11 RADIOACTIVE WASTE MANAGEMENT

This chapter of the final safety evaluation report (FSER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 11, "Radioactive Waste Management," of the NuScale Power, LLC (hereinafter referred to as the applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 5 of the DCA, dated July 29, 2020 (Agencywide Document Access and Management System (ADAMS), Accession No. ML20225A071). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the DCA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this safety evaluation to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the DCA and not converted.

This chapter describes the results of the NRC staff's review of the NuScale design-basis and realistic radioactive source terms and radioactive waste management systems (RWMS).

The RWMS in the NuScale design consists of the liquid radioactive waste system (LRWS), gaseous radioactive waste system (GRWS), solid radioactive waste system (SRWS), and process and effluent radiation monitoring instrumentation and sampling system (PERMISS). 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 <u>Source Terms</u>

11.1.1 Introduction

Operation of the NuScale Power Plant, which is designed to operate with up to 12 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, 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 conducting shielding analyses of structures, systems, and components (SSCs); establishing radiation zones; and reevaluating 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 of 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, for large light-water reactors (LWRs). Therefore, the realistic coolant source term is developed using an alternate methodology to the GALE86 code. This methodology is reviewed by the NRC staff in Section 11.1.4 of this SER.

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 are produced 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 for Light Water Reactors."

In support of the NRC staff's review of the applicant's proposed alternate methodology described in NuScale technical report TR-1116-52065-P, Revision 1, "Effluent Release (GALE Replacement) Methodology and Results," the NRC issued an audit plan on May 8, 2018 (ADAMS Accession No. ML18116A603), requesting access to nondocketed information, including calculation packages and Microsoft® Excel spreadsheets in native format. 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. It also facilitated the NRC staff's evaluation of applicant responses to electronic requests for additional Information (RAIs) related to the alternate methodology. The NRC audit report documents the NRC staff's and applicant's interactions and audit observations (ADAMS Accession No. ML18330A232).

11.1.2 Summary of Application

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

DCA Part 2, Tier 2: The applicant described the design-basis and realistic coolant source terms in DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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-1116-52065-P, Revision 1, used in developing the realistic coolant source terms.

Fission products, water activation products, and CRUD that 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; and Electric Power Research Institute (EPRI) TR-1009903, "Tritium Management Model." DCA Part 2, Tier 2, 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 (DBFFF) described in Section 11.1.4 of this SER. DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC items with this area of review.

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

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

Technical Reports: There is a technical report associated with this area of review described in NuScale TR-1116-52065-P, Revision 1.

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

11.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for the source terms 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 "Design-Specific Review Standard for NuScale SMR [Small Modular Reactor] Design" (DSRS), Section 11.1, "Source Terms," issued June 2016 (ADAMS Accession No. ML15355A333). The following summarizes 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, 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

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, 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 DCA Part 2, Tier 2, Revision 3, Section 11.1, and NuScale TR-1116-52065-P, Revision 1, against the applicable NRC regulations and guidance in DSRS Section 11.1.

The NRC staff's review of DCA Part 2, Tier 2, Revision 3, Section 11.1, and NuScale TR-1116-52065-P, Revision 1, found that the applicant provided this information as a replacement methodology for the GALE 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, 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. For the determination of the CRUD isotopes, the applicant used the ANSI/ANS-18.1-1999 standard 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 SCALE, TRITON, and ORIGEN computer codes to determine the isotopic distribution.

In addition to the source term data provided by the applicant, the applicant described the plant parameters necessary to calculate coolant source terms in DCA Part 2, Tier 2, Table 11.1-2. The staff held discussions during public meetings and audits with NuScale and verified that the information contained in DCA Part 2, Tier 2, Table 11.1-2, is accurate and can be used to calculate subsequent coolant source terms. 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 and documented its findings in the NRC audit report of the NuScale effluent release (GALE replacement) methodology (ADAMS Accession No. ML18330A232). The NRC staff used the audit to understand and verify that various source terms for water activation, CRUD, and fission product radionuclides were included in FSAR tables by reviewing and ensuring that those radionuclides that could have an impact on effluent release doses, and were calculated in applicant spreadsheets, were included in FSAR tables.

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-1116-52065-P 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. During the audit with NuScale, the NRC staff contacted EPRI for questions on industry fuel failure data that the applicant used in NuScale TR-1116-52065-P. 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). Although NuScale TR-1116-52065-P presents data and statistical analyses using historical fuel failure data (e.g., early 1970s through 1999), the EPRI source of validated data from FRED only dates back to 2000. 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 ppm (from 2007–2016 for U.S. PWRs).

The NRC staff and applicant discussed an acceptable approach to determine a conservative RFFF for the NuScale design, and the applicant agreed to use EPRI FRED data from 2007 through 2016 (10 years) for U.S. PWRs, and the maximum fuel failure value determined from those 10 years. The NRC staff finds that the proposed RFFF is 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. The NRC staff documented the results of this audit, including the staff's review of the FRED database, in an audit report dated April 30, 2018 (ADAMS Accession No. ML18103A198).

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, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures." 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 License (COL) Item 11.2-3.

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 also discusses the varying concentrations of tritium in DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 11.2, Section 11.3, and Table 12.2-23, and NuScale TR-1116-52065-P. Based on the staff's assessment of the tritium concentrations in DCA Part 2, Tier 2, Table 11.1-8, the staff finds that the applicant has appropriately calculated tritium concentrations for subsequent doses analysis in DCA Part 2, Tier 2, Sections 11.2, 11.3, and 12.2.

11.1.5 Combined License Information Items

There are no COL information items in this section.

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 the applicant's alternate method described in NuScale TR-1116-25065-P for calculating liquid and gaseous effluent releases during normal operations, including AOOs. The NRC staff further determined that these coolant source terms processed by the LRWS and GRWS, as discussed in Sections 11.2 and 11.3, respectively, of this SER, will meet the applicable requirements of 10 CFR Part 20; 10 CFR Part 50, Appendix A, GDC 60; 10 CFR Part 50, Appendix I; 10 CFR 50.34(a) and (b); 10 CFR 52.47(a)(5); 10 CFR 52.79(a)(3); and 10 CFR 52.79(a)(1)(i) and (ii). This conclusion is based on the following:

- The NuScale alternate method for developing normal liquid and gaseous effluent source terms is acceptable and 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 emergency classification levels

(ECLs); 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 that is 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.

The NRC staff finds that the source terms in the NuScale design, as also discussed in Sections 11.2 and 11.3 of this SER, are acceptable, and their use in calculating doses associated with normal operations, 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."

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 satisfactory 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 (CVCS), the SRWS, the containment evacuation system (CES), the reactor component cooling water system (RCCWS), 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

DCA Part 2, Tier 1: The Tier 1 information associated with this section is in DCA Part 2, Tier 1, Section 3.12, "Radioactive Waste Building."

DCA Part 2, Tier 1, Section 3.12, provides the ITAAC associated with the design of the radioactive waste building (RWB). The ITAAC verify that the building is built to seismic category RW-IIa as described 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. This conforms to the information and conclusions made for RWMS seismic categories for SSCs in the RWB and as discussed in DCA Part 2, Tier 2 Sections 11.2, 11.3, and 11.4.

DCA Part 2, Tier 2: The applicant described the system in DCA Part 2, Tier 2, Section 11.2, summarized as follows.

In DCA Part 2, Tier 2, 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 as a result of normal operation, including AOOs. The LRWS, located in the reactor building and RWB, is a nonsafety-related system. The reactor building is a seismic Category I structure, and the RWB is a seismic Category II structure that also meets the guidance of RG 1.143 for an RW-IIa safety classification (evaluated in Section 3.2.1 of this SER). 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 Chapter 17 of this SER) ensures that LRWS equipment and installation are in accordance with the codes and standards of RG 1.143, which are listed in DCA Part 2, Tier 2, Table 11.2-10, "Codes and Standards from Regulatory Guide 1.143, Revision 2, Table 1," for SSCs in radioactive waste facilities.

