LRWS discharge is automatically isolated upon an alarm caused by a low-dilution flow indication, a low-pressure indication in the discharge pipe annulus, or a high-radiation alarm in a discharge line radiation monitor.

- The LRWS includes adequate processing capability and holdup capacity.

- A COL applicant referencing the NuScale standard design will calculate the liquid effluent doses to members of the public using site-specific parameters and compare those doses with the design objectives in 10 CFR Part 50, Appendix I, to ensure compliance with 10 CFR 20.1302 and 40 CFR Part 190 under COL Item 11.2-1.

- The NuScale design demonstrates compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 61, as it meets the guidelines of RG 1.143, Revision 2, by providing sufficient storage space and treatment capacity to ensure adequate safety under normal operations, AOOs, and postulated accident conditions. The commitment to providing the necessary storage space and treatment capacity fulfills the requirements of 10 CFR 20.1406 and the guidance of RGs 4.21 and 1.143 for minimizing the contamination of the facility and generation of radioactive waste. It also satisfies the concerns of IE Bulletin No. 80-10 in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases to the environment.

- GDC 64 is met for liquid radwaste by providing a liquid effluent radiation monitor to check the effluent discharge from the UWS basin during normal operations and AOOs and by providing other liquid process monitors for various systems and various area and airborne radiation monitors in locations where liquid radwaste leaks may occur.

- The NuScale LRWS design includes features to minimize contamination in accordance with 10 CFR 20.1406. (SER Chapter 12 includes a more detailed evaluation of 10 CFR 20.1406 compliance.)

- A COL applicant referencing the NuScale certified design will perform a site-specific evaluation using the site-specific source term and dilution flow to calculate offsite doses from liquid effluents and ensure compliance with the effluent concentration limits in 10 CFR Part 20, Appendix B, Table 2, Column 2, under COL Item 11.2-3.

- A COL applicant referencing the NuScale standard design will demonstrate compliance with 10 CFR Part 50, Appendix I, Section II.D, for offsite individual and population doses resulting from liquid effluents by preparing a site-specific CBA using the guidance in RG 1.110 under COL Item 11.2-4.

- The NuScale design provides sufficient information and design features to satisfy the guidance of RG 1.143, Revision 2, for radioactive waste processing systems in establishing the seismic and quality group classifications for system components and structures housing LRWS components to support the staff's finding on GDC 60 and 61.

- A COL applicant referencing the NuScale standard design will develop and implement an ODCM that describes the methodology and parameters for calculating offsite doses for gaseous and liquid effluents, using the guidance of NEI 07-09A. The COL applicant is responsible for ensuring that the design objectives in 10 CFR Part 50, Appendix I, and the requirements in 10 CFR 20.1301(e), which incorporates by reference 40 CFR Part 190 for facilities within the nuclear fuel cycle, including nuclear power plants, are satisfied during operation.

- A COL applicant referencing the NuScale standard design will perform a site-specific evaluation for a postulated accidental release from the failure of the only outdoor tank containing radioactive liquids, the PSCS tank, in accordance with BTP 11-6. The COL applicant is responsible for ensuring that the dose limit to members of the public in 10 CFR 20.1302 is satisfied under COL Item 11.2-2.

- The NuScale design provides TS to address the requirements of 10 CFR 50.36a in TS 5.5.1, which include an established, implemented, and maintained ODCM. TS 5.5.2 has provisions for a radioactive effluent control program that conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining ALARA the doses to members of the public from radioactive effluents. TS 5.6.1 and TS 5.6.2 specify annual reporting requirements in describing the results of the radiological monitoring program and summarizing the quantities of radioactive liquid effluents released to the environment. In addition, TS 5.5.6 provides for a surveillance program to ensure that an accidental liquid tank release does not exceed the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2.

- The NuScale design provides ITAAC associated with the liquid radioactive waste system to comply with the requirements of 10 CFR 52.47(b)(1).

- In FSAR Section 11.2, the applicant addresses the liquid effluent aspects of 10 CFR 52.137(a)(5), because the applicant identifies the means for controlling and limiting liquid radioactive effluents within the limits set forth in 10 CFR Part 20.

11.3 Gaseous Waste Management System

11.3.1 Introduction

The GRWS is designed to process the gaseous waste stream from the LRWS degasifier and the CES, provide holdup for radioactive decay of xenon and krypton, and convey the gaseous effluent to the RWB heating, ventilation, and air conditioning (HVAC) system (RWBVS), which transports the effluent to the RXB HVAC system (RBVS) for monitoring and release. The GRWS filters out particulate carryover and delays the noble gases through activated charcoal beds until they have decayed sufficiently to allow release to the environment.

Primary gaseous effluent sources include the LRWS degasifier, the CES (FSAR Section 9.3.6, “Containment Evacuation System”), the RWBVS (FSAR Section 9.4.3, “Radioactive Waste Building Ventilation System”), and other sources exhausted by the RBVS (FSAR Section 9.4.2, “Reactor Building and Spent Fuel Pool Area Ventilation System”). In addition, small releases that occur in the turbine generator building from the main condenser air removal system (CARS) (FSAR Section 10.4.2, “Condenser Air Removal System”) and turbine gland sealing system (FSAR Section 10.4.3, Turbine Gland Sealing System”) are monitored but directly released to the environment.

11.3.2 Summary of Application

SDAA Part 8: SDAA Part 8, Section 3.12, includes the information associated with this section.

SDAA Part 8, Section 3.14, provides the ITAAC associated with the design of components. The ITAAC verify that components that process gaseous radioactive waste, including components associated with the LRWS degasifier that are classified RW-IIa in RG 1.143, are constructed to RW-IIa design criteria. This conforms to the information and conclusions made for RWMS categories for SSCs in the RWB and as discussed in FSAR Sections 11.2 and 11.3.

FSAR Section 11.3: The applicant described the system in FSAR Section 11.3, summarized as follows.

In FSAR Section 11.3, the applicant described the design of the GRWS and its functions in monitoring, controlling, collecting, processing, handling, storing, and disposing of gaseous radioactive waste generated as the result of normal operation and AOOs. The GRWS is a non-safety-related system and serves no safety function. A failure of the GRWS does not compromise safety-related systems or components and does not prevent the safe shutdown of the plant. The portions of the RWB that house the GRWS are designed to seismic Category RW-IIa requirements in accordance with RG 1.143. SER Section 3.2 includes a detailed evaluation of the seismic design. FSAR Table 11.3-3, “Gaseous Radioactive Waste System Equipment Malfunction Analysis,” describes the failure scenarios considered for the GRWS. FSAR Tables 11.3-2, “Major Equipment Design Parameters,” and 11.3-10, “Classification of Structures, Systems, and Components,” provide the RG 1.143 classification of the GRWS components, as well as additional information related to component classifications and design parameters. The decay and guard beds are classified as RW-IIa up to and including the nearest isolation valves.

FSAR Figure 11.3-1, “Gaseous Radioactive Waste System Diagram,” presents a diagram of the GRWS. FSAR Table 11.3-1, “Gaseous Radioactive Waste System Design Parameters,” lists various system nominal values, including element delay times, adsorption coefficients, temperatures, and flow rates. FSAR Table 11.3-2 lists information relating to, but not limited to, design pressures, design temperatures, flow rates, and component materials for the following system components: vapor condenser package assembly, charcoal drying heater, charcoal guard bed, and charcoal decay bed vessel.

The GRWS is a passive, once-through, ambient temperature charcoal delay system that receives hydrogen-bearing gas containing fission gases from the LRWS degasifier. The GRWS also receives gaseous waste inputs from the individual NuScale Power Modules through the CES if high radiation is detected in the CES exhaust. The GRWS filters particulate carryover, removes moisture, delays the gas to allow radioactive decay, and conveys it to the RBVS through the RWBVS for release to the environment from the plant exhaust stack as a monitored release.

The key components of the GRWS, described in FSAR Section 11.3.2.1, “Component Description,” include the following:

- two waste gas coolers
- two moisture separators
- one charcoal guard bed
- two charcoal decay beds (each consisting of four charcoal vessels connected in series)
- one charcoal drying heater
- three oxygen analyzers
- two hydrogen analyzers

The waste gas input from the liquid radioactive waste degasifier (and potentially the CES) is diluted automatically with nitrogen to maintain a hydrogen concentration of less than 4 percent. Because the waste gas input flow is not constant, nitrogen is supplied to maintain a positive GRWS pressure and a constant flow. The waste gas input into the GRWS passes through a vapor condenser package assembly that contains a waste gas cooler (cooled by chilled water) and a moisture separator. The moisture separator includes level control drain valves piped to the equipment drain sump in the RWDS. The drain line passes through a drain trap to prevent radioactive gas from passing to the RWDS in the event of a system failure. After the vapor condenser, the waste gas stream passes through two redundant oxygen analyzers, two hydrogen analyzers, and a manual sample port. If high oxygen levels are detected, the inlet stream to the GRWS is automatically isolated, and a nitrogen purge flushes the GRWS. Termination of nitrogen flushing and restart of normal operations are manually initiated.

The waste gas then passes through a charcoal guard bed located in an ambient-temperature-controlled shielded cubicle. Because the guard bed is at ambient room temperature, it warms the gas from the gas cooler (lowering its relative humidity) to improve fission gas capture efficiency in the decay beds. The guard bed also acts as a backup moisture removal device. The guard bed includes a safety-relief valve, differential pressure instrumentation, and a means to dry or replace charcoal. Charcoal drying is manually initiated by remotely operated valves and a normally deenergized charcoal drying heater, which provide a heated nitrogen flow to the guard bed. The heated, moisture-laden nitrogen is recycled back to the inlet of the vapor condenser. The guard bed also contains a fire sensor that automatically activates a nitrogen purge upon detecting a fire.

The conditioned waste gas then flows into either one of the two charcoal decay beds, each consisting of four charcoal vessels connected in series. Entrance into the first vessel and exit from the last vessel is through the top of the vessel to minimize the potential of charcoal loss. Each decay bed contains activated charcoal optimized for xenon and krypton retention. Like the guard bed, the decay beds contain differential pressure instrumentation, fire detection instrumentation, safety-relief valves, and the ability to either dry or replace charcoal. In addition, the decay beds contain radiation monitors that automatically isolate flow in the event of a high-radiation indication.

Finally, the processed waste gas is released to the RWBVS, which interfaces with the RBVS that provides the monitored effluent path to the environment. The GRWS outlet also has an offline radiation monitor that can take samples before the waste gas is sent to the ventilation systems.

In assessing the radiological impacts associated with radioactive gaseous effluent discharges, FSAR Table 11.3-4, “Gaseous Effluent Release Calculational Inputs,” and Table 11.3-6, “GASPAR Code Input Parameter Values,” present information supporting the development of the gaseous effluent source term, as well as compliance with (1) the effluent concentration limits of 10 CFR Part 20, Appendix B, Table 2, Column 1, (2) 10 CFR 20.1301(e) in meeting the EPA environmental radiation protection standards of 40 CFR Part 190, and (3) the numerical guides and design objectives of 10 CFR Part 50, Appendix I.

The applicant’s results show the expected annual releases of airborne radioactivity and gaseous effluent concentrations in unrestricted areas and confirm that gaseous effluent doses to members of the public comply with the NRC regulations. The applicant’s results also demonstrate compliance with the ALARA requirements of 10 CFR Part 50, Appendix I, and the acceptance criteria in SRP Section 11.3, “Gaseous Waste Management System,” for evaluating a postulated leak of radioactivity in BTP 11-5, Revision 4, “Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure,” issued January 2016 (ML15027A302), from a GRWS component containing radioactivity.

Technical Specifications: TS 5.5.1 and 5.5.2 are associated with the GRWS and are found in SDAA Part 4, Volume 1. The TS also include the following reports: 5.5.6, “Explosive Gas and Storage Tank Radioactivity Monitoring”; 5.6.1, “Annual Radiological Environmental Operating Report”; and 5.6.2, “Radioactive Effluent Release Report.”

Technical Reports: NuScale TR-123242, Revision 1 is associated with this area of review.

Topical Reports: No topical reports are associated with this area of review.