DCA Part 2, Tier 2, Figures 11.2-1a 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): reactor-coolant-grade wastewater collected from the primary coolant system letdown through the CVCS and RWDS equipment drains
- high-conductivity waste (HCW): RWDS floor drains, SRWS decant water, RCCWS, and other liquid wastes with potentially high dissolved and suspended solids content collected from various drains and sumps
- detergent waste: detergent wastes from hand decontamination processes and personnel decontamination showers
- chemical waste: waste collected by the RWDS and manually introduced into the LRWS after sampling and analysis (Section 9.3.3)

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 qualified coating meeting the requirements of RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants." 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 the quality of the liquid before recycling or release. If the water quality is not acceptable, the water is returned to a collection tank for further treatment.

The LRWS design in DCA Part 2, Tier 2, 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 with provisions for local monitors. The LRWS operates on a batch basis with manual start and automatic stops. Parameters such as tank levels, processing flow rates, differential pressures across filters, and ion exchange columns are indicated, alarmed, or both, to provide information on operational and equipment performance. DCA Part 2, Tier 2, 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 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 DCA provides the text and tables in DCA Part 2, Tier 2, Section 11.2, to present information supporting the development of the liquid source term, as well as compliance with the ECLs 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 Section 11.2 of this SER, 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.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: TS 5.5.1 and 5.5.2 are associated with the LRWS and are found in DCA Part 4, Volume 1. The TS also include the following reports: 5.6.1, “Annual Radiological Environmental Operating Report,” and 5.6.2, “Radioactive Effluent Release Report.”

Technical Reports: There is a technical report associated with this area of review described in NuScale TR-1116-52065-P.

Topical Reports: There are no topical reports 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 ECLs 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

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, 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, 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, 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, “Combined License Applications for Nuclear Power Plants,” 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, 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, 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, issued 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, as it relates to guidance for calculating doses from PWR plants

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

- Nuclear Energy Institute (NEI) 08-08A, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination,” Revision 0, and RG 4.21, as they relate to acceptable levels of detail and content needed to demonstrate compliance with 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 DCA Part 2, Tier 2, Revision 3, Section 11.2, against the applicable NRC regulations and guidance in SRP Section 11.2 and in DSRS Section 11.2.

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, as well as 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 by providing 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 RG 1.143, Revision 2, guidance specified by the applicant in the FSAR. The applicant stated that the LRWS will conform to Regulatory Position C.7 in RG 1.143, Revision 2. DCA Part 2, Tier 2, Table 3.2-1, also identifies the seismic category, quality group, and safety class for components of the LRWS. The QA program is stated to be designed in accordance with ANSI/ANS-55.6, “Liquid Radioactive Waste Processing Systems for Light Water Reactor

Plants," which agrees with RG 1.143, Revision 2, Regulatory Position C.7. In determining the design for radwaste systems, the applicant provided DCA Part 2, Tier 2, Table 11.2-1, to reflect the guidance specified in RG 1.143, Revision 2, to meet 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."

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 valves. This definition is consistent with system boundaries reviewed in the past, and the NRC staff finds this boundary definition acceptable.

The applicant provided source terms for various component inventories and radwaste safety classifications for each radwaste SSC in DCA Part 2, Tier 2, Tables 3.2-1, 11.2-11, 11.2-13, 11.3-2, 11.4-1, and 11.4-6. The NRC staff reviewed the information, performed a confirmatory analysis using the source term inventories, and verified the safety classifications of each installed radwaste SSC. 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 in its definition 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 also by identifying the system boundaries. The applicant provided descriptions of the LRWS in DCA Part 2, Tier 2, Section 11.2.2, to describe the planned pathways for a release to the environment. The radiation monitoring to limit or control releases to the environment is discussed in SER Section 11.5. The applicant also provided DCA Part 2, Tier 2, Tables 11.2-5 and 11.2-7, 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 per 10 CFR 50.34a(e)(2). The staff's evaluation of the effluent release results is discussed in the next section.

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

The NRC staff reviewed DCA Part 2, Tier 2, Section 11.2, 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's tables and LADTAP II input and output files, the NRC staff performed a confirmatory analysis of the liquid effluent release doses in DCA Part 2, Tier 2, Table 11.2-7.

The facility design reduces liquid radioactive effluent discharges from the LRWS to the environment by processing liquid wastes from the CVCS, SRWS, CES, RCCWS, and RWDS 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 by providing 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 its tables.

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 E-5, "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 two effluent release dilution flow rates. The distinction between these two values, which were pertinent to the verification of the calculation of the liquid effluent release dose, are used to calculate the discharge concentrations and an assumed river flowrate for the subsequent dose calculations. The NRC staff performed confirmatory calculations based on the information provided, and the results are summarized in Table 11.2-1 below.

In DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Table 11.2-5, for the liquid effluent curies per year (Ci/yr) source term for the analysis. The NRC staff confirmed this source term and the results produced by NuScale's effluent release methodology in Section 11.1 of this SER. The NRC staff also confirmed the use of the RG 1.109 default methodology values since this design does not have site-specific values to reference for the calculation.

To confirm the results provided by the applicant in DCA Part 2, Tier 2, Table 11.2-7, the NRC staff used the LADTAP II inputs reported in DCA Part 2, Tier 2, Table 11.2-6. The results of the NRC staff's confirmatory calculation verified the estimated total body, organ, and age group doses. In DCA Part 2, Tier 2, Section 11.2.3.1 and DCA Part 2, Tier 2, 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 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 Results	Design Objective
Total Body	0.028 (2.8)	0.0276 (2.76)	0.03 (3)
Max Organ Exposed Organ Age Group	0.098 (9.8) Bone Child	0.0976 (9.76) Bone Child	0.1 (10)

The NRC staff's review determined that the applicant had accurately 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 at any chosen site. A COL applicant that references the NuScale Power Plant DC will perform a site-specific evaluation using the site-specific dilution flow. The NRC staff finds these results acceptable, since a COL applicant referencing this design will address site-specific conditions, consistent with COL Item 11.2-4. Following COL Item 11.2-4, the COL applicant is to calculate doses to members of the public following the guidance of RG 1.109 and RG 1.113, using site-specific parameters, and to 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

DCA Part 2, Tier 2, Section 11.2.3.4, "Site-Specific Cost-Benefit Analysis," describes the LRWS design for use at any site with flexibility to incorporate site-specific requirements with minor modifications, such as technology preference, degree of automated operation, and radioactive waste storage. 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. The applicant stated that, for the NuScale designs, the CBA demonstrates that the addition of items of reasonably demonstrated technology will not provide a more favorable cost benefit but does not include the CBA in DCA Part 2, Tier 2, Section 11.2.3.4. Under COL Item 11.02-5, the COL applicant will provide the site-specific CBA to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix I, Sections II.A and II.D. COL Item 11.02-5 states that a COL applicant that references the NuScale Power Plant DC will perform a CBA as required by 10 CFR 50.34a and 10 CFR Part 50, Appendix I, to demonstrate conformance to regulatory requirements. The applicant is to perform the CBA using the guidance of RG 1.110.

Because the CBA requires site-specific information, which is outside the scope of the FSAR, the NRC staff finds the inclusion of COL Item 11.2-5 acceptable.

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

FSAR Table 11.2-8, "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 ECL 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 11.2-8 and verified that the unity rule calculation was less than one for all radionuclides identified in Table 11.2-8. The results of the NRC staff's confirmatory analysis determined the information contained in Table 11.2-8 is acceptable and the applicant has demonstrated compliance with the 10 CFR Part 20, Appendix B, effluent concentration limits.

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 11.2-2. COL Item 11.2-2 states that a COL applicant that references the NuScale Power Plant DC 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 11.2-2, 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 DCA Part 2, Tier 2, Section 11.2, against the criteria in 10 CFR 20.1406 for minimizing contamination. Compliance with 10 CFR 20.1406 is achieved when the applicant identifies those design features used to minimize the release of radioactive liquid to the environment.

In DCA Part 2, Tier 2, Section 11.2.1, "Design Basis," the applicant described how, for consistency with 10 CFR 20.1406, 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.

Additionally, the applicant included a COL item (COL Item 11.2-1) stating the following:

> A COL applicant that references the NuScale Power Plant design certification will ensure mobile equipment used and connected to plant systems is in accordance with ANSI/ANS-40.37, Regulatory Guide (RG) 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10 and 10 CFR 50.34a.

In FSAR Section 11.2.2.4, the applicant noted that design features are provided in accordance with the requirements of 10 CFR 20.1406, following the guidance of RG 4.21 to the extent practicable, to reduce contamination of the facility and the environment, facilitate eventual decommissioning, and reduce the generation of radioactive waste. FSAR Section 12.3.6 and SER Section 12.3 provide additional information on compliance with 10 CFR 20.1406.

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. Considering the potential future use of mobile or temporary equipment in accordance with site-specific requirements, the LRWS provides connections for such equipment.

Under COL Item 11.2-1, the COL applicant is to determine that the use of mobile or temporary equipment and interconnections to plant systems conform to regulatory requirements in 10 CFR 50.34a, 10 CFR 20.1406, and the guidance in RG 1.143, Revision 2. The NRC staff finds the inclusion of this COL item would be acceptable for meeting the guidance of RG 1.143, Revision 2, and the requirements of 10 CFR 50.34a and 10 CFR 20.1406 by ensuring that potential mobile or temporary equipment meets the design guidance for SSCs containing radioactive waste.

The COL applicant is responsible for identifying mobile or portable LRWS connections that are considered nonradioactive but may later become radioactive through contact with or contamination by radioactive systems and for the preparation of operating procedures for mobile or portable LRWS connections in conformance with the guidance in IE Bulletin No. 80-10. Under COL Item 11.2-1, an applicant will be responsible for preparing and providing the piping and instrumentation diagrams to show the connections between existing systems and the mobile or temporary equipment that may become contaminated.

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. This does not require a reevaluation of LRWS with limiting conditions or controls for operation based on more conservative analysis and calculation assumptions.

The applicant identified the pool surge control system (PSCS) storage tank as the limiting source of radioactive material in DCA Part 2, Tier 2, Section 11.2.3.2, and identified the radioactive source term found in FSAR Table 12.2-10 and the volume and description of the tank in FSAR Section 9.1.3.2.4. The NRC staff's review of DCA Part 2, Tier 2, Section 11.2.3.2, confirms that the limiting radiological consequence from a failure of an active component could occur from the PSCS tank containing radioactive material.

Thus, the applicant provided COL Item 11.2-3 in DCA Part 2, Tier 2, 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 transport radionuclides at a site. The NRC staff finds that the postulated failure of a tank and its associated components and the design acceptable to meet the requirements of GDC 60 and 61 for the control of releases of radioactive materials to the environment and provide an adequate level of safety during normal reactor operation, including AOOs, because the applicant described COL Item 11.2-3, which will have a COL applicant perform an analysis in accordance with BTP 11-6.

11.2.4.1.9 Technical Specifications

From the review of DCA Part 2, Tier 2, Revision 3, Chapter 16, "Technical Specifications", the NRC staff determined that there are no TS directly associated with liquid waste storage and processing. However, DCA Part 2, Tier 2, Chapter 16, TS 5.5.1, requires an established, implemented, and maintained offsite dose calculation manual (ODCM). TS 5.5.2 has provisions for a 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.

DCA Part 2, Tier 2, Chapter 16, 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 liquid effluents released to the environment. As stated in TS 5.5.1, COL licensee-initiated changes to the ODCM must be justified by calculation, and changes will maintain levels of radioactivity in effluents that 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 would also require the radioactive effluent controls program, which is contained 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.

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. A COL applicant will address the implementation of such programs in a plant- and site-specific ODCM under COL Item 11.5-1, as described in DCA Part 2, Tier 2, Table 1.8-2. Section 11.5 of this SER contains the NRC staff's evaluation of the applicant's proposed ODCM.

11.2.5 Combined License Items

Table 11.2-2 of this SER lists the COL items, descriptions, and DCA Part 2, Tier 2 section for the LRWS, from DCA Part 2, Tier 2, Table 1.8-2.

Table 11.2-2 NuScale COL Items for DCA Part 2, Tier 2, Section 11.2

COL Item No.	COL Item Description	DCA Part 2, Tier 2 Section No.
11.2-1	A COL Applicant that references the NuScale Power Plant design certification will ensure mobile equipment used and connected to plant systems is in accordance with ANSI/ANS-40.37, Regulatory Guide (RG) 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10 and 10 CFR 50.34a.	11.2.1
11.2-2	A COL Applicant that references the NuScale Power Plant design certification 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
11.2-3	A COL applicant that references the NuScale Power Plant design certification 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
11.2-4	A COL Applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation using the site-specific dilution flow.	11.2.3.3
11.2-5	A COL Applicant that references the NuScale Power Plant design certification will perform a cost-benefit analysis as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory requirements. This cost-benefit analysis is to be performed using the guidance of RG 1.110.	11.2.3.4

The NRC staff finds the above list to be complete. Also, the list adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items that need to be included in DCA Part 2, Tier 2, Table 1.8-2, for the LRWS.

11.2.6 Conclusion

The NRC staff has determined that the NuScale design meets the applicable requirements discussed in Section 11.2 of this SER. 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 it has met the requirements of 10 CF 20.1301; 10 CFR 20.1302; 10 CFR 20.1406; 10 CFR Part 20, Appendix B, Table 2 values; 10 CFR 50.34a; 10 CFR 50.36, "Technical Specifications"; 10 CFR Part 50, Appendix A, GDC 60, 61, and 64; 10 CFR Part 50 Appendix I; 10 CFR 52.47, "Contents of Applications; Technical Information"; 40 CFR Part 190, 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 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.

- A COL applicant referencing the NuScale certified design will demonstrate compliance with 10 CFR 20.1406 and 10 CFR 50.34a and conform with NRC IE Bulletin No. 80-10 and industry guidance in ANSI/ANS-40.37-2009, "Mobile Low-Level Radioactive Waste Processing Systems," for mobile equipment used and connected to plant systems under COL Item 11.2-1, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified 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 11.2-2, as described in DCA Part 2, Tier 2, Table 1.8-2.

- 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.

- A COL applicant referencing the NuScale certified design will perform a site-specific evaluation using the site-specific dilution flow to calculate offsite doses from liquid effluents and ensure compliance with the ECLs in 10 CFR Part 20, Appendix B, Table 2, Column 2; the dose limits to members of the public in 10 CFR 20.1302 and 10 CFR 20.1301(e); and the ALARA design objectives in 10 CFR Part 50, Appendix I, under COL Item 11.2-4, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified 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 11.2-5, as described in DCA Part 2, Tier 2, Table 1.8-2.

- 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 LRWS components to support the staff's finding on GDC 60 and 61.

- A COL applicant referencing the NuScale certified design will develop an ODCM that describes the methodology and parameters for calculating offsite doses for gaseous and liquid effluents, using the guidance of NEI 07-09A, "Generic FSAR Template Guidance for the Offsite Dose Calculation Manual (ODCM) Program Description," Revision 0, as stated in COL Item 11.5-2. The COL applicant is responsible for ensuring that the design objectives in 10 CFR Part 50, Appendix I, and compliance with 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 under COL Item 11.5-2, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified design will perform a site-specific evaluation for a postulated accidental release from the failure of a tank containing radioactive liquids 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 11.2-3, as described in DCA Part 2, Tier 2, Table 1.8-2.

11.3 Gaseous Waste Management System

11.3.1 Introduction

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 reactor building 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. Design and performance of the charcoal delay system are in accordance with NUREG-0017, Revision 1, as modified by NuScale TR-1116-52065-P, Revision 1.