11.3.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for the GRWS area of review, associated acceptance criteria, and review interfaces with other SRP sections appear in SRP Section 11.3. The following summarizes the regulatory requirements:

- 10 CFR 20.1301, as it relates to dose limits for individual members of the public

- 10 CFR 20.1302, as it relates to limits on doses to members of the public and gaseous effluent concentrations and doses in unrestricted areas

- 10 CFR 20.1406, as it relates to facility design and operational procedures for minimizing facility contamination and the generation of radioactive waste

- 10 CFR Part 20, Appendix B, Table 2, Column 1, as it relates to the airborne (gaseous) effluent concentration limits for release to the environment

- 10 CFR 50.34a, as it relates to the inclusion of sufficient design information to demonstrate compliance with the design objectives for equipment necessary to control releases of radioactive gaseous effluents to the environment

- 10 CFR 50.36a, as it relates to TS requiring that operating procedures be developed for radiological monitoring and sampling equipment as part of the administrative controls and surveillance on effluent controls in meeting the ALARA criterion and 10 CFR 20.1301 dose limits

- 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, as they relate to numerical guidelines and design objectives and limiting conditions for operation in meeting dose criteria and the ALARA criterion in Appendix I

- GDC 60, as it relates to the design of the GRWS to control releases of gaseous radioactive effluents

- GDC 61, as it relates to the design of the GRWS to ensure adequate safety under normal operations and postulated accident conditions

- GDC 64, as it relates to the design of the GRWS to monitor for radioactivity that may be released from normal operations, including AOOs and postulated accidents

- GDC 3, “Fire protection,” as it relates to the design of gaseous waste-handling and treatment systems to minimize the effects of explosive mixtures of hydrogen and oxygen

- 40 CFR Part 190 (EPA’s generally applicable environmental radiation standards), as implemented under 10 CFR 20.1301(e), as it relates to controlling doses within the EPA’s generally applicable environmental radiation standards

- 10 CFR 52.47(b)(1), which requires that applications for DCs contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC and provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations

- 10 CFR 52.137(a)(5), as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits established in 10 CFR Part 20

The following documents contain the regulatory positions and guidance for meeting the relevant requirements of the regulations identified above:

- RG 1.109, Revision 1, as it relates to demonstrating compliance with the numerical guidelines for dose design objectives and the ALARA criterion of 10 CFR Part 50, Appendix I

- RG 1.110, Revision 1, as it relates to performing a CBA for reducing cumulative doses to populations by using available technology

- RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," issued July 1977 (ML003740354), as it relates to the modeling and derivations of atmospheric dispersion and deposition parameters in demonstrating compliance with the numerical guidelines and ALARA criterion of 10 CFR Part 50, Appendix I

- RG 1.112, Revision 1, as it relates to the acceptable methods for calculating annual average releases of radioactivity in effluents.

- RG 1.206, Revision 1, as it relates to the minimum information requirements specified in 10 CFR 52.79 to be submitted in a COL application

- RG 1.140, Revision 2, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," issued June 2001 (ML011710150), as it relates to the design, testing, and maintenance of normal ventilation exhaust systems at nuclear power plants

- RG 1.143, Revision 2, as it relates to the seismic design and quality group classification of components used in the GRWS and the structures housing this system, as well as provisions used to control leakage

- RG 1.33, Revision 2, as it relates to QA for the operation of the GRWS provisions for the sampling and monitoring of radioactive materials in process and effluent streams and control of radioactive effluent releases to the environment

- RG 4.21, Revision 0, as it relates to minimizing both the contamination of equipment, plant facilities, and the environment and the generation of radioactive waste during plant operation

- BTP 11-5, Revision 3, as it relates to the assessment of radiological impacts associated with the failure of a GRWS component

- NUREG-0017, Revision 1, as it relates to the methodology to calculate gaseous and liquid effluent releases

- NUREG-1301, Revision 0, as it relates to ODCM guidance for PWR plants

- NUREG/CR-4653, "GASPAR II—Technical Reference and User Guide," issued March 1987, as it relates to the methodology to calculate gaseous effluent doses

- IE Bulletin No. 80-10, as it relates to methods and procedures used in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled releases of radioactivity

- ANSI/ANS-18.1-1999, as it relates to the methodology for determining the source term for normal reactor operations, including AOOs

- NEI 08-08A, Revision 0, and RG 4.21, as they relate to acceptable levels of detail and content needed to demonstrate compliance with the programmatic elements of 10 CFR 20.1406

- GL 89-01, Supplement No. 1, as it relates to an operational program that addresses the development of a site-specific radiological environmental monitoring program

11.3.4 Technical Evaluation

The NRC staff evaluated the information in FSAR Section 11.3 against the applicable NRC regulations and guidance in SRP Section 11.3 and DSRS Section 11.3, “Gaseous Waste Management System,” issued June 2016 (ML15355A335).

11.3.4.1 Design Considerations

11.3.4.1.1 General Design Criteria 3, 60, 61, and 64 and 10 CFR 50.34a

The applicant must meet the requirements of GDC 60 and 61 with respect to controlling releases of radioactive materials to the environment. GDC 60 requires that the nuclear power unit design include provisions to handle radioactive wastes produced during normal reactor operations, including AOOs. GDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions. GDC 64 requires that the GRWS be designed to monitor radiation levels and radioactivity in effluents, as well as radioactive leakages and spills, during routine operations, including AOOs. GDC 3 requires the system to be analyzed to minimize the effects of explosive gas mixtures of hydrogen and oxygen.

The NRC staff considered the ability of the proposed GRWS to meet the demands of the plant resulting from AOOs and has concluded that the system capacity and design are adequate to meet the anticipated needs of the plant. The applicant met the requirements of GDC 60 and 61 with respect to controlling releases through its description of the charcoal delay beds and systems for treating gaseous input to protect the charcoal decay beds. The applicant also discussed the retention ability of the beds for krypton and xenon gases. GDC 64 is met for gaseous radioactivity releases because the design provides gaseous effluent radiation monitors to check the effluent discharges from the RBVS, CARS exhaust, turbine gland sealing system exhaust, PSCS, and auxiliary boiler system during normal operations and AOOs.

In assessing compliance with GDC 60 and 61, the NRC staff reviewed the QA provisions and guidance in RG 1.143, Revision 2, specified by the applicant in the FSAR. The applicant stated that the GRWS will conform to RG 1.143, Revision 2. FSAR Table 11.3-2 identifies the safety class for components of the GRWS. The safety classifications are based on the guidance in RG 1.143, Revision 2, which specifies that the safety classifications of radwaste system components should be based on the A_1 and A_2 values in 10 CFR Part 71, Appendix A, if following the guidance of RG 1.143, Revision 2. The applicant defined the boundaries of the radwaste systems classifications as those components that include connected piping and/or other equipment, up to and including the system isolation valves. This definition is consistent with system boundaries reviewed in the past, and the NRC staff finds this boundary definition acceptable.

The NRC staff performed confirmatory calculations of the radiation source terms and system information and determined that the applicant had correctly calculated radwaste system seismic categories. The NRC staff's confirmatory calculations were performed using the appropriate source terms for the equipment identified in FSAR Table 11.3-2. The NRC staff verified the listed RG 1.143 safety classification category for the equipment included in this table. The NRC staff determined that the applicant has appropriately classified the GRWS components using the guidance contained in RG 1.143 to satisfy requirements in GDC 60, 61, and 64. The staff's review finds that the applicant has followed the guidance contained in RG 1.143 by defining component source terms and correctly determining the radwaste safety classification. As a result, the staff finds that the applicant conformed to GDC 61 as it relates to the design of the GRWS SSCs.

For compliance with 10 CFR 50.34a, the applicant must provide sufficient information to demonstrate that the design objectives of equipment necessary to treat and control releases of radioactive effluents into the environment have been met. The requirements of 10 CFR 50.34a related to describing the gaseous radwaste system are met by describing the process used to treat and handle gaseous waste and by identifying the system boundaries. The remainder of 10 CFR 50.34a is addressed by meeting the ALARA design objective for gaseous effluents, which is discussed in the section below on meeting the Appendix I, design objectives. The applicant described the GRWS in FSAR Section 11.3.2, "System Description," including the planned pathways for a release to the environment. SER Section 11.5 discusses radiation monitoring to limit or control releases to the environment. The applicant also provided FSAR Table 11.3-5, "Gaseous Estimated Discharge for Normal Effluents," to summarize the releases to the environment that meet the requirements to quantify each of the principal radionuclides of the gases, halides, and particulates expected to be released annually to unrestricted areas in gaseous effluents produced during normal reactor operations. The next section discusses the staff's evaluation of the effluent release results.

The NuScale design demonstrates compliance with the GDC 3 requirements for the design of the gaseous waste-handling and treatment systems to minimize the generation of explosive gas mixtures, prevent fires and explosions, and include design features to mitigate a fire if one were to occur, despite the features in place to prevent one from occurring. Nitrogen from the nitrogen distribution system dilutes the waste gas input from the degasifier and CES to maintain a hydrogen concentration of less than 4 percent, which is less than the flammability and explosive concentration range for hydrogen. In addition, after the vapor condenser, the waste gas stream passes through two oxygen analyzers, two hydrogen analyzers, and a manual sample port. If high oxygen is detected, the inlet stream to the GRWS automatically isolates and a nitrogen purge flushes the GRWS. In addition, the guard bed includes a fire detector that automatically activates a nitrogen purge upon detecting a fire. Finally, the staff notes that NuScale referenced the guidance contained in RG 1.189, "Fire Protection for Nuclear Power Plants," as it relates to the conduct of fire hazards analysis involving the presence of combustible gases and inflammable materials. As a result, the staff finds that the fire and explosive gas controls for the GRWS are sufficient within the context of the gaseous radwaste system design and controlling releases from the gaseous radwaste system. (SER Chapter 9 provides a holistic review of fire protection for the NuScale design.)

11.3.4.1.2 10 CFR Part 50, Appendix I, Gaseous Effluent Doses

The NRC staff reviewed the GRWS to determine whether it complies with the requirements of 10 CFR Part 50, Appendix I. The applicant calculated the gaseous effluent release doses using the NRC-endorsed GASPAR II computer code, approved for use by the NRC. Using the information in the provided tables and GASPAR II input and output files, the NRC staff performed a confirmatory analysis of the gaseous effluent release doses in FSAR Table 11.3-7, “Gaseous Effluent Dose Results for 10 CFR 50 Appendix I,” and found them acceptable.

The GRWS is designed to process the gaseous waste stream from the LRWS degasifier and the CES, provide holdup for radioactive decay of xenon and krypton, and transport the gaseous effluent to the RWBVS, which transports the effluent to the RBVS for monitoring and release. The charcoal delay system is in accordance with NUREG-0017, Revision 1. Discharge from the liquid radioactive waste degasifier to the GRWS is initially diluted by nitrogen flow in the GRWS to maintain a hydrogen concentration of less than 4 percent. The gaseous waste is processed by the charcoal guard bed and delay beds and then goes to the plant exhaust stack in the RBVS. The RBVS exhaust stack combines and monitors exhaust flow from the RWBVS and RBVS before releasing the effluents to the environment. Primary gaseous effluent sources, besides gaseous radioactive waste, include the CES, RWBVS, and others exhausted by the RBVS. In addition, multiple release points and effluent releases occur from the turbine generator building, the CARS, and the turbine gland sealing system. These releases are quantified and monitored, but they are directly discharged to the environment. The NRC determined that the description of the GRWS is acceptable and used the applicant’s parameters to perform a confirmatory calculation of the gaseous effluent release doses based on the plant design.

In FSAR Table 11.3-6, “GASPAR Code Input Parameter Values,” the applicant summarized the inputs used for the GASPAR II code. This table includes a pointer to FSAR Table 11.3-5, for the Ci/yr source term required for the analysis. The NRC staff confirmed this source term and the results produced by NuScale’s effluent release methodology, as discussed in SER Section 11.1. The NRC staff confirmed the use of the RG 1.109 default values and assumed parameters for the NuScale design, as referenced in FSAR Table 11.3-6, for its confirmatory calculation. In addition to the values in RG 1.109, the applicant referenced the routine release (X/Q) and relative deposition factor (D/Q) values from FSAR Table 2.0-1, “Site Parameters.”

To confirm the applicant’s results in FSAR Table 11.3-7, the NRC staff used the GASPAR II inputs reported in FSAR Table 11.3-6. The results of the NRC staff’s confirmatory calculation verified the estimated gamma air dose, beta air dose, total body dose, skin dose, and the maximum organ doses. SER Table 11.3-1 compares the applicant’s and NRC staff’s results to the design objectives in 10 CFR Part 50, Appendix I.

Table 11.3-1 Comparison of the Applicant’s and the NRC’s Estimated Annual Individual Doses from Gaseous Effluent Releases

Pathway	Applicant Results	NRC Staff Results	Design Objectives
Gamma Air	2.0×10^{-4} mGy (2.0×10^{-2} mrad)	2.0×10^{-4} mGy (2.0×10^{-2} mrad)	1×10^{-1} mGy (10 mrad)
Beta Air	1.0×10^{-3} mGy (1.0×10^{-1} mrad)	1.0×10^{-3} mGy (1.0×10^{-1} mrad)	2×10^{-1} mGy (20 mrad)
Total Body	2.8×10^{-3} mSv (2.8×10^{-1} mrem)	2.8×10^{-3} mSv (2.8×10^{-1} mrem)	5×10^{-2} mSv (5 mrem)
Skin	4.0×10^{-3} mSv (4.0×10^{-1} mrem)	3.9×10^{-3} mSv (3.9×10^{-1} mrem)	1.5×10^{-1} mSv (15 mrem)

Pathway	Applicant Results	NRC Staff Results	Design Objectives
Max Organ (Thyroid)	1.80×10^{-2} mSv (1.80 mrem) (maximum dose is to child and infant)	1.89×10^{-2} mSv (1.89 mrem) (maximum dose calculated is to child)	1.5×10^{-1} mSv (15 mrem)

The NRC staff's review determined that the applicant had accurately calculated the gaseous effluent release doses and that the results are within the ALARA design objectives in 10 CFR Part 50, Appendix I. The NRC staff's confirmatory calculations demonstrate that the reported FSAR values are acceptable. The results are below the limits in 10 CFR Part 50, Appendix I, and thus are acceptable.

The NRC staff's determination is based on the use of non-site-specific data for the analyses. Presently, the applicant uses conservative estimates in its GASPAR II analysis to show the bounding results for gaseous releases at any chosen site. A COL applicant that references the NuScale Power Plant SDA will perform a site-specific evaluation using the site-specific meteorological data. The applicant included COL Item 11.3-2, which calls for a COL applicant to calculate these doses to a member of the public using the guidance of RG 1.109 and RG 1.111. An applicant following the COL item will use site-specific parameters to develop gaseous effluent releases based on any specific site. This COL item is consistent with SRP guidance, and the NRC staff finds this COL item acceptable since a COL applicant will develop a site-specific dose analysis that complies with 10 CFR Part 50, Appendix I.

11.3.4.1.3 Site-Specific Cost-Benefit Analysis

FSAR Section 11.3.2.4, "Site-Specific Cost-Benefit Analysis," provides COL Item 11.3-1. COL Item 11.3-1 states that an applicant that references the NuScale Power Plant US460 standard design will perform a site-specific CBA using the guidance of RG 1.110 to demonstrate compliance with 10 CFR Part 50, Appendix I, Section II.D. RG 1.110 describes an acceptable method of performing a CBA to show that the GRWS design includes all items of reasonably demonstrated technology for reducing to ALARA levels each reactor's cumulative population doses from releases of radioactive materials. The CBA requires site-specific information, which is outside the scope of the requested SDA. The COL applicant will provide the site-specific CBA to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix I, II.D, under COL Item 11.3-1. The NRC staff finds the inclusion of COL Item 11.3-1 acceptable.

11.3.4.1.4 10 CFR Part 20, Appendix B, Gaseous Effluent Concentration Limits

FSAR Table 11.3-5 shows that the sum of the fractions (i.e., unity rule calculation) determined from summing the ratio of the assumed discharge concentration and respective effluent concentration limit for each radionuclide released in the gaseous effluent met the unity rule calculation specified in Note 4 of 10 CFR Part 20. The NRC staff finds that the gaseous effluent site concentrations are below the limits in 10 CFR Part 20, Appendix B, Table 2, Column 1, and the unity calculation described in Note 4. The NRC staff performed a confirmatory analysis of the results presented in FSAR Table 11.3-5. The results of the NRC staff's confirmatory analysis determined the information contained in FSAR Table 11.3-5 is acceptable, and the applicant has demonstrated compliance with the 10 CFR Part 20, Appendix B, effluent concentration limits.

11.3.4.1.5 10 CFR 20.1301(e), Compliance with 40 CFR Part 190

The NRC staff reviewed the applicant's GASPAR II code input and output files evaluating the dose from gaseous effluent releases to members of the public in unrestricted areas for compliance with 10 CFR Part 50, Appendix I; 10 CFR 20.1302; and 40 CFR Part 190. Input values pertaining to environmental characteristics, such as meteorology, release rates, and exposure pathways, rely on site-specific information addressed by the applicant in COL Item 11.3-2.

Under COL Item 11.3-2 in FSAR Section 11.3.3, "Radioactive Effluent Releases," the COL applicant will calculate doses to members of the public using site-specific parameters following the guidance of RG 1.109 and RG 1.113 and compare doses from gaseous effluents in 10 CFR Part 50, Appendix I; 10 CFR 20.1302; and 40 CFR Part 190. The NRC staff finds that, because the site-specific input parameter values used in the GASPAR II code calculation of gaseous effluent doses are outside the scope of the requested SDA, the inclusion of COL Item 11.3-2 is acceptable.

11.3.4.1.6 Minimization of Contamination, 10 CFR 20.1406

The NRC staff reviewed the information in FSAR Sections 11.3 and 12.3.6 against the criteria in 10 CFR 20.1406 for minimizing contamination. In 10 CFR 20.1406, the NRC requires, in part, that design features and procedures 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. In FSAR Section 12.3.6, the applicant described the use of RG 4.21 in meeting the requirements of 10 CFR 20.1406. RG 4.21 describes an acceptable method for implementing a program to meet the 10 CFR 20.1406 requirements listed above.

The tables in FSAR Section 12.3 provide information on how the design of the GRWS includes provisions to minimize the contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste. Information in the tables in FSAR Section 12.3 includes the use of low-leakage valves, radiation monitoring to detect leakage, and selection of components for a 60-year operating life and incorporating easily removable components.

FSAR Section 12.3.1.1.2, "Valves" indicates that double isolation valves are used at the interface between contaminated and non-contaminated systems to prevent cross contamination and that valves are designed to fail to the safe position. In addition, the NuScale design includes various radiation monitors that would identify unexpected gaseous leaks to non-radioactive systems, such as radiation monitoring in the auxiliary boiler system. These design features minimize contamination in accordance with 10 CFR 20.1406 and satisfy IE Bulletin No. 80-10 in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases to the environment.

FSAR Section 12.3.6 and SER Section 12.3 provide additional information on compliance with 10 CFR 20.1406, including programmatic requirements to be addressed by a COL applicant.

11.3.4.1.7 Mobile or Temporary Equipment

The GRWS does not include the use of mobile or temporary equipment in the design.

11.3.4.1.8 Postulated Gaseous Effluent Radioactive Releases from a Waste Gas System Leak or Failure

The NRC staff reviewed FSAR Section 11.3.3.1, “Radioactive Effluent Releases and Dose Calculation due to Gaseous Radioactive Waste System Leak or Failure,” which refers to the analysis to support the calculations in BTP 11-5. Failure of the GRWS is postulated, which results in a release of gaseous radionuclides. The analysis of a GRWS leak or failure follows the guidance of BTP 11-5 and demonstrates compliance with regulatory limits. The applicant provided a list of input parameters found in FSAR Table 11.3-8, “Gaseous Effluent Dose Evaluation for Gaseous Radioactive Waste System Failure,” which describes the release source term, the assumed dispersion factor, and the anticipated offsite dose consequence. FSAR Section 11.3.3.1 states that the assumed source term is based on a 1 percent failed fuel fraction and provides additional information on the Table 11.3-8 source term and the waste gas system failure analysis. In its review of this information, the staff confirmed that the applicant provided the necessary source term information to perform the calculation, as described by the guidance in BTP 11-5. This information allowed the NRC staff to perform a confirmatory calculation, which determined that the calculated exclusion area boundary dose was 5.97×10^{-2} millisieverts (mSv) (5.97 millirem (mrem)). The applicant stated that its expected offsite dose consequence is less than 1×10^{-1} mSv (10 mrem). The NRC staff reviewed COL Item 11.3-3, which specifies that a COL applicant perform this analysis using the guidance in BTP 11-5 and site-specific parameters. Consistent with COL Item 11.3-3, the staff finds that the applicant has provided the source term necessary for a COL applicant to verify the dose analysis using site-specific information.

11.3.4.1.9 Technical Specifications

In SDAA Part 4, TS 5.5.6 requires a program to control levels of potentially explosive gas mixtures in the GRWS and limit the quantity of radioactivity contained in gas delay beds, such that offsite doses would not exceed 1 mSv (100 mrem) in the event of a bed failure. Additionally, in SDAA Part 4, TS 5.5.1 and TS 5.5.4, “Steam Generator Program,” have provisions for direction in managing releases of radioactive effluents. TS 5.6.1 and TS 5.6.2 specify annual reporting requirements for the submission of the results of the radiological monitoring program and summaries of the quantities of radioactive gaseous effluents released in the environment. As stated in TS 5.5.1, changes to the ODCM initiated by the COL applicant must be justified by calculation, and these changes must keep levels of effluent radioactivity that meet the requirements of 10 CFR 20.1302; 40 CFR Part 190; 10 CFR 50.36a; and 10 CFR Part 50, Appendix I.

The TS would address the radioactive effluent controls program, which is described in the ODCM, to include instrumentation to monitor and control gaseous effluent discharges; meet limits on effluent concentrations released to unrestricted areas; monitor, sample, and analyze gaseous effluents before and during releases; set limits on annual and quarterly dose commitments to a member of the public; and assess cumulative doses from radioactive gaseous effluents. The NRC staff determined that these requirements are acceptable and agreed that further implementation of such programs will be addressed in a plant and site-specific ODCM. SER Section 11.5 contains the NRC staff’s evaluation of the applicant’s proposed ODCM.

11.3.5 Combined License Information Items

SER Table 11.3-2 lists COL item numbers and descriptions related to the GRWS from FSAR Table 1.8-1, “Combined License Information Items.”

Table 11.3-2 NuScale COL Items for SDAA Part 2, Section 11.3

COL Item No.	COL Item Description	FSAR Section No.
11.3-1	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific cost-benefit analysis using the guidance in Regulatory Guide 1.110.	11.3.2.4
11.3-2	An applicant that references the NuScale Power Plant US460 standard design will calculate doses to members of the public using the site-specific parameters, compare those gaseous effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.3.3
11.3-3	An applicant that references the NuScale Power Plant US460 standard design will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.	11.3.3.1

The NRC staff finds that the list adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items to be included in the FSAR for the GRWS.

11.3.6 Conclusion

The NRC staff concludes that the GRWS, as a shared system, includes the equipment necessary to collect, process, hold for decay, and control releases of radioactive materials in gaseous effluents generated by normal operations, including AOOs. The applicant provided sufficient design information to demonstrate that the GRWS design is adequate to meet the requirements of 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 20.1406; 10 CFR Part 20, Appendix B, Table 2; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR Part 50, Appendix A, GDC 3, 60, 61, and 64; 10 CFR Part 50, Appendix I; 40 CFR 190; 10 CFR 52.47(b)(1); 10 CFR 52.137(a)(5); and SRP and DSRS Section 11.3 acceptance criteria. This conclusion is based on the following:

- The NuScale design demonstrates compliance with 10 CFR 50.34a, as it relates to the inclusion of sufficient design information and system design features that are necessary for collecting, processing, holding for radioactive decay, controlling, and monitoring safe discharges of gaseous wastes. The design conforms to the guidelines of SRP and DSRS Section 11.3.