Discharge from the GRWS is initially diluted by nitrogen flow in the GRWS and then diluted by the airflow through the plant exhaust stack in the RBVS. The gaseous effluents meet the objectives of 10 CFR Part 50, Appendix I, and 10 CFR Part 20, Appendix B. Small amounts of radioactive or hydrogen-bearing gaseous wastes not directly collected by the GRWS are captured by the plant HVAC systems. Exhaust flow from the RWBVS and RBVS are combined and monitored by the RBVS exhaust stack radiation effluent monitor before release to the environment (see Section 11.5). Primary gaseous effluent sources, besides gaseous radioactive waste, include the CES (FSAR Section 9.3.6), RWBVS (FSAR Section 9.4.3), and other sources exhausted by the RBVS (FSAR Section 9.4.2). In addition, small releases that occur in the turbine generator building from the main condenser air removal system (CARS) (Section 10.4.2) and turbine gland sealing system (Section 10.4.3) are monitored but directly released to the environment, as illustrated in FSAR Figure 11.5-1.

11.3.2 Summary of Application

DCA Part 2, Tier 1: The Tier 1 information associated with this section is found in DCA Part 2, Tier 1, Section 3.12.

DCA Part 2, Tier 1, Section 3.12, provides the ITAAC associated with the design of the RWB. The ITAAC verify that the building is built to seismic category RW-IIa as described in RG 1.143. This conforms to the information and conclusions made for RWMS seismic categories for SSCs in the RWB and as discussed in FSAR Sections 11.2, 11.3, and 11.4.

DCA Part 2, Tier 2: The applicant described the system in DCA Part 2, Tier 2, Section 11.3, summarized as follows.

In DCA Part 2, Tier 2, 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 nonsafety-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 RWB, which houses the GRWS, is designed to seismic Category II requirements, evaluated in Section 3.2 of this SER. FSAR Table 11.3-3, “Gaseous Radioactive Waste System Equipment Malfunction Analysis,” describes the failure scenarios considered for the GRWS.

DCA Part 2, Tier 2, Figures 11.3-1a and 11.3-1b, “Gaseous Radioactive Waste System Diagram,” present the process design of the GRWS. DCA Part 2, Tier 2, Table 11.3-1, “Gaseous Radioactive Waste System Design Parameters,” lists various system nominal values, including element delay times, absorption coefficients, temperatures, and flow rates. DCA Part 2, Tier 2, Table 11.3-2, “Major Equipment Design Parameters,” lists information relating to, but not limited to, design pressures, design temperatures, flow rates, and component materials for the following system components: gas coolers, charcoal guard bed, charcoal decay bed vessels, moisture separators, and the charcoal drying heater.

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 via 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 via the RWBVS for release to the environment through the plant exhaust stack as a monitored release (see Section 11.5).

The key components of the GRWS described in DCA Part 2, Tier 2, Section 11.3.2.1, include the following:

- two waste gas coolers
- two moisture separators
- one charcoal guard bed
- eight charcoal decay beds
- 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 1 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 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.

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 associated with radioactive gaseous effluent discharges, DCA Part 2, Tier 2, 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 ECLs 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 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, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," from a GRWS component containing radioactivity.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The following TS are associated with the GRWS and are found in Part 4, Volume 1, of the DCA: 3.4.8, 5.5.1, 5.5.2, and 5.5.6.

The TS also includes the following reports: 5.6.1 and 5.6.2.

Technical Reports: There is a technical report associated with this area of review described in NuScale TR-1116-52065-P.

Topical Reports: There are no topical reports 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) ECLs 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 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

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, 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, 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, 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, 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, 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 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 DCA Part 2, Tier 2, Revision 3, Section 11.3, against the applicable NRC regulations and guidance in SRP Section 11.3 and in DSRS Section 11.3.

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 gaseous delay beds. The applicant also discussed the retention ability of the beds for krypton and xenon gases. GDC 64 is met by providing gaseous effluent radiation monitors to check the effluent discharge from the reactor building HVAC, CARS exhaust, turbine gland sealing system exhaust, PSCS, and auxiliary boiler system normal operations and AOOs.

In assessing compliance with GDC 60 and 61, the NRC staff reviewed the QA provisions and RG 1.143, Revision 2, guidance specified by the applicant in the FSAR. The applicant stated that the GRWS will conform to Regulatory Position C.7 in RG 1.143, Revision 2, and RG 1.33, which specify the QA guidance to follow. DCA Part 2, Tier 2, Table 3.2-1, also identifies the seismic category, quality group, and safety class for components of the GRWS. In determining the design for radwaste systems, the applicant provided DCA Part 2, Tier 2, Table 11.3-2, to reflect the guidance specified in RG 1.143, Revision 2, to meet the values for A_1 and A_2 in 10 CFR Part 71. 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's confirmatory calculations were performed using the listed activities in DCA Part 2, Tier 2, Table 11.3-2. The NRC staff verified the listed RG 1.143 safety classification category for each equipment listed in this table. The NRC staff determined that the applicant has appropriately calculated the seismic categories of GRWS components using the guidance contained in RG 1.143 to satisfy 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 gaseous waste and also by identifying the system boundaries. The applicant provided descriptions of the GRWS in DCA Part 2, Tier 2, Section 11.3.2, to describe the planned pathways for a release to the environment. The radiation monitoring to limit or control releases to the environment is discussed in SER Section 11.5. The applicant also provided DCA Part 2, Tier 2, 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 staff's evaluation of the effluent release results is discussed in the next section.

In assessing compliance with GDC 3, the NuScale design demonstrates compliance with the requirements for the design of the gaseous waste-handling and treatment systems to minimize the generation of explosive gas mixtures and effects of explosive mixtures of hydrogen and oxygen on subsystems and components by using 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.

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 applicant-provided tables and GASPAR II input and output files, the NRC staff performed confirmatory analysis of the gaseous effluent release doses in DCA Part 2, Tier 2, Table 11.3-8, 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 GRWS is initially diluted by nitrogen flow in the GRWS and then diluted by the airflow through the plant exhaust stack in the RBVS. The RBVS exhaust stack radiation effluent monitor 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 other sources exhausted by the RBVS. In addition, other 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 parameters described by the applicant to perform a confirmatory calculation of the gaseous effluent release doses based on the plant design.

In DCA Part 2, Tier 2, Table 11.3-6, the applicant summarized the inputs used for the GASPAR II code. This table includes a pointer to DCA Part 2, Tier 2 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 Section 11.1 of this SER. The NRC staff confirmed the use of the RG 1.109 default values, and assumed parameters for the NuScale design, as referenced in DCA Part 2, Tier 2, Table 11.3-6, for its confirmatory calculation. In addition to the values in RG 1.109, the applicant referenced the X/Q and D/Q values from DCA Part 2, Tier 2, Table 2.0-1.

To confirm the applicant's results in DCA Part 2, Tier 2, Table 11.3-8, the NRC staff used the GASPAR II inputs reported in DCA Part 2, Tier 2, 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. In DCA Part 2, Tier 2, Table 11.3-8, the applicant provided its results, which are summarized in Table 11.3-1 of this SER. This table 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 Results	Design Objectives
Gamma Air	7.0×10^{-4} mGy (7.0×10^{-2} mrad)	7.0×10^{-4} mGy (7.0×10^{-2} mrad)	1×10^{-1} mGy (10 mrad)
Beta Air	2.0×10^{-3} mGy (2.0×10^{-1} mrad)	2.3×10^{-3} mGy (2.3×10^{-1} mrad)	2×10^{-1} mGy (20 mrad)
Total Body	5.0×10^{-4} mSv (5.0×10^{-2} mrem)	5.0×10^{-4} mSv (5.0×10^{-2} mrem)	5×10^{-2} mSv (5 mrem)
Skin	2.2×10^{-3} mSv (2.2×10^{-1} mrem)	2.1×10^{-3} mSv (2.1×10^{-1} mrem)	1.5×10^{-1} mSv (15 mrem)
Max Organ (Thyroid)	2.92×10^{-2} mSv (2.92 mrem)	2.92 x 10-2 mSv (2.92 mrem)	1.5 x 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 DC 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 NRC staff guidance in the SRP, 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