- The NuScale design demonstrates compliance with the requirements of GDC 61, using the guidelines of RG 1.143, Revision 2, by providing sufficient treatment capacity, retention in charcoal delay beds, and holdup for radioactive decay to ensure adequate safety under normal operations, AOOs, and postulated accident conditions. It also addresses the concerns of IE Bulletin No. 80-10 in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled radioactive releases to the environment.

- GDC 64 is met for gaseous radwaste by providing a gaseous effluent radiation monitor at potential gaseous effluent discharge paths, other gaseous radioactive material process monitors for various systems, and various area and airborne radiation monitors in locations where liquid radwaste leaks may occur.

- The NuScale design meets the requirements of GDC 60, with respect to controlling releases of gaseous effluents by monitoring radioactive gas discharges through the plant exhaust stack through the RWBVS. Gaseous effluent releases are continuously monitored by an offline radiation monitor and integrated sampling system that measures and records exhaust stack flow, particulate, iodine, and noble gases. In addition, if high radiation is detected in the GRWS connection line to the RWBVS, the GRWS outlet valve is closed to stop system flow to the RWBVS.

- The gaseous effluent site concentrations tabulated in FSAR Table 11.3-5 are below the limits in 10 CFR Part 20, Appendix B, Table 2, Column 1.

- The NuScale design demonstrates compliance with 10 CFR Part 50, Appendix A, GDC 3, as it relates to the design of gaseous waste-handling and treatment systems to minimize the generation of explosive gas mixtures and the effects of explosive mixtures of hydrogen and oxygen on subsystems and components using RG 1.189, as it relates to the conduct of fire hazards analysis involving the presence of combustible gases and inflammable materials.

- The NuScale design provides sufficient information and design features, satisfying the guidance of RG 1.143, Revision 2, for radioactive waste processing systems in establishing the seismic and quality group classifications for system components and structures housing components.

- The NuScale GRWS design includes design features to minimize contamination in accordance with 10 CFR 20.1406. (SER Chapter 12 includes a more detailed evaluation of 10 CFR 20.1406 compliance.) The COL applicant will provide the program necessary to meet 10 CFR 20.1406, as discussed in FSAR Chapter 12.

- A COL applicant referencing the NuScale design will develop and implement an ODCM that describes the methodology and parameters used for calculating offsite doses for gaseous and liquid effluents, using the guidance of NEI 07-09A. The COL applicant is responsible for ensuring that the design objectives in 10 CFR Part 50, Appendix I, and the requirements in 10 CFR 20.1301(e), which incorporates by reference 40 CFR Part 190 for facilities within the nuclear fuel cycle, are satisfied during operation.

- A COL applicant referencing the NuScale design will demonstrate compliance with 10 CFR Part 50, Appendix I, Section II.D, for offsite individual and population doses resulting from gaseous effluents by preparing a site-specific CBA using the guidance in RG 1.110 under COL Item 11.3-1.

- A COL applicant referencing the NuScale design will calculate the gaseous effluent doses to members of the public using site-specific parameters and compare the gaseous effluent doses with the design objectives in 10 CFR Part 50, Appendix I, to ensure compliance with 10 CFR 20.1302 and 10 CFR 20 1301(e), which incorporate by reference 40 CFR Part 190 under COL Item 11.3-2. This ensures compliance with the public dose limits in 10 CFR 20.1301.

- A COL applicant referencing the NuScale design will perform a site-specific analysis for a postulated accidental release of gaseous radionuclides from failure of the GRWS, in accordance with the guidance in BTP 11-5.

- The NuScale design includes TS to address the requirements of 10 CFR 50.36a in providing TS 5.5.1, which requires an established, implemented, and maintained ODCM. TS 5.5.2 has provisions for a radioactive effluent control program that conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining ALARA the doses to members of the public from radioactive effluents. TS 5.6.1 and TS 5.6.2 specify annual reporting requirements in describing the results of the radiological monitoring program and provide summaries of the quantities of radioactive gaseous effluents released to the environment. In addition, TS 5.5.6 provides a surveillance program to ensure that explosive gas concentrations in the gaseous radwaste system will not be reached and that the quantities of radioactive material in the waste gas system will not exceed dose limits in the event of an uncontrolled release of the tank contents.

- The NuScale design provides ITAAC associated with the gaseous radioactive waste system to comply with the requirements of 10 CFR 52.47(b)(1).

- In FSAR Section 11.3, the applicant addresses the gaseous effluent aspects of 10 CFR 52.137(a)(5), because the applicant identifies the means for controlling and limiting gaseous radioactive effluents within the limits set forth in 10 CFR Part 20.

11.4 Solid Waste Management System

11.4.1 Introduction

The SRWS is designed to process both wet and dry solid waste from various plant systems produced during normal operation and AOOs, including startup, shutdown, and refueling operations. Solid radioactive waste is dewatered, decontaminated, surveyed, sorted, and classified for solid waste storage and eventual shipment to licensed offsite facilities, as appropriate.

11.4.2 Summary of Application

SDAA Part 8: SDAA Part 8, Section 3.12, provides the ITAAC associated with the design of the RWB. The ITAAC verify that the below-grade and above-grade portions of the building used for storage and processing of radioactive waste are built to the criteria in RG 1.143 for radwaste management facilities classification RW-IIa. This is based on the radionuclide contents of SSCs in the RWB and as discussed in FSAR Sections 11.2, 11.3, and 11.4 and FSAR Chapter 12.

FSAR Section 11.4: The applicant described the solid waste management system in FSAR Section 11.4, summarized as follows.

The SRWS is located in the RWB. The portions of the building below-grade and areas designated for storage and processing of radioactive waste are classified as RW-IIa (see FSAR Section 3.2.1.4, “Safety Classification RW-IIa”) and have adequate space for onsite storage of various solid waste containers in waste storage areas in the RWB. The SRWS includes the wet solid waste (WSW) system and dry solid waste (DSW) system. There are also areas for packaging, storing, and shipping waste, including mixed wastes.

The SRWS is a non-safety-related system and has no safety-related functions. The SRWS is designed to do the following:

- Collect, process, sample, package, and store WSW generated from the chemical and volume control system, pool cleanup system, and LRWS, using permanently installed equipment.

- Collect, segregate, sample, package, and store compactible and noncompactible DSW.

- Collect, sample, segregate, package, and ship mixed and oily wastes.

- Provide sufficient storage space for packaged solid wastes.

- Process and package waste into disposal containers that are approved by the U.S. Department of Transportation and are acceptable to licensed waste disposal facilities for offsite shipment and burial.

- Meet Federal regulations and protect workers and the general public from radiation by maintaining dose levels ALARA.

- Transfer liquid wastes to the RWDS or LRWS.

FSAR Table 11.4-1, "List of Systems, Structures, and Components Design Parameters," describes design information related to the two spent resin storage tanks (SRSTs), the two-phase separator tanks (PSTs), the two SRST transfer pumps, the two PST transfer pumps, the dewatering skid, and the compactor.

The SRWS is designed primarily to handle three types of generated wastes: WSW, DSW, and miscellaneous wastes. The generated solid waste varies in characteristics and contamination level and is further divided into the following waste streams:

- WSW, such as spent resin, spent charcoal, filter membranes and reverse osmosis filter membranes, and spent cartridge filters

- DSW, such as ventilation filters, activated charcoal (bulk and filter elements), rags, paper, plastic, rubber, scrap wood, glass, concrete, metal, and failed equipment parts and tools

- miscellaneous waste, including mixed waste and oily sludge

The SWRS is designed to comply with the requirements of 10 CFR 61.55, "Waste Classification"; 10 CFR 61.56, "Waste Characteristics"; 10 CFR Part 71; and 49 CFR Parts 171–180 for wet and dry radioactive solid waste packaged for offsite shipment and disposal.

FSAR Table 11.4-2, "Estimated Annual Volumes of Dry Solid Waste," lists the estimated values for various types of dry solid waste, as well as the container type, container volume, and number of containers required to store the annual amounts of waste. During some AOOs, such as refueling, the rate of DSW generation is higher than during normal operations. Major equipment items, such as core components and containment vessel components, are not processed in the SRWS.

The WSW processing system receives and processes three major waste streams:

(1) radioactive spent resin and spent charcoal
(2) spent cartridge filters
(3) filter membranes and reverse osmosis

FSAR Table 11.4-3, “Estimated Annual Volumes of Wet Solid Waste,” lists the estimated volumes of various types of WSW generated, as well as the container type, container volume, and number of containers required to store the annual amounts of waste.

In FSAR Section 11.4.1.3, NuScale states that mixed waste is a combination of radioactive waste mixed with hazardous waste listed in the Resource Conservation and Recovery Act of 1976, as amended, as defined in 40 CFR Part 261, “Identification and Listing of Hazardous Waste,” Subpart D, “Lists of Hazardous Wastes.” The volume of generated mixed waste is expected to be extremely low. Mixed waste can be disposed of only in a permitted mixed waste disposal facility. Mixed waste is collected near the source and transferred in 208-liter (55-gallon) drums to a permitted facility.

The generation of contaminated oil is expected to be very low. The main source of oily waste is expected to come from floor drains. The oil is directed to the SRWS from the LRWS oil separators and is manually collected in drums. The drums of contaminated oil are sent to an offsite treatment facility.

FSAR Figure 11.4-1, “Block Diagram of the Solid Radioactive Waste System”; Figure 11.4-2a, “Process Flow Diagram for Wet Solid Waste”; and Figure 11.4-2b, “Solid Radioactive Waste System Diagram,” present the SRWS process flow diagrams.

The boundaries of the SRWS begin at the connection to a particular waste stream source and end at the packaged waste container offsite shipment. For WSW, these connections usually involve flanged joints and boundary valves at the system inlets. For DSW, the boundaries are not always physical because much of DSW is collected from a variety of locations and transported through corridors to the solid radioactive waste sorting area.

For spent resins and granular activated charcoal, the SRWS starts downstream of the boundary valve from each demineralizer and carbon bed. Spent resin is sluiced into the SRSTs or PSTs for decay and eventually sent to waste containers. The containers are processed (dewatered and sealed) and placed in the storage area until shipped off site for further processing or disposal.

For spent cartridge filters, the SRWS starts at the filter extraction point. Operators remove the spent filter from the filter housing and place it in a shielded spent filter transfer cask.

FSAR Table 11.4-4, “Solid Radioactive Waste System Equipment Malfunction Analysis,” describes malfunctions, results (consequences), and mitigating and alternative actions for the following system components: SRST, spent resin transfer pump, PST, phase separator transfer pump, dewatering skid, and high-integrity containers.

The COL licensee will conduct SRWS preoperational inspections and testing to ensure operation of components and processes as discussed in FSAR Section 14.2, “Initial Plant Test Program.” The COL licensee will develop administrative procedures and operational controls for waste processing, process parameters, and surveillance requirements for waste processing in accordance with NEI 07-10A, Revision 0 (March 2009).

In assessing the radiological impacts associated with radioactive solid effluent discharges, the SRWS does not release effluents directly to the environment. Liquids removed from solid waste processing are transferred to the LRWS for further processing. During the operation of the SRWS, such as processing and packaging solid waste, the expelled air is captured by the RWBVS to prevent unmonitored contamination being released to the environment. Waste is classified as Class A, Class B, Class C, or greater than Class C, as specified in 10 CFR 61.55 and 10 CFR 61.56, according to the site process control program (PCP). FSAR Tables 11.4-2 and 11.4-3 present expected waste classifications of solid wastes for influents processed and shipped.

In FSAR Section 11.4.2, “Radioactive Effluent Releases,” the applicant stated that any liquids and gases generated from the operation of the SRWS are processed by the LRWS (described in FSAR Section 11.2) and the RWBVS (described in FSAR Sections 9.4, “Air Conditioning, Heating, Cooling, and Ventilation Systems,” and 11.3). As a result, the radiological impacts associated with the expected liquid and gaseous effluents generated during the operation of the plant, including those from the SRWS, are addressed in FSAR Sections 11.2 and 11.3 for the LRWS and GRWS, respectively. SER Sections 11.2 and 11.3 present the NRC staff’s evaluation of liquid and gaseous effluent releases and doses, respectively.

Technical Specifications: TS 5.5.1 and 5.5.2 are associated with the SRWS and are found in SDAA Part 4, Volume 1. The TS also include the following reports: 5.5.6, “Explosive Gas and Storage Tank Radioactivity Monitoring”; 5.6.1, “Annual Radiological Environmental Operating Report”; and 5.6.2, “Radioactive Effluent Release Report.” TS 5.6.2 specifies that these programs are to be consistent with the objectives outlined in the ODCM and PCP.