DCA Part 2, Tier 2, Section 11.3.2.5, "Site-Specific Cost-Benefit Analysis," describes the GRWS design for use at any site with flexibility to incorporate site-specific requirements with minor modifications, such as technology preference, degree of automated operation, and radioactive waste storage. RG 1.110 describes an acceptable method of performing a CBA to demonstrate 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 applicant stated that the CBA for the NuScale design demonstrates that the addition of items of reasonably demonstrated technology will not provide a more favorable CBA result but did not include a CBA in DCA Part 2, Tier 2, Section 11.3.2.5. The CBA requires site-specific information, which is outside the scope of the requested DC. The COL applicant will provide the site-specific CBA to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix I, Sections II.A and II.D, under COL Item 11.3-1. The CBA is to be performed using the guidance of RG 1.110. 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

DCA Part 2, Tier 2, 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 ECL for each radionuclide released in gaseous effluent met the "unity" rule calculation specified in Note 4 of 10 CFR Part 20. The NRC staff notes 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 DCA Part 2, Tier 2, Table 11.3-5. The results of the NRC staff's confirmatory analysis determined the information contained in DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 11.3.3, 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 DC, 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 DCA Part 2, Tier 2, Section 11.3.2.3, "Design Features," against the criteria in 10 CFR 20.1406 for minimizing contamination. Compliance with 10 CFR 20.1406 is achieved when the applicant identifies those design features used to minimize the release of radioactive gases to the environment.

The applicant provided DCA Part 2, Tier 2, Section 11.3.2.3, to meet the requirements of 10 CFR 20.1406. In this section, the applicant committed to the use of RG 4.21 and to the principles described in the RG to demonstrate compliance with 10 CFR 20.1406. DCA Part 2, Tier 2, Section 12.3.6, and Section 12.3 of this SER provide additional information on compliance with 10 CFR 20.1406.

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 DCA Part 2, Tier 2, Section 11.3.3.1, 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 DCA Part 2, Tier 2, Table 11.3-9, which describes the release source term, the assumed dispersion factor, and the anticipated offsite dose consequence. DCA Part 2, Tier 2, Section 11.3.3.1, states that the assumed source term is based on a 1-percent failed fuel fraction and presents the information contained in Table 11.3-9 as the source term used for the analysis. In 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 1.46 x 10^{-2} millisieverts (mSv) (1.46 millirem (mrem)). The applicant stated that its expected offsite dose consequence is less than 1 x 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

From the review of DCA Part 2, Tier 2, Revision 3, Chapter 16, the NRC staff determined that TS 5.5.12 would require 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 DCA Part 2, Tier 2, Chapter 16, TS 5.5.1 and TS 5.5.4, “Steam Generator Program,” have provisions for direction in managing releases of radioactive effluents and the control and handling of concentrated wastes for disposal. 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 contained 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 under COL Item 11.5-2, as described in DCA Part 2, Tier 2, Chapter 1, Table 1.8-2. Section 11.5 of this SER contains the NRC staff’s evaluation of the applicant’s proposed ODCM.

11.3.5 Combined License Items

Table 11.3-2 lists COL item numbers and descriptions related to the GRWS, from DCA Part 2, Tier 2, Table 1.8-2.

Table 11.3-2 NuScale COL Items for DCA Part 2, Tier 2, Section 11.3

COL Item No.	COL Item Description	DCA Part 2, Tier 2 Section No.
11.3-1	A COL Applicant that references the NuScale Power Plant design certification will perform a site-specific cost-benefit analysis.	11.3.2.5
11.3-2	A COL Applicant that references the NuScale Power Plant design certification 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	A COL Applicant that references the NuScale Power Plant design certification 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 the above list to be complete. Also, the list adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items that need to be included in DCA Part 2, Tier 2, Chapter 1, Table 1.8-2, 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 it has met 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; and NRC guidance 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 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. This commitment fulfills the requirements of 10 CFR 20.1406 and the guidance of RGs 4.21 and 1.143 in minimizing the contamination of the facility and generation of radioactive wastes. 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.

- The NuScale design meets the requirements of 10 CFR Part 50, Appendix A, GDC 60, with respect to controlling releases of gaseous effluents by monitoring radioactive gas discharges through the plant exhaust stack via 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. If high radiation is detected, the GRWS outlet valve is closed to stop system flow to the RWBVS.

- 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 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.

- A COL applicant referencing the NuScale certified design will develop 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, as stated in COL Item 11.5-2. The COL applicant is responsible for ensuring that the design objectives in 10 CFR Part 50, Appendix I, and compliance with 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 under COL Item 11.5-2, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified 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, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified 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, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified design will perform a site-specific evaluation for a postulated accidental release of gaseous radionuclides from failure of the GRWS, in accordance with the guidance in BTP 11-5. 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 11.3-2, as described in DCA Part 2, Tier 2, Table 1.8-2.

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. The SRWS processes solid waste by dewatering, decontaminating, surveying, sorting, and classifying solid waste for storage and eventual shipment to licensed offsite facilities.

11.4.2 Summary of Application

DCA Part 2, Tier 1: The Tier 1 information associated with this section is found in DCA Part 2, Tier 1, Section 3.12.

DCA Part 2, Tier 1, Section 3.12, provides the ITAAC associated with the design of the RWB. The ITAAC verify that the building is built to seismic category Rw-IIa, as described in RG 1.143. This conforms to the information and conclusions made for RWMS seismic categories for SSCs in the RWB and as discussed in DCA Part 2, Tier 2, Sections 11.2, 11.3, and 11.4.

DCA Part 2, Tier 2: The applicant described the system in DCA Part 2, Tier 2, Section 11.4, summarized as follows.

The SRWS is located in the RWB, which is classified as an Rw-IIa building (see FSAR Section 3.7.2) and has adequate space for onsite storage of various solid waste containers plus space for mobile processing equipment. The SRWS includes the wet solid waste (WSW) system, dry solid waste (DSW) system, mixed waste system, and an onsite storage area.

The SRWS is a nonsafety-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 CVCS, pool cleanup system, and LRWS, using both permanently installed and mobile equipment in the SRWS. Wastes from these systems are mainly WSW and consist of spent resins, spent charcoal, cartridge filters, filter membranes, and sludge.

- Collect, segregate, sample, package, and store compactible and noncompactible DSW, including HVAC filters, failed tools and equipment parts, used personnel protective equipment, rags, paper, plastic, rubber, scrap wood, concrete, and metal.

- 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.

DCA Part 2, Tier 2, 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 break pot tanks, the two PST break pot tanks, the two SRST transfer pumps, and the two PST transfer pumps.

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 process filter cartridges, tubular ultrafiltration and reverse osmosis filter membranes, and granular activated charcoal

- 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.

The DSW includes HVAC filters, tools and equipment, used personnel protective equipment, rags, paper, wood, and miscellaneous cleaning supplies. Noncontaminated wastes are normally separated from contaminated DSW as close as possible to the point of generation. Additional sorting is done to segregate waste types to facilitate efficient packaging and monitoring of the waste for classification according to 10 CFR 61.55.

DCA Part 2, Tier 2, 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 from tubular ultrafiltration and reverse osmosis

DCA Part 2, Tier 2, 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.

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. 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.

DCA Part 2, Tier 2, 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. Spent carbon is normally sent directly to waste containers to avoid mixing with resins.

For spent cartridge filters, the SRWS starts at the filter extraction point. With the help of a monorail hoist, the spent filter is removed from the filter housing and placed in a shielded spent filter transfer cask. The cask is transferred to a cart and then taken to the RWB. The spent filter is placed in a waste container. Once the waste container is full, it is dewatered, sealed, and stored for eventual offsite processing and disposal.

DCA Part 2, Tier 2, 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: PST, phase separator transfer pump, SRST, SRST transfer pump, high-integrity containers, and the break pot tank.

The COL licensee will conduct SRWS preoperational inspections and testing to ensure that all subsystems are ready to operate and meet their design bases and performance specifications and that all automatic control functions are fully operational. 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 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, in accordance with 10 CFR 61.55 and 10 CFR 61.56, according to the site process control program (PCP). DCA Part 2, Tier 2, Tables 11.4-2 and 11.4-3, present expected waste classifications of solid wastes for influents processed and shipped.

In DCA Part 2, Tier 2, Section 11.4.3, “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 DCA Part 2, Tier 2, Section 11.2) and the plant ventilation system (described in

DCA Part 2, Tier 2, Section 9.4). 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 DCA Part 2, Tier 2, Sections 11.2 and 11.3, for the LRWS and GRWS, respectively. Sections 11.2 and 11.3 of this SER present the NRC staff's evaluation of liquid and gaseous effluent releases and doses, respectively.

ITAAC: There are no ITAAC items for the SRWS area of review.

Technical Specifications: The following TS are associated with the SRWS and are found in DCA Part 4: 3.4.8, 5.5.1, 5.5.2, and 5.5.6.

The TS also includes the following reports: 5.6.1 and 5.6.2.

Technical Reports: There are no technical reports associated with this area of the review.

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

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 20.1302 and 10 CFR 20.1301(e), as they relate to radioactive materials released in gaseous and liquid effluents to 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

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, “Design Guidance for Solid Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants,” Revision 4, issued January 2016

- SRP Section 11.4, Appendix 11.4-A, 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, 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 (RETS) (included in NUREG-1301)

- NUREG-1301, as it relates to the development of a plant-specific PCP or, alternatively, a COL applicant may use the NRC-approved NEI PCP Template 07-10A, Revision 0, “Generic FSAR Template Guidance for Process Control Program (PCP),” issued March 2009 (ADAMS Accession No. 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 DCA Part 2, Tier 2, Section 13.4, 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, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable”

- RG 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable”

- 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 DCA Part 2, Tier 2, Revision 3, Section 11.4, against the applicable NRC regulations and guidance in SRP Section 11.4 and in DSRS Section 11.4.

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 DCA Part 2, Tier 2, Tables 11.4-2 and 11.4-3. The NRC staff observed the applicant's commitments to following BTP 11-3 and ANSI/55.1-1992, "American National Standard for Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants." This guidance mentions a minimum onsite storage assumption of 30 days. The applicant included DCA Part 2, Tier 2, Table 11.4-3, to state the volumes of Class A and Class B/C waste generated per year. Because the applicant described the 30.5-meter (100-foot) elevation of the RWB as providing more than 92.9 square meters (1,000 square feet) of available space for storage, 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, by verifying the building floor space includes more than 92.9 square meters (1,000 square feet) of floor space.

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 Regulatory Position C.7 of RG 1.143, Revision 2, and RG 1.33, which specifies the QA guidance to follow. DCA Part 2, Tier 2, Table 3.2-1, also identifies the seismic category, quality group, and safety class for components of the SRWS. In determining the design guidance for radwaste systems, the applicant provided DCA Part 2, Tier 2, Table 11.4-3, to reflect the guidance specified in RG 1.143, Revision 2, to meet 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's confirmatory calculations were performed using the activities listed in DCA Part 2, Tier 2, Table 11.4-1. The NRC staff verified the listed RG 1.143 safety classification category for each 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 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 also by identifying the system boundaries. The applicant provided DCA Part 2, Tier 2, Table 11.4-1, to identify the planned waste generation rates along with the planned shipped waste rates. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Sections 11.4.2 and 12.3, “Radiation Protection Design Features,” indicates that the SRWS and features of the compound building 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 compound building and are monitored before being released to the environment through the RBVS plant stack monitor. Finally, the design includes radiation monitors installed on system components and in the compound building to monitor ambient radiation exposure rates and airborne radioactivity levels and alert operators of changing conditions and when to take corrective steps.

11.4.4.1.2 10 CFR Part 50, Appendix I

The radiological impacts noted in DCA Part 2, Tier 2, Section 11.4.3, associated with the operation of the SRWS are addressed by the NRC staff’s review of the LRWS and GRWS since the SRWS does not release liquid and gaseous effluents directly to the environment. Sections 11.2 and 11.3 of this SER present the NRC staff’s evaluation. 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 ECLs 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; the design objectives of 10 CFR Part 50, Appendix I; and 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 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 Sections 11.2 and 11.3 of this SER. The NRC staff finds this acceptable.

11.4.4.1.3 Site-Specific Cost-Benefit Analysis

DCA Part 2, Tier 2, Section 11.4.2.8, states that the SRWS does not release effluents to the environment, and 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 DCA Part 2, Tier 2, Section 11.4, against the criteria in 10 CFR 20.1406 for minimizing contamination. In DCA Part 2, Tier 2, Section 11.4.3, the applicant stated that the SRWS is designed in accordance with the requirements of 10 CFR 20.1406. The design features describe the ways in which the SRWS was designed to limit leakage and control the spread of contamination, facilitate decommissioning, and reduce the generation of radioactive waste, as discussed in FSAR Section 12.3.

11.4.4.1.5 Process Control Program

A COL applicant is to provide the PCP under COL Item 11.4-2. 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 PCP contains the planned effluent discharge flow rates and addresses the numerical guidelines in 10 CFR Part 50, Appendix I. The PCP will be developed to meet the requirements in 10 CFR Part 71 and will conform to the guidance in 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; and RGs 1.109, 1.111, and 1.113.

The NRC staff finds this response acceptable since DCA Part 2, Tier 2 includes COL Item 11.4-2, which states, “The COL applicant will develop a site-specific process control program following the guidance of NEI 07-10A.”

11.4.4.1.6 Mobile Equipment

DCA Part 2, Tier 2, Section 11.4.2.9, states the following:

> The SRWS is designed with modular equipment (spent resin dewatering system) and options for additional mobile equipment (shredders, laundry unit, etc.). The purpose of modular and mobile equipment is to provide ease of equipment replacement due to either advances in treatment technologies or equipment problems.

The applicant provided COL Item 11.4-1 to address mobile equipment in this design. This COL item states the following:

> A COL applicant that references the NuScale Power Plant design certification will describe mobile equipment used and connected to plant systems in accordance with ANSI/ANS-40.37, Regulatory Guide 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10, and 10 CFR 50.34a.

The applicant specified that a COL applicant is to provide reasonable assurance that provisions meet ANSI/ANS-40.37 and also conform to the regulatory requirements of 10 CFR 50.34a, 10 CFR 20.1406, and RG 1.143, Revision 2 (COL Item 11.4-1).

11.4.4.1.7 Technical Specifications

In DCA Part 2, Tier 2, Chapter 16, the applicant described the TS associated with the RWMSs. DCA Part 2, Tier 2, Chapter 16, 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. DCA Part 2, Tier 2, 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 under COL Item 11.4-2 (evaluated previously).

11.4.5 Combined License Items

Table 11.4-1 of this SER lists COL item numbers and descriptions related to the SRWS, from DCA Part 2, Tier 2, Table 11.4.

Table 11.4-1 NuScale COL Items for DCA Part 2, Tier 2, Section 11.4

COL Item No.	COL Item Description	DCA Part 2, Tier 2 Section No.
11.4-1	A COL Applicant that references the NuScale Power Plant design certification will describe any mobile equipment used and connected to plant systems in accordance with ANSI/ANS 40.37, Regulatory Guide 1.143, 10 CFR 20.1406, IE Bulletin 80-10, and 10 CFR 50.34a.	11.4.2.9
11.4-2	A COL Applicant that references the NuScale Power Plant design certification will develop a site-specific Process Control Program following the guidance of NEI 07-10A (Reference 11.4-3).	11.4.3

The NRC staff finds the above list to be complete. Also, the list adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items that need to be included in DCA Part 2, Tier 2, Table 1.8-2, for the SRWS.

11.4.6 Conclusion

The NRC staff concludes that the SRWS, as a modular system with options for additional mobile equipment, 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; 10 CFR 52.47; 40 CFR Part 190; 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 safe storage of 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 GDC 61 by meeting the guidelines of RG 1.143, Revision 2, in providing sufficient wet and solid waste processing capacities and storage space to ensure adequate safety under normal operations, AOOs, and postulated accident conditions.