11.4.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for the SRWS area of review, associated acceptance criteria, and review interfaces with other SRP sections appear in SRP Section 11.4, “Solid Waste Management System.” The following summarizes the regulatory requirements:

- 10 CFR 20.1301, as it relates to dose limits for individual members of the public
- 10 CFR 20.1302, as it relates to limits on doses to members of the public and effluent concentrations and doses in unrestricted areas
- 10 CFR 20.1406, as it relates to the design and operational procedures for minimizing contamination, facilitating eventual decommissioning, and minimizing the generation of radioactive wastes
- 10 CFR 50.34a, as it relates to providing sufficient information and design features to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents from the SRWS to unrestricted areas are kept ALARA
- 10 CFR Part 50, Appendix I, Sections II.A, II.B, II.C, and II.D, as they relate to the numerical guides, design objectives, and limiting conditions for operation to meet the ALARA criterion for equipment installed to process and treat wet and solid radioactive wastes

- 40 CFR Part 190 (EPA's generally applicable environmental radiation standards), as implemented under 10 CFR 20.1301(e) and as it relates to controlling doses within the EPA's generally applicable environmental radiation standards

- GDC 60, as it relates to the design of the SRWS to control the release of radioactive materials in liquid and gaseous effluents from the SRWS and to handle wet and solid wastes produced during normal plant operation, including AOOs

- GDC 61, as it relates to the system design for solid radioactive waste systems and the ability of such systems containing radioactivity to ensure adequate safety under normal operations and AOOs and provide suitable shielding for radiation protection

- GDC 63, "Monitoring Fuel and Waste Storage," as it relates to the ability of the SRWS to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions

- 10 CFR 52.47(b)(1), which requires that applications for DCs contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC and provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations

- 10 CFR 52.137(a)(5), as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits established in 10 CFR Part 20

The following guidance documents contain the regulatory positions and guidance for meeting the relevant requirements of the regulations identified above:

- SRP Section 11.4, BTP 11-3, Revision 4, "Design Guidance for Solid Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," issued January 2016

- SRP Section 11.4, Appendix 11.4A, including updated guidance from SECY-93-323, "Withdrawal of Proposed Rulemaking to Establish Procedures and Criteria for On-Site Storage of Low-Level Radioactive Waste After January 1, 1996," dated November 29, 1993, and SECY-94-198, "Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste," dated August 1, 1994, with respect to long-term onsite storage (e.g., for several years but within the operational life of the plant)

- RG 1.143, Revision 2, as it relates to the seismic design; quality group classification of components; general guidelines for design, construction, and testing criteria for radioactive waste systems; and general QA guidelines for RWMSs

- RG 4.21, Revision 0, as it relates to minimizing the contamination of equipment, plant facilities, and the environment and the generation of radioactive waste during plant operation

- GL 89-01, as it relates to the restructuring of the PCP and radiological effluent technical specification (included in NUREG-1301)

- NUREG-1301, Revision 0, as it relates to the development of a plant-specific PCP or, alternatively, a COL applicant using the NRC-approved NEI PCP Template 07-10A, Revision 0, “Generic FSAR Template Guidance for Process Control Program (PCP),” issued March 2009 (ML091460627), to meet this regulatory milestone until a plant-specific PCP is prepared, before fuel load, under the requirements of a license condition described in FSAR Section 13.4, Operational Programs,” of a COL application

- Regulatory Issue Summary 2008-32, “Interim Low-Level Radioactive Waste Storage at Reactor Sites,” dated December 30, 2008, as it relates to the use of NRC and industry guidance in addressing limited access to radioactive waste disposal facilities

- RG 8.8, Revision 3, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” issued June 1978 (ML003739549)

- RG 8.10, Revision 2, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable,” issued August 2016 (ML16105A136)

- IE Bulletin No. 80-10, as it relates to methods and procedures used in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled releases of radioactivity

11.4.4 Technical Evaluation

The NRC staff evaluated the information in FSAR Revision 1, Section 11.4, against the applicable NRC regulations and guidance in SRP Section 11.4 and DSRS Section 11.4, “Solid Waste Management System,” issued June 2016 (ML15355A336).

11.4.4.1 Design Considerations

11.4.4.1.1 General Design Criteria 60 and 61 and 10 CFR 50.34a

GDC 60 requires the nuclear power unit design to include provisions to handle radioactive wastes produced during normal reactor operations, including AOOs. GDC 61 requires that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions.

The NRC staff reviewed the waste generation rates described in FSAR Tables 11.4-2 and 11.4-3. The NRC staff observed the applicant’s commitments to meeting the design criteria in RG 1.143 and BTP 11-3. The design also meets the criteria in ANSI/ANS-55.1-1992, “Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants,” to have a minimum onsite storage capacity for solid waste of 30 days. The applicant included FSAR Table 11.4-3 to state the volumes of Class A and Class B/C waste generated per year. FSAR Section 11.4.1.4, “Packaging, Storage, and Shipping,” describes below-grade areas for storing waste and above-grade areas for Class A waste. FSAR Section 11.4.1.4 states that, at expected waste generation rates, there is storage capacity for at least 30 days. The staff reviewed the layout drawings in FSAR Chapter 12 and identified a high-integrity container storage room and another room for storing waste below grade in the RWB. The staff also identified areas above-grade to store waste. Based on this, and the waste generation rates described in the FSAR including FSAR tables 11.4-2 and 11.4-3, the NRC staff has determined that the applicant has enough storage space to meet the minimum onsite storage time of 30 days as described in BTP 11-3 and ANSI/ANS-55.1-1992.

The requirements of GDC 60 and 61 may be met by conforming to the regulatory positions in RG 1.143, Revision 2, as they relate to the seismic design, quality group classification of components used in the SRWS and structures housing the systems, provisions used to control leakage, and definitions of discharge paths, beginning with interfaces with plant primary systems and terminating at the point of controlled discharges.

The NRC staff reviewed the QA provisions and RG 1.143, Revision 2, specified by the applicant in the FSAR. The applicant stated that the SRWS will conform to RG 1.143, Revision 2, which specifies the QA guidance to follow. FSAR Table 11.4-1 identifies the safety class and other design information for components of the SRWS. In determining the design guidance for radwaste systems, the applicant provided FSAR Table 11.4-1 to reflect the guidance specified in RG 1.143, Revision 2, to classify components based on the A_1 and A_2 values in 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” Appendix A, “Reportable Safety Events.” The applicant defined the boundaries of the radwaste systems classifications as those components up to and including the system isolation valves. This definition is consistent with system boundaries reviewed in the past, and the NRC staff finds this boundary definition acceptable.

The NRC staff’s confirmatory calculations of the data provided by the applicant determined that the applicant had correctly calculated radwaste system seismic categories. The NRC staff performed the confirmatory calculations using the source terms listed in FSAR Table 11.4-1. The NRC staff verified the listed RG 1.143 safety classification categories for equipment listed in this table. The NRC staff determined that the applicant has appropriately calculated the seismic categories of SRWS components using the guidance contained in RG 1.143 to satisfy the requirements in GDC 60, 61, and 64.

For compliance with 10 CFR 50.34a, the applicant must provide sufficient information to demonstrate that the design objectives of equipment necessary to treat and control releases of radioactive effluents into the environment have been met. The requirements of 10 CFR 50.34a are met by describing the process used to treat and handle solid waste and by identifying the system boundaries. The applicant provided FSAR Tables 11.4-2 and 11.4-3 to identify the planned waste generation rates, along with the planned shipped waste rates. FSAR Table 11.4-1 also includes the waste classifications for the waste streams generated from the SRWS.

The requirements of GDC 61 and GDC 63 specify that the SRWS include shielding and ventilation design features to protect workers and control releases of gaseous radioactivity in the environment. A review of FSAR Section 12.3 indicates that the SRWS and features of the RWB include measures to shield components expected to contain higher levels of radioactivity and display higher radiation exposure rates. Similarly, gaseous phases released from tanks, vessels, and the waste compactor are captured by the ventilation system of the RWB and are monitored before being released to the environment through the RBVS plant stack monitor. Finally, FSAR Chapter 12 identifies area radiation monitors in selected areas containing SRWS components and airborne radiation monitors in the RWB to monitor ambient radiation exposure rates and airborne radioactivity levels and alert operators of changing conditions and when to take corrective steps. The design also includes monitors at potential discharge paths to ensure that the potential releases of radioactive material are appropriately monitored and controlled. Based on this, the staff finds that GDC 61 and GDC 63 are appropriately addressed for the SRWS.

11.4.4.1.2 Effluent Release Requirements

The NRC staff's review of the LRWS and GRWS in SER Sections 11.2 and 11.3 addressed the radiological impacts of effluent releases, since the SRWS does not release liquid and gaseous effluents directly to the environment. Therefore, SER Sections 11.2 and 11.3 present the NRC staff's evaluation of compliance with effluent release requirements. The evaluation considers liquid and gaseous effluents generated during the processing of solid and wet wastes and whether the equipment and design features are acceptable and meet the requirements of 10 CFR 20.1302; the effluent concentration limits of 10 CFR Part 20 (Appendix B, Table 2, Columns 1 and 2); the requirements of 10 CFR 20.1406 to minimize the contamination of the facility and environment; and the design objectives of 10 CFR Part 50, Appendix I. Sections 11.2 and 11.3 also address the requirements of 10 CFR 20.1301(e) to control doses within the EPA's generally applicable environmental radiation standards under 40 CFR Part 190. The 10 CFR 20.1301 dose limits, including the requirements of 40 CFR Part 190, required by 10 CFR 20.1301(e), also direct radiation dose to be considered. The radwaste systems in the NuScale design are not expected to cause any significant direct radiation exposure offsite because the system components are located indoors and inside of appropriately shielded areas (with the more significant sources having the greatest shielding).

The SRWS is designed to send liquid and gaseous effluents to the LRWS and RWBVS, respectively. Other than solid waste shipments off site, the SRWS does not release effluents directly to the environment. Any contribution to the offsite dose consequences from the SRWS is included in the evaluations of the LRWS and GRWS in SER Sections 11.2 and 11.3. The NRC staff finds this acceptable.

FSAR Section 11.4.3, "Malfunction Analysis" and FSAR Table 11.4-4 describes potential solid radwaste system failures, results, and mitigating actions. These failures should not result in any significant release to the environment that is not accounted for in SER Sections 11.2 and 11.3 based on facility design features and the physical form of solid radioactive waste. The information in Table 11.4-4 and the minimization of contamination design features discussed below and in FSAR Chapter 12, provide assurance that contamination of the facility will be minimized to the extent practicable and that if accidental malfunctions were to occur that they can be addressed with minimal contamination of the facility and the environment and while keeping doses to workers and the public as low as is reasonably achievable.

11.4.4.1.3 Site-Specific Cost-Benefit Analysis

FSAR Section 11.4.2 states that the SRWS does not release effluents to the environment and that an SRWS CBA is not performed separately from the evaluations in Sections 11.2 and 11.3. The NRC staff finds this acceptable.

11.4.4.1.4 Minimization of Contamination

The NRC staff reviewed the information presented in FSAR Sections 11.4 and 12.3.6 and compared it to the criteria in 10 CFR 20.1406 for minimizing contamination. The regulations in 10 CFR 20.1406 require, in part, that design features and procedures 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.

In FSAR Section 12.3.6, the applicant described the use of RG 4.21 in meeting the requirements of 10 CFR 20.1406. RG 4.21 describes an acceptable method for implementing a program that will minimize radioactive waste generation and contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste.

FSAR Table 12.3-38, "Regulatory Guide 4.21 Design Features for Solid Radioactive Waste System," describes how the design of the SRWS includes provisions to minimize the contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste. The information included in FSAR Table 12.3-38 includes a design with above-ground piping when able, and double walled piping when the piping is underground; epoxy-coated floors and drainpipes to direct leakage to floor drains; and tank cubicles that are designed with stainless steel walls and floors to contain the contents of the tank and that have sloped floors to direct leaks to floor drains. FSAR Section 12.3.6 and SER Section 12.3 provide additional information on compliance with 10 CFR 20.1406, including programmatic requirements to be addressed by a COL applicant.