- 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, exclusive of dry wastes identified as Class A wastes under 10 CFR 61.55. The approach to low-level radioactive waste (LLRW) management presumes that LLRW will be disposed of by shipment to an authorized recipient under 10 CFR 20.2001(a)(1). Under that approach, the COL applicant should demonstrate the capability of the means included in the design to process DSW and WSW so that these wastes meet the classification and characterization definitions in 10 CFR 61.55 and 10 CFR 61.56, respectively.

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

- Under COL Item 11.4-1, as described in DCA Part 2, Tier 2, Table 1.8-2, a COL applicant referencing the NuScale certified design will demonstrate compliance with 10 CFR 20.1406, 10 CFR 50.34a, and NRC IE Bulletin No. 80-10 and conform with industry guidance in ANSI/ANS-40.37-2009 for mobile equipment used and connected to plant systems.

- The SRWS does not contribute to offsite effluent doses, as assessed by the requirements 10 CFR 20.1301, 10 CFR 20.1302, 10 CFR Part 20, Appendix B, and 10 CFR Part 50, Appendix I.

- The NuScale design implements a plant-specific PCP as an operational program, described in DCA Part 2, Tier 2, Sections 11.4.3 and 13.4, for the processing of LLRW. 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 11.4-2, as described in DCA Part 2, Tier 2, Table 1.8-2.

11.5 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

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 postaccident 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 and to collect samples on filtration and in adsorbent media. The PERMISS establishes alarm setpoints to indicate when excessive radioactivity levels are present; tracks and records rates of radioactivity releases; and initiates protective isolation actions, such as terminating or diverting process or effluent flows. A 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 to those located in control rooms.

11.5.2 Summary of Application

DCA Part 2, Tier 1: The Tier 1 information associated with this section is found in DCA Part 2, Tier 1, Section 2.7, "Radiation Monitoring—Module Specific"; Section 3.9, "Radiation Monitoring—NuScale Power Modules 1–12"; Section 3.17, "Radiation Monitoring Power Modules 1–6"; and Section 3.18, "Radiation Monitoring Power Modules 7–12." Each of these sections discusses the nonsafety-related automatic actions for various radiation monitors for module-specific and shared radiation monitors.

DCA Part 2, Tier 2: The applicant described the system in DCA Part 2, Tier 2, Section 11.5, summarized as follows.

In DCA Part 2, Tier 2, Section 11.5, 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 PERMISS is used to extract and collect liquid and gaseous samples from various process and effluent streams for analyses conducted in laboratory settings. The system consists of skid-mounted radiation monitoring equipment and permanently installed sampling lines, with the equipment located at points to measure radioactivity or collect samples that are 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 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. PERMISS subsystems and components are found throughout the plant as design requirements of plant systems.

In DCA Part 2, Tier 2, Section 11.5.1, "Design Bases," the applicant presented the design basis and criteria of the system. DCA Part 2, Tier 2, Section 11.5.2, describes the system. DCA Part 2, Tier 2, Tables 11.5-1 and 11.5-2, describe system features used for sampling and monitoring, as well as operational characteristics of radiation monitors, and identify subsystems that include automatic control functions in terminating or diverting process flows and effluent releases. DCA Part 2, Tier 2, Table 11.5-4, describes the effluent and process radiation monitoring system dynamic ranges. DCA Part 2, Tier 2, Figures 11.5-1 - 11.5-6, depict the monitor locations and descriptions. Except for specific subsystems, failure of the PERMISS does not compromise safety-related systems or components and does not prevent the safe shutdown of the plant.

DCA Part 2, Tier 2, Section 11.5.2, identifies the means to collect process and effluent samples for radiological analyses. The instrumentation and controls of the LRWS, GRWS, and SRWS, as they relate to the interface and operation of the PERMISS, are not required for safety.

In DCA Part 2, Tier 2, Section 9.4.2, "Reactor Building and Spent Fuel Pool Area Ventilation System"; Section 10.4.2, "Condenser Air Removal System"; Section 10.4.3, "Turbine Gland Sealing System"; Section 9.1.3.2, "Pool Surge Control System"; Section 11.2, "Liquid Waste Management System"; and Section 9.2.9, "Utility Water System," the applicant described the design bases, operation, and monitoring of such liquid radioactive waste and building ventilation systems.

The applicant stated that 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 SRP Section 11.5, “Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems,” and associated regulatory guidance. For requirements related to Three Mile Island (TMI), the applicant stated that the design conforms to 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii) in monitoring gaseous effluents from potential accident release points using regulatory guidance.

ITAAC: The ITAAC associated with this section are in DCA Part 2, Tier 1, Section 2.7, “Radiation Monitoring”; Section 3.0, “Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria”; Section 3.9, “Radiation Monitoring—NuScale Power Modules 1–12”; Section 3.17, “Radiation Monitoring Power Modules 1–6”; and Section 3.18, “Radiation Monitoring Power Modules 7–12.”

Technical Specifications: The following TS associated with the PERMISS are in DCA Part 4: 5.5.1, 5.5.2, 5.5.4, and 5.5.6.

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

Topical Reports: There are no topical reports 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 review interfaces with other SRP sections appear in SRP Section 11.5. The following summarizes the regulatory requirements:

- 10 CFR 20.1302 and 10 CFR 20.1301(e), as they relate to monitoring radioactivity in plant radiological effluents to unrestricted areas, noting that these criteria apply to all effluent releases resulting from operation during normal plant operations and AOOs

- 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, pursuant to 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 effluent releases from the LRWS, GRWS, and SRWS and designing these systems to handle radioactive materials produced during normal plant operation, including AOOs

- GDC 63 and GDC 64, as they relate to the designs of the LRWS, GRWS, and SRWS and 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 a DCA 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

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

- RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” 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 4, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” issued June 2006, as it relates to accident monitoring instrumentation and performance of radiation monitoring systems, as well as additional guidance on the application of RG 1.97 provided in SRP Chapter 7, “Instrumentation and Controls,” BTP 7-10, “Guidance on Application of Regulatory Guide 1.97,” Revision 5, issued March 2007, on postaccident monitoring variables.

- 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, as it relates to the design, implementation, and QA of effluent monitoring and sampling systems

- RG 4.21, as it relates to minimizing the contamination of equipment, plant facilities, and the environment and minimizing 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, 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,” 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, 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 RETS (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 (ADAMS Accession No. 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 DCA Part 2, Tier 2, Section 13.4

- EPRI TR-104788-R2, “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 DCA Part 2, Tier 2, Revision 3, Sections 11.5 and 11.6, against the applicable NRC regulations and guidance in SRP Section 11.5 and in DSRS Sections 11.5 and 11.6.

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 monitoring instrumentation required for plant safety and protection
- radiation instrumentation required for monitoring plant operations and safe shutdown
- radiation monitoring of liquid and gaseous effluent releases

In DCA Part 2, Tier 2, Section 11.5, the applicant described the design basis of the process and effluent radiological monitoring system 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. 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. The PERMISS includes beta and gamma radiation-sensitive detectors working in redundant channels, as provided by the design. The radiation detectors can detect the types and energies of radiation emitted from fuel, radioactive wastes, and process and effluent streams. 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. The PERMISS generates signals to initiate the operation of certain equipment to control radioactive releases under AOOs and accident conditions. The COL licensee will test the PERMISS before operations and is responsible for calibrating all skid-mounted PERMISS subsystems installed in the plant. The PERMISS provides for periodic inspection of components to ensure the operational readiness and integrity of all PERMISS subsystems.

The discussion of the PERMISS is divided into two parts—effluent monitoring, described in DCA Part 2, Tier 2, Section 11.5.2.1, and process radiation monitoring, described in DCA Part 2, Tier 2, Section 11.5.2.2. The NRC staff has reviewed the information in these sections and corresponding tables and verified that the listed monitors are provided with locations, function descriptions, and anticipated ranges during operation.