11.4.4.1.5 Process Control Program

A PCP ensures that the production of solid waste is handled in accordance with 10 CFR Part 71 and the guidance of BTP 11-3. The applicant stated that the PCP will follow the guidance of NEI 07-10A and will be developed to meet the requirements in 10 CFR Part 71.

In addition, in FSAR Section 13.4, COL Item 13.4-1 specifies that an applicant that references the SDA will provide site-specific information, including implementation milestones, for operational programs, including the PCP. The NRC staff finds this acceptable since the FSAR states that the design will follow the guidance of NEI 07-10A, which is acceptable guidance for developing a PCP.

11.4.4.1.6 Mobile Equipment

The SRWS does not include mobile equipment for the NuScale SDA design.

11.4.4.1.7 Technical Specifications

In SDAA Part 4, the applicant described the TS associated with the RWMSs. SDAA Part 4, Section 5.6.2, "Radioactive Effluent Release Report," requires that the annual report include a summary of the quantities of the radioactive liquid and gaseous effluents and solid waste released from the unit. SDAA Part 4, Section 5.6.2, also requires that the information in the annual summary be consistent with the objectives outlined in the PCP and comply with the requirements of 10 CFR 50.36a and 10 CFR Part 50, Appendix I. The NRC staff finds the TS requirements acceptable, as the plant-specific PCP will address the implementation of such programs.

11.4.5 Combined License Information Items

FSAR Section 11.4 contains no COL items. However, in FSAR Section 13.4, COL Item 13.4-1 specifies that an applicant that references the SDA will provide site-specific information, including implementation milestones, for operational programs, including the PCP.

The NRC staff finds that this adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items that need to be included in the FSAR for the SRWS.

11.4.6 Conclusion

The NRC staff concludes that the SRWS includes the equipment necessary to collect, hold, process, package, and store WSW and DSW and control releases of radioactive materials associated with the operation of the SRWS. The applicant provided sufficient design information to demonstrate that it has met the requirements of 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 50.34a; 10 CFR Part 50, Appendix A, GDC 60, 61, and 63; 10 CFR Part 50, Appendix I; 40 CFR Part 190; 10 CFR 52.137(a)(5); and SRP and DSRS Section 11.4 acceptance criteria. This conclusion is based on the following:

- The NuScale design demonstrates compliance with 10 CFR 50.34a, as it relates to the inclusion of sufficient design information and system design features that are necessary for collecting, holding, processing, handling, packaging, and safely storing wet and dry solid radioactive wastes. The design conforms to the guidelines of BTP 11-3 and SRP Section 11.4, Appendix 11.4-A. The NuScale design demonstrates compliance with the requirements of 10 CFR 50.34a in providing sufficient wet and solid waste processing capacities and storage space to ensure adequate safety. The NuScale design of the radioactive waste storage area in the RWB includes provisions for 30 days of onsite storage of processed solid and wet wastes.

- The NuScale design provides sufficient information and design features to satisfy the guidance of RG 1.143, Revision 2, for SRWS processing systems. The SRWS system and components and structure housing the SRWS are appropriately classified in accordance with RG 1.143, Revision 2.

- The SRWS includes 10 CFR 20.1406 design features, as provided in FSAR Table 12.3-38.

- The SRWS does not directly contribute to offsite effluent doses (since any liquid or gas from the solid radwaste systems are processed by those systems before release), as assessed by the requirements 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR Part 20, Appendix B; 10 CFR Part 50, Appendix I; and GDC 60. There is also no significant direct radiation exposure to the public from the SRWS.

- The SRWS components expected to contain higher levels of radioactivity are shielded. In addition, gaseous phases released from tanks, vessels, and the waste compactor are captured by the ventilation system of the RWB and are monitored before being released to the environment through the RBVS plant stack monitor. Finally, FSAR Chapter 12 includes area radiation monitors in selected areas containing SRWS components and airborne radiation monitors in the RWB to monitor ambient radiation exposure rates and airborne radioactivity levels and alert operators of changing conditions and when to take corrective steps. Therefore, the SRWS includes appropriate design features to address the requirements of GDC 61 and GDC 63.

- The NuScale design implements a plant-specific PCP as an operational program, described in FSAR Sections 11.4.2 and 13.4, for the processing of low-level radioactive waste. The PCP addresses plant-specific operating procedures and acceptance criteria as they relate to the treatment and processing of radioactive wastes, such that waste products generated by the SRWS will meet the classification and characterization definitions in 10 CFR 61.55 and 10 CFR 61.56, respectively. The implementation of a PCP is specified under COL Item 13.4-1, as described in FSAR Table 1.8-1, “Combined License Information Items.” The PCP will also be developed to meet the requirements of 10 CFR Part 71.

- In FSAR Section 11.4, the applicant addresses aspects of 10 CFR 52.137(a)(5) related to solid radioactive waste, because the applicant identifies the types and quantities of solid radioactive waste and addresses the means of keeping radiation exposures within the limits set forth in 10 CFR Part 20, from solid radioactive waste.

11.5 Process and Effluent Radiation Monitoring Instrumentation and Sampling System

11.5.1 Introduction

The PERMISS is used to monitor liquid and gaseous process streams and effluent releases from the RWMS during normal operation, AOOs, and post-accident conditions. The system includes radiation monitors to detect and measure radioactivity and radiation levels and to indicate radioactive release rates or concentration levels in process and effluent streams. The PERMISS includes sampling systems to extract samples from process or effluent streams. The PERMISS establishes alarm setpoints to indicate when excessive radioactivity levels are present to ensure requirements are met; tracks and records rates of radioactivity releases; and initiates protective isolation actions, such as terminating or diverting process or effluent flows. The system consists of radiation monitoring equipment and permanently installed sampling lines, with the equipment located at points to measure radioactivity or collect samples representative of process flows and effluent releases. Samples collected on filtration and in adsorbent media are evaluated by laboratory analyses to confirm measurement results recorded by radiation monitors and to determine radioactivity levels associated with radionuclides that are not readily detected by radiation monitoring devices. The system includes local instrumentation readout panels and alarm functions. In addition, all monitors provide radiation indications in the main control room and all radiation monitor alarms will also be heard in the main control room.

11.5.2 Summary of Application

SDAA Part 8: The ITAAC information associated with FSAR Section 11.5 is found in SDAA Part 8, Section 2.7, “Radiation Monitoring—Module Specific,” and Section 3.9, “Radiation Monitoring—Shared Systems.” Each of these sections discusses the non-safety-related automatic actions for various radiation monitors for module-specific and shared radiation monitors.

FSAR Section 11.5: The applicant described the system in FSAR Section 11.5, summarized as follows.

In FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System,” the applicant described the PERMISS and its functions in monitoring, recording, tracking, and controlling radioactivity levels, release rates, and concentrations in effluents during operations, AOOs, and accident conditions. The system provides the means to terminate and isolate process flows and effluent releases upon detecting elevated levels of radioactivity. The system consists of radiation monitoring equipment and sampling lines, with the equipment located at points to measure radioactivity or collect samples that are representative of process flows and effluent releases. The system also collects liquid and gaseous samples from various process and effluent streams for analyses conducted in laboratory settings. Samples collected on filtration and in adsorbent media are evaluated by laboratory analyses to confirm measurement results recorded by radiation monitors and determine radioactivity levels associated with radionuclides that are not readily detected by radiation monitoring devices. The system includes local instrumentation readout panels and alarm functions, in addition to those located in the main control room. The PERMISS subsystems and components are found throughout the plant as design requirements of plant systems.

In FSAR Section 11.5.1, “System Description,” the applicant described the system with FSAR references for each system that includes a process and effluent radiation monitor.

FSAR Table 11.5-1, “Process and Effluent Radiation Monitoring Instrumentation Characteristics,” describes the process and effluent radiation monitoring instrumentation, including information on the system where the monitors are located, the type and quantity of the radiation monitors, the location of the monitors, and general information regarding the function of the radiation monitors. FSAR Tables 11.5-2, “Provisions for Sampling Gaseous Process and Effluent Streams,” and 11.5-3, “Provisions for Sampling Liquid Process and Effluent Streams,” provide information on gaseous and liquid process and effluent radionuclide sample points, respectively. FSAR Table 11.5-4, “Effluent and Process Monitoring Off Normal Radiation Conditions,” discusses the effluent and process monitoring off-normal radiation conditions and lists the various system responses to high radiation levels. This includes information on system response and potential manual actions if high radiation is detected.

The applicant stated that the PERMISS design features provide the means to detect, measure, and control liquid and gaseous effluent releases, in accordance with the concentration limits of 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2; 10 CFR 20.1301(e), insofar as it requires meeting the EPA environmental radiation protection standards of 40 CFR Part 190; and the design objectives of 10 CFR Part 50, Appendix I. The applicant stated that the design complies with the requirements of 10 CFR 50.34a and GDC 60, 63, and 64, using the acceptance criteria of DSRS Section 11.5, “Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems,” and associated regulatory guidance. For requirements related to lessons learned from the accident at Three Mile Island (TMI), the applicant stated that the design conforms to 10 CFR 50.34(f)(2)(xvii), as it relates to postaccident radiation monitoring, and 10 CFR 50.34(f)(2)(xxvii) in monitoring gaseous effluents from potential accident release points using regulatory guidance. In addition to the monitoring information in FSAR Section 11.5, FSAR Section 7.1.1.2.2, “Post-Accident Monitoring,” and Table 7.1-7, “Summary of Post-accident Monitoring Variables,” discuss post-accident monitoring.

In addition to the information in FSAR Section 11.5, the sections referenced in FSAR Section 11.5.1 discuss the systems that include process and effluent radiation monitors, including information related to the radiation monitors, such as the location, purpose, and function of some of the radiation monitors.

Technical Specifications: The following TSs associated with the PERMISS are in SDAA Part 4: 5.5.1, 5.5.2, 5.5.4, and 5.5.6. The PERMISS is the primary monitoring equipment used for the ODCM and the radioactive effluent control program, described in TSs 5.5.1 and 5.5.2. TS 5.5.4 refers to the PERMISS as it relates to radiation monitoring for primary-to-secondary leakage. TS 5.5.6 refers to the PERMISS as it relates to monitoring the quantities of radioactivity in storage tanks and the GRWS charcoal beds and potential releases of radioactivity.

Technical Reports: No technical reports are associated with this area of review.

Topical Reports: No topical reports are associated with this area of review.

11.5.3 Regulatory Basis

The relevant requirements of the Commission's regulations for the PERMISS area of review, associated acceptance criteria, and interfaces with other SRP sections appear in SRP Section 11.5. The following summarizes the regulatory requirements:

- 10 CFR 20.1301, as it relates to dose limits for individual members of the public
- 10 CFR 20.1302, as it relates to limits on doses to members of the public and effluent concentrations and doses in unrestricted areas
- 10 CFR 50.34a, as it relates to equipment design and procedures used to control releases of radioactive material to the environment within the numerical guides in 10 CFR Part 50, Appendix I
- 10 CFR 50.36a, as it relates to operating procedures and equipment installed in RWMSs, under 10 CFR 50.34a, to ensure that releases of radioactive materials to unrestricted areas are kept ALARA
- 10 CFR Part 50, Appendix I, as it relates to numerical guides and design objectives to meet the requirements of 10 CFR 50.34a and 10 CFR 50.36a, which specify that radioactive effluents released to unrestricted areas and doses to members of the public be kept ALARA
- 10 CFR 20.1406, as it relates to the design and operational procedures in minimizing contamination of the facility, facilitating eventual decommissioning, and minimizing the generation of radioactive waste
- GDC 13, "Instrumentation and Control," as it stipulates, in part, that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions, as appropriate, to ensure adequate safety
- GDC 60, as it relates to controlling radioactive effluent releases and handling radioactive materials produced during normal plant operation, including AOOs
- GDC 63 and GDC 64, as they relate to the designs of the capabilities to monitor and control radiation levels and radioactivity in effluents, as well as radioactive leakages and spills, during routine operation, AOOs, and postulated accidents, and to initiate appropriate safety actions
- 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii), with regard to monitoring gaseous effluents from all potential accident release points, which are consistent with the requirements of GDC 63 and GDC 64 and which correspond to TMI Action Plan Items II.F.1 and III.D.3.3, respectively

- 10 CFR 52.47(b)(1), which requires that an SDAA contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analysis are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act of 1954, as amended, and NRC regulations

- 10 CFR 52.137(a)(5), as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits established in 10 CFR Part 20

The following documents contain the regulatory positions and guidance for meeting the relevant requirements of the regulations identified above:

- RG 1.21, Revision 3, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” issued September 2021 (ML21139A224), as it relates to guidance for the design, implementation, and QA of effluent monitoring and sampling systems

- RG 1.33, as it relates to QA for the operation of safety-related equipment that is part of the PERMISS

- RG 1.97, Revision 5, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” issued April 2019 (ML18136A762), as it relates to accident monitoring instrumentation and performance of radiation monitoring systems (additional guidance on the application of RG 1.97 is provided in the DSRS for NuScale SMR Design, Chapter 7, “Instrumentation and Controls”).

- RG 4.15, Revision 2, “Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment,” issued July 2007 (ML071790506), as it relates to the design, implementation, and QA of effluent monitoring and sampling systems

- RG 4.21, as it relates to minimizing both the contamination of equipment, plant facilities, and the environment and the generation of radioactive waste during plant operation

- Radiological Assessment BTP, Revision 1, issued November 1979, as it relates to the conduct of environmental monitoring (included as Appendix A to NUREG-1301)

- NUREG-0133, “Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants: A Guidance Manual for Users of Standard Technical Specifications,” issued October 1978 (ML091050057), as it relates to the format and contents of ODCMs

- SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses and Acceptance Criteria,” issued October 2005 (ML052770257), as it relates to descriptions of operational programs

- ANSI/Health Physics Society (HPS) N13.1-2011, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities,” as it relates to sampling and monitoring airborne releases from stacks

- ANSI N42.18-2004, “Specification and Performance of On Site Instrumentation for Continuously Monitoring Radioactivity in Effluents,” as it relates to the performance of radiation monitoring equipment
- SRP Section 11.5, Appendix 11.5-A, “Design Guidance for Radiological Effluent Monitors Providing Signals for Initiating Termination of Flow or Other Modification of Effluent Stream Properties,” as it relates to the design of automatic control functions
- NRC IE Bulletin No. 80-10, as it relates to methods and procedures used in avoiding the cross-contamination of nonradioactive systems and unmonitored and uncontrolled releases of radioactivity
- GL 89-01, as it relates to the restructuring of the ODCM and radiological effluent technical specification (included in NUREG-1301)
- NUREG-1301, as it relates to the development of a plant-specific ODCM, or the use of NEI 07-09A, Revision 0, found acceptable by the NRC staff in a letter dated January 27, 2009 (ML083530745), to meet this regulatory milestone until a site-specific ODCM is prepared, before fuel load, under the requirements of a license condition described in FSAR Section 13.4
- EPRI TR-104788, Revision 2, “PWR Primary-to-Secondary Leak Guidelines—Revision 2,” issued February 2000, as it relates to an industry-developed approach for calculating and monitoring primary-to-secondary leak rates
- Information Notice (IN) 2005-24, “Nonconservatism in Leakage Detection Sensitivity,” dated August 3, 2005, as it relates to reactor coolant activity assumptions for containment radiation gas channel monitors

11.5.4 Technical Evaluation

The NRC staff evaluated the information in FSAR Sections 11.5 and 11.6 against the applicable NRC regulations and guidance in SRP Section 11.5 and in DSRS Sections 11.5, “Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems,” issued June 2016 (ML15355A337), and 11.6, “Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring,” issued June 2016 (ML15355A338).

11.5.4.1 Design Considerations

11.5.4.1.1 Effluent and Process PERMISS

The primary purpose of the PERMISS is to provide information characterizing the types and amounts of radioactivity in process streams and liquid and gaseous effluents. Other objectives are to alert control room operators of abnormal levels of radioactivity in process streams and in liquid and gaseous effluents and provide signals that initiate automatic functions, isolate process streams, and terminate effluent discharges if predetermined radioactivity levels or release rates exceed alarm setpoints. Another function of the PERMISS is to provide the means to collect samples from process and effluent streams for radiological analysis. The design objectives and criteria of the PERMISS are intended to address the following:

- radiation instrumentation required for monitoring plant operations and safe shutdown and to alert operators of unusual or unexpected plant radiological conditions

- radiation monitoring of liquid and gaseous effluent releases during normal operations and as a result of potential accidents, as well as associated equipment to mitigate unintended radiation releases and the potential spread of radioactive material

In FSAR Section 11.5, the applicant described the PERMISS and its functions in monitoring, recording and tracking, and controlling radioactivity levels, release rates, and concentration levels in effluents during plant operation, AOOs, and accident conditions with FSAR tables and FSAR section references. The PERMISS consists of skid-mounted and permanently installed sampling and monitoring equipment designed to indicate operational radiation levels and releases of radioactive materials, equipment or component failures, and improper operation. Many of the PERMISS monitors include beta and gamma radiation-sensitive detectors working in redundant channels, as provided by the design, where appropriate. The radiation detectors detect the types and energies of radiation in the various systems in the plant, radioactive wastes, and process and effluent streams. Depending on the location, high radiation levels could indicate small fission product leakage from the fuel, liquid or gas leakage from plant systems, or other unintended or unusual plant conditions. High radiation in some monitors results in automatic actions, while others alert the operators for manual action, as appropriate. Actions as a result of high radiation may be related to preventing or mitigating a release of radioactivity, the spread of contamination, dose to plant personnel, or otherwise assuring appropriate plant operation. FSAR Tables 11.5-1 and 11.5-4 provide information on the monitors, including their locations, ranges, functions, and response to high radiation. Local readout and alarm panel indicators are located at specific areas to provide information on the radiological status of plant systems and function to alert personnel of abnormal conditions. There are also high-radiation alarms and radiation monitor data available in the control room for all the PERMISS monitors. High radiation levels on monitors result in automatic actions when appropriate. For example, when the containment evacuation system receives a high-radiation signal, the purge gas supply to the vacuum pumps is automatically shut, and the discharge path is switched from the RBVS to the GRWS. Other monitors, such as the RXB and RWBVS monitors, include no automatic action, but they alarm and alert the operators to find the source of the radiation and to take appropriate action. The COL licensee will test the PERMISS before operations and is responsible for calibrating and testing the PERMISS subsystems installed in the plant. NuScale COL Item 13.5-6 specifies that the COL applicant will develop calibration and test procedures, and FSAR Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," provides initial testing of automatic actuations on a high-radiation signal from selected PERMISS monitors.

The discussion of the PERMISS is divided into two parts—effluent monitoring and process radiation monitoring, as listed in FSAR Section 11.5.1. The NRC staff has reviewed the information in FSAR Section 11.5, including the list in Section 11.5-1 and Table 11.5-1, and verified that, because the listed monitors are provided with locations, function descriptions, and anticipated ranges during operation, they provide reasonable assurance that applicable requirements will be met.

The NRC staff reviewed the radiation monitors discussed in FSAR Section 11.5 and Section 7.1, "Fundamental Design Principles," and evaluated the monitors against the committed guidance of RG 1.97. RG 1.97 refers to Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 497-2016, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." In its review of FSAR Section 11.5, the

NRC staff uses this standard to verify those postaccident monitoring variables discussed in the design and to ensure that monitoring exists to determine the magnitude of releases of radioactive materials. For the PERMISS monitors, there are only Type E variables, as stated in FSAR Table 7.1-7, Summary of Post-accident Monitoring Variables." In reviewing FSAR Table 7.1-7, the staff noted that RXB noble gas and particulate and halogen activity monitors are Type E, as well as the air-cooled condenser vacuum pump exhaust noble gas monitors. In addition, various area and airborne radiation monitors (discussed in FSAR Chapter 12) that monitor the plant areas for radiation and radioactivity are classified as Type E. IEEE 497-2016 indicates that Type E variables provide primary information to the accident management personnel and are required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases. The staff finds that the monitors identified as postaccident monitors are appropriate for postaccident monitoring; since the gaseous effluent pathway is through the RXB stack, it is appropriate for the air monitors monitoring potential gaseous releases in the RXB to be Type E. It is also appropriate for the air-cooled condenser system vacuum pump exhaust monitors to be Type E for potential releases from that pathway. The area and airborne radiation monitors in FSAR Chapter 12 address the other Type E radiation monitors that monitor postaccident radiation levels in plant areas. The staff notes that the radiation monitors under the bioshield (discussed primarily in FSAR Chapter 12) are Type B, Type C, and Type F variables. In evaluating the information provided in the sections and tables described, the staff finds that the applicant has provided the necessary process and effluent radiation monitor postaccident monitoring variables specified in RG 1.97 and IEEE 497-2016.

11.5.4.1.2 General Design Criteria 13, 60, 63, and 64

GDC 13 requires instrumentation to monitor variables and systems during plant operation, AOOs, and accident conditions, which includes instrumentation used for monitoring the fission product barriers. GDC 60 requires that the nuclear power unit be designed to control releases of radioactive material, including during AOOs. GDC 63 requires appropriate systems in fuel storage and radioactive waste systems to detect conditions that may result in the loss of residual heat removal capability and in excessive radiation levels. GDC 64 requires that a plant monitor radioactive releases. A review of the effluent monitoring determined that the applicant has provided monitors for all release points. To limit or terminate releases to the environment, the applicant provided monitors for the RWBVS, CARS exhaust, turbine gland steam outlet, PCWS, auxiliary boiler system, LRWS, and UWS. The NRC staff reviewed the potential release points in the NuScale design and verified that radiation monitoring was provided at all release points (see SER Sections 11.2.4.1.2 and 11.3.4.1.2). As discussed in SER Section 11.5.4.1.1, monitors for postaccident releases are designated as Type E variables. The staff verified that the RBVS exhaust stack noble gas monitor has an extended range for detecting noble gas radiation levels following potential accident conditions. In addition, the applicant described setpoints for high-radiation alarms that will ensure compliance with 10 CFR Part 20 and 10 CFR Part 50. Therefore, the NRC staff has determined that the applicant meets the requirements of GDC 13, 60, 63, and 64 by providing monitoring on all release points on effluent stacks, discharges, and outdoor tank vent lines.

Argon can be added to the primary coolant to support primary-to-secondary leakage detection, as discussed in FSAR Section 5.2.3.4.1, "Prevention of Sensitization and Intergranular Corrosion of Austenitic Stainless Steel. The applicant discussed information on the Ar-41 monitors used for primary-to-secondary leakage detection, including the ranges, in an audit item response (ML23304A474) and in DCA response to RAI 9236 dated February 6, 2018 (ML18037A732), as specified in the response to the audit item. The applicant discussed extending the low-end range of the monitor to indicate leakage based on the RCS concentrations of Ar-41 present. The applicant established the low-end of the radiation monitor range by using the primary coolant activity concentration of Ar-41 and developed the corresponding secondary coolant activity concentration using the applicable parameters from FSAR Table 11.1-2, with an Ar-41 injection concentration of 3.7 kilobecquerels per cubic centimeter (kBq/cm^3) (0.1 microcurie per cubic centimeter ($\mu Ci/cm^3$)) and a primary-to-secondary leak rate of 2.5 kilograms per day, per unit (5.5 pounds per day, per unit).

The applicant also found a viable method to inject Ar-40, to be neutron-activated in the RCS, to be able to maintain an activated level of Ar-41 at 3.7 kBq/cm^3 (0.1 $\mu Ci/cm^3$) in the RCS for leakage detection and determination if necessary. Although the applicant can extend the range of the CES radiation monitor and maintain levels of radioactive Ar-41 in the RCS, the monitor is not required by TS to measure leakage but only to indicate RCS leakage, because there are four other channel indicators of RCS leakage to satisfy TS requirements.

The constant level of Ar-41 in the RCS is also used to detect primary-to-secondary leakage to the Air Cooled Condenser System and through the main steam system line radiation monitors. In FSAR Section 11.1.1.2, “Activation Products” and FSAR Section 9.3.4, “Chemical and Volume Control System,” the applicant discussed the use and methodology of implementing argon injection for determining primary coolant leakage by following the guidance in the EPRI technical document “Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines—Revision 4,” issued November 2011, which is Reference 7.2.13 in NuScale TR-123242, Revision 1. The NRC staff has reviewed this methodology, which has been used successfully at PWRs. This referenced guideline contains the methodology and equations for argon injection and recommends a target baseline RCS concentration of 3.7 kBq/cm^3 (0.1 $\mu Ci/cm^3$). This process complies with the requirements of GDC 13, 60, and 64 by providing monitoring to indicate leakage of the RCS, which will allow operators to perform the necessary actions to control radioactive material.

11.5.4.1.3 Three Mile Island-Related Requirements, 10 CFR 50.34(f)

The NuScale design describes monitoring for the release of noble gases from designated release points and provides continuous monitoring and sampling for radioiodine and particulate releases from accident release points. The RBVS and the RWBVS can provide this sampling and monitoring capability to alert plant personnel. The RWBVS is ultimately directed to the RBVS, where all releases are monitored by the plant stack radiation monitors. In addition, the NuScale design monitors the control room conditions for airborne radiation. The NuScale design also includes a variety of in-plant area and airborne radiation monitors to monitor both routine operation and potential accident conditions. The described effluent and control room monitors, as well as appropriately designed area and airborne radiation monitors, follow the guidance of RG 1.97 and all have a designated Type E variable, as specified in FSAR Section 7.1.1.2.2 and Table 7.1-7. Therefore, the NRC staff finds these monitors acceptable for meeting the requirements in 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii).

11.5.4.1.4 Minimization of Contamination

FSAR Section 11.5 does not discuss minimization of contamination but it is discussed in FSAR Section 12.3.6. The staff’s review of this topic is discussed in SER Chapter 12. However, the staff notes that the process and effluent monitors play a role in minimizing the spread of contamination, in accordance with 10 CFR 20.1406, by detecting and allowing for sampling of potential effluent releases and the unintended spread of radioactive material. On high radiation detection, some monitors result in automatic actuations to contain radioactive material or to alert operators to take appropriate actions to mitigate the spread of radioactive material.

11.5.4.1.5 10 CFR 52.47(b)(1) for ITAAC

SER Chapter 14 contains the staff's review of ITAAC related to radioactive waste systems and radiation effluent monitoring. The review in SER Chapter 14 contains the staff's assessment on ITAAC in SDAA Part 8, Sections 2.7 and 3.9.

11.5.4.1.6 Technical Specifications

In SDAA Part 4, the applicant provided the TS and TS Bases associated with the RWMS. In TS 3.4.7, the applicant described the RCS leakage detection instrumentation. The radiation monitor used in this TS indicates leakage in the CES. In conjunction with the condensate level and pressure channels, it is used to determine RCS leakage into the CES.

TS 5.5.1 and 5.5.2 provide directions for managing releases of radioactive effluents and the control and handling of concentrated wastes for disposal. TS 5.5.6 specifies the quantity of radioactivity contained in gas storage tanks and in unprotected outdoor liquid storage tanks, in accordance with BTP 11-5 and BTP 11-6, respectively. TS 5.5.6 requires concentration limits and surveillances of hydrogen and oxygen in the GRWS, whether or not the system is designed to withstand a hydrogen explosion, and ensures that the quantity of radioactivity in each gas storage tank is less than the amount that would result in a whole body exposure greater than or equal to 1 mSv (0.1 rem) to any individual in an unrestricted area in the event of a gas tank failure. It also ensures that the quantity of radioactivity in all outdoor liquid tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the LWMS, is less than the effluent concentration limits in 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of a tank failure.

In TS 5.6.1 and TS 5.6.2, the applicant specified annual reporting requirements in describing the results of the radiological monitoring program and summarized the quantities of radioactive liquid and gaseous effluents released into the environment. TS 5.5.1 states that COL-initiated changes to the ODCM are to be documented with sufficient information by analyses or evaluations and comply with 10 CFR 20.1302; 40 CFR Part 190; 10 CFR 50.36a; and 10 CFR Part 50, Appendix I. TS 5.5.2, contained in the ODCM, includes alarm setpoints for effluent monitors; monitoring, sampling, and analysis of liquid and gaseous effluents to comply with 10 CFR 20.1302; determination of cumulative and projected public dose limits from liquid and gaseous effluents and noble gases to comply with 10 CFR Part 50, Appendix I; and annual public dose limits to comply with 40 CFR Part 190. In FSAR Section 11.5.1.4, the applicant specified that the ODCM and radiological environmental monitoring program are developed and implemented in accordance with NEI 07-09A.

11.5.4.1.7 Offsite Dose Calculation Manual

The applicant, in FSAR Section 11.5.1.2, "Effluent Instrumentation Alarm Setpoints," stated that effluent alarm setpoints are determined in accordance with the guidance of NUREG-1301 and NUREG-0133, such that releases do not exceed the limits specified in 10 CFR Part 20, Appendix B, Table 2. In FSAR Section 11.5.1.3, "Effluent Release Controls," the applicant stated that the ODCM is to contain a description of the methodology and parameters used to calculate offsite doses for gaseous and liquid effluents. The applicant stated that an ODCM will be developed and implemented in accordance with the recommendations and guidance of NEI 07-09A.

The NRC staff reviewed the applicant's submittal against the requirements of 10 CFR Part 50, as it relates to a program that provides the means to calculate offsite doses to the public resulting from gaseous and liquid effluents, and found it acceptable. NEI 07-09A describes an acceptable way to develop a facility ODCM and REMP. Given the use of NEI 07-09A in the development of an ODCM, the staff finds this acceptable.

11.5.4.1.8 Radiological Environmental Monitoring Program

The applicant, in FSAR Section 11.5.1.2, stated that effluent alarm setpoints are determined in accordance with the guidance of NUREG-1301 and NUREG-0133, such that releases do not exceed the limits specified in 10 CFR Part 20, Appendix B, Table 2. The applicant, in FSAR Section 11.5.1.4, stated that a REMP will be developed to follow the guidance contained in NEI 07-09A.

The NRC staff reviewed the applicant's submittal against the requirements of 10 CFR Part 20 and 10 CFR Part 50, as it relates to a program that provides the means to monitor and quantify radiation and radioactivity levels in the environs of the plant associated with gaseous and liquid effluent releases and the direct external radiation from contained sources of radioactive materials in tanks and equipment and in buildings. The NRC staff finds the submittal acceptable because the application specifies the commitment to follow NEI 07-09A, which provides the generic template for the ODCM and REMP, as well as a commitment to establishing alarm setpoints in accordance with NUREG-1301 and NUREG-0133.

11.5.5 Combined License Information Items

FSAR Section 11.5 contains no COL items. However, the staff notes that NuScale COL Item 13.5-6 specifies that the COL applicant will develop calibration and test procedures, including for the PERMISS monitors.

The NRC staff finds that this adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items that need to be included in the FSAR for the PERMISS.

11.5.6 Conclusion

The NRC staff has determined that the NuScale design meets the applicable requirements discussed above. The NRC staff concludes that the PERMISS includes the necessary equipment to measure and control releases of radioactive materials in plant process streams and liquid and gaseous effluents; alert control room operators of abnormal levels of radioactivity in process streams and liquid and gaseous effluents; and provide signals that initiate automatic or manual actuations, as appropriate, such as isolating process streams, and terminate effluent discharges if predetermined radioactivity levels or release rates exceed alarm set points. The applicant provided sufficient design information to demonstrate that it has met the requirements of 10 CFR 20.1301; 10 CFR 20.1302; 10 CFR 20.1406; 10 CFR 50.34a; 10 CFR 50.36a; GDC 13, 60, 63, and 64; 10 CFR Part 50, Appendix I; 10 CFR 52.47; and SRP and DSRS Section 11.5 acceptance criteria. This conclusion is based on the following:

- The NuScale design demonstrates compliance with 10 CFR 50.34a and GDC 13, 60, 63, and 64 by providing the means to monitor and control liquid and gaseous effluent releases. The NuScale SDA describes the effluent and process radiation monitoring and sampling provisions based on the system in which they are located. The PERMISS design conforms to the guidelines of SRP Section 11.5. The PERMISS monitors liquid effluent releases through a sole discharge line and gaseous effluent releases to the environment through potential discharge paths.

- Operating in conjunction with the LRWS, GRWS, SRWS, and other plant systems, the PERMISS in the NuScale design is used to control and monitor radioactive effluent releases. The NRC staff determined that it provides the means to comply with the dose limits in 10 CFR 20.1301 and 10 CFR 20.1302 by ensuring that annual average concentrations of radioactive materials in liquid and gaseous effluents released into unrestricted areas will not exceed the effluent concentration limits specified in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2.

- In conjunction with the LRWS, GRWS, SRWS, and other plant systems, the PERMISS in the NuScale design complies with 10 CFR Part 50, Appendix I, Sections II.A, II.B, and II.C, in ensuring that offsite doses resulting from liquid and gaseous effluent releases are ALARA and will not exceed the numerical guides and design objectives in 10 CFR Part 50, Appendix I, and it complies with 10 CFR 50.34a and 10 CFR 50.36a. SER Sections 11.2, 11.3, and 11.4 for the LRWS, GRWS, and SRWS, respectively, address compliance with 10 CFR Part 50, Appendix I, Section II.D, as it relates to the CBA for reducing population doses.

- The NuScale design describes monitoring and sampling of potential release points for routine operation and for accident conditions. The NuScale design monitors the control room conditions for airborne radiation and includes a variety of in-plant area and airborne radiation monitors to monitor both routine operation and potential accident conditions. Finally, NuScale follows the guidance of RG 1.97. As a result, the NRC staff finds these monitors acceptable for meeting the requirements in 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii).

- The PERMISS provides a means to monitor and sample radioactive material release points and to monitor and sample for the potential unintended spread of contamination. As a result, the PREMISS plays a role in addressing the requirements of 10 CFR 20.1406. SER Chapter 12 includes additional information on compliance with 10 CFR 20.1406.

- The NuScale design provides the plans for preoperational testing and initial operations of the PERMISS, including the RCS and steam generator leakage detection instrumentation to comply with the requirements in 10 CFR 50.34(b)(6)(iii) and the ITAAC to comply with the requirements of 10 CFR 52.47(b)(1).

- A COL applicant referencing the NuScale certified design will develop an ODCM that describes the methodology and parameters used to calculate offsite doses for gaseous and liquid effluents, based on the guidance of NEI 07-09A. The COL applicant is responsible for ensuring that the design objectives in 10 CFR Part 50, Appendix I, and the requirements in 10 CFR 20.1301(e), which incorporates by reference 40 CFR Part 190 for facilities within the nuclear fuel cycle, including nuclear power plants, are satisfied.

- A COL applicant referencing the NuScale SDA will develop a REMP using the guidance in NEI 07-09A that will consider local land use census data to identify potential exposure pathways from liquid and gaseous effluents and direct external radiation from SSCs.

11.6 Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring

11.6.1 Introduction/Summary

The NRC staff's evaluation presented in SER Section 11.5 includes the review performed for SER Section 11.6. SER Section 11.5 discusses all relevant requirements and how to comply with them.

11.6.2 Regulatory Basis

SRP Section 11.5 and DSRS Sections 11.5 and 11.6 present the relevant requirements in the NRC regulations.

The guidance in SRP Section 11.5 lists the acceptance criteria adequate to meet the above requirements and review interfaces with other SRP sections.

11.6.3 Technical Evaluation

The NRC staff's evaluation presented in SER Section 11.5 includes the review performed for SER Section 11.6. SER Section 11.5 discusses all relevant requirements and how to comply with them.

11.6.4 Combined License Information Items

Section 11.6 contains no COL items.

11.6.5 Conclusion

SER Section 11.5.6 provides the conclusion applicable to this section. The NRC staff concludes that the PERMISS includes the necessary equipment to measure and control releases of radioactive materials in plant process streams and liquid and gaseous effluents; alert control room operators of abnormal levels of radioactivity in process streams and liquid and gaseous effluents; and provide signals that initiate automatic safety functions, isolate process streams, and terminate effluent discharges if predetermined radioactivity levels or release rates exceed alarm set points, as discussed in SER Section 11.5.