The NRC staff review of the radiation monitors discussed in DCA Part 2, Tier 2, Section 11.5.2.1, 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.) 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." In its review of DCA Part 2, Tier 2, Section 11.5, the NRC staff uses this standard to verify those Type C and Type E variables discussed in the design and to ensure that monitoring exists to determine the magnitude of releases of radioactive materials. For this design review, there are no Type C variables discussed for radiation monitoring. In addition to the review of DCA Part 2, Tier 2, Section 11.5, the staff also reviewed information contained in DCA Part 2, Tier 2, Section 7.1.1.2.2, for the variable definitions to confirm that NuScale had appropriately referenced IEEE Std. 497-2002. The staff's review confirmed that the information contained in DCA Part 2, Tier 2, Table 7.1-7, Figure 7.1-2, Table 12.3-10, and Table 12.3-11, is consistent. In evaluating the information provided in the sections and tables described, the staff finds that the applicant has provided the necessary radiation monitors to meet the Type E variables requested in RG 1.97.

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, 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 loss of residual heat removal capability and 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 reactor building HVAC, CARS exhaust, turbine gland-sealing system exhaust, PSCS, 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 11.2.4.2 and 11.3.4.1.2). 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.

The applicant discussed the range of the Ar-41 CES radiation monitor and that Ar-41 is used to indicate leakage into the containment vessel since Ar-41 is one of the radionuclides available in the RCS. 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 radiation monitor range's low end by using the primary coolant activity concentration of Ar-41 and developed the corresponding secondary coolant activity concentration using the applicable parameters from DCA Part 2, Tier 2, 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 (μCi/cm^3)) and a primary-to-secondary leak rate of 1.6 kilograms per day, per unit (3.53 pounds per day, per unit).

The applicant has 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 μCi/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 CARS and through the main steam system (MSS) line radiation monitors. In DCA Part 2, Tier 2, Section 11.5.2, the applicant discussed the use and methodology of implementing argon injection for determining primary coolant leakage by following the guidance in EPRI Technical Document, "Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines—Revision 4," which is Reference 7.2.40 in NuScale TR-1116-52065-P, 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 μCi/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. In addition, COL Item 5.4-1 provides for the use of the argon injection method, if necessary, for leak rate determinations.

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 reactor building HVAC system, the RWB HVAC system, and the annex building HVAC system have the ability to provide this sampling and monitoring to alert plant personnel. The RWBVS is ultimately directed to the RBVS, where all release is monitored by the plant stack radiation monitors. In addition, the NuScale design monitors the control room conditions for airborne radiation. The described effluent and control room monitors follow the guidance of RG 1.97, and all have a designated Type E variable, as discussed in SER Section 11.5.4.1.1. As a result, the NRC staff finds these monitors acceptable for meeting the requirements in 10 CFR 50.34(f)(2)(xxvii) and 10 CFR 50.34(f)(2)(xxviii) because the applicant followed the guidance in RG 1.97.

11.5.4.1.4 Minimization of Contamination

Section 11.5 does not discuss minimization of contamination. This topic is discussed in Section 12.3 of this SER.

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

The staff's review of ITAAC related to radioactive waste systems and radiation effluent monitoring is contained in SER Section 14.3.8.4.5. This review contains the staff's assessment on ITAAC in DCA Part 2, Tier 1, Sections 2.7, 3.9, 3.17, and 3.18.

11.5.4.1.6 Technical Specifications

In DCA 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 5 mSv (0.5 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 ECLs 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 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. A plant- and site-specific ODCM will address the implementation of such programs under COL Items 11.5-1, 11.5-2, and 11.5-3.

11.5.4.1.7 Offsite Dose Calculation Manual

DCA Part 2, Tier 2, Section 11.5.2.4, states 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. DCA Part 2, Tier 2, Section 11.5.2.6, discusses that a COL applicant is to provide an ODCM. The applicant noted 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 provided COL Item 11.5-2, which states that a COL applicant that references the NuScale Power DC will develop an ODCM that describes the methodology and parameters used for calculation of offsite doses for gaseous and liquid effluents, using the 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. Both DCA Part 2, Tier 2, Section 11.5.2.6 and Table 1.8-2, provide a COL item for the COL applicant to address the ODCM. Under COL Item 11.5-2, the COL applicant is required to prepare the ODCM following NEI 07-09A as an alternative to providing the ODCM at the time of application because the ODCM requires plant- and site-specific information, which is outside the scope of the requested DC. The NRC staff therefore finds the inclusion of COL Item 11.5-2 acceptable.

11.5.4.1.8 Radiological Environmental Monitoring Program

DCA Part 2, Tier 2, Section 11.5.2.7, discusses that a COL applicant is to provide a radiological environmental monitoring program (REMP). The applicant stated that the REMP is to follow the guidance of NUREG-1301 and NUREG-0133 and consider local land use census data for the identification of additional pathways for inclusion in the analysis of gaseous and liquid effluent doses. The applicant provided COL Item 11.5-3, which states that a COL applicant that references the NuScale Power Plant DC will develop a REMP consistent with the guidance in NUREG-1301 and NUREG-0133.

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 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 found the submittal acceptable. Both DCA Part 2, Tier 2, Section 11.5.5 and Table 1.8-2, provide a COL item for the COL applicant to develop the REMP. Under COL Item 11.5-3, the COL applicant will develop the REMP following the guidance in NUREG-1301 and NUREG-0133. Because the REMP requires plant- and site-specific information, which is outside the scope of the requested DC, the NRC staff finds the inclusion of COL Item 11.5-3 acceptable.

11.5.5 Combined License Items

Table 11.5-1 lists COL item numbers and descriptions related to the PERMISS, from DCA Part 2, Tier 2, Table 1.8-2.

Table 11.5-1 NuScale COL Items for DCA Part 2, Tier 2, Section 11.5

COL Item No.	COL Item Description	DCA Part 2, Tier 2 Section No.
11.5-1	A COL Applicant that references the NuScale Power Plant design certification will describe site specific process and effluent monitoring and sampling system components and address the guidance provided in ANSI N13.1-2011, ANSI N42.18-2004 and RGs 1.21, 1.33 and 4.15.	11.5.2
11.5-2	A COL Applicant that references the NuScale Power design certification will develop an offsite dose calculation manual (ODCM) that contains a description of the methodology and parameters used for calculation of offsite doses for gaseous and liquid effluents, using the guidance of NEI 07-09A (Reference 11.5-8).	11.5.2.6
11.5-3	A COL Applicant that references the NuScale Power design certification will develop a radiological environmental monitoring program (REMP), consistent with the guidance in NUREG-1301 and NUREG-0133 that considers local land use census data for the identification of potential radiation pathways radioactive materials present in liquid and gaseous effluents, and direct external radiation from systems, structures, and components.	11.5.2.7

The NRC staff finds the above list to be complete. Also, the list adequately describes actions necessary for the COL applicant. The NRC staff identified no additional COL items that need to be included in DCA Part 2, Tier 2, Table 1.8-2, 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 functions, isolate 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 DSRS Section 11.5 acceptance criteria. This conclusion is based on the following:

- The NuScale design demonstrates compliance with 10 CFR 50.34a and 10 CFR Part 50, Appendix A, GDC 13, 60, 63, and 64, by providing the means to monitor and control liquid and gaseous effluent releases. In the NuScale design, effluent and process radiation monitoring and sampling provisions are described on a per-system basis to include the RBVS, CARS, TGSS, PSCS, LRWS, UWS, CWS, CRVS, CRHS, RWBVS, ABVS, GRWS, CES, MSS, CFDS, PSS, CVCS, RCCWS, SCWS, ABS, BPDS, DWS, CPS, RWDS, and SRWS. The PERMISS design conforms to the guidelines of DSRS

Section 11.5. The PERMISS monitors liquid effluent releases through a sole discharge line and gaseous effluent releases to the environment via the plant exhaust stack.

- Operating in conjunction with the LRWS, GRWS, and SRWS, 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 ECLs specified in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2.

- In conjunction with the LRWS, GRWS, and SRWS, 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 complies with 10 CFR 50.34a and 10 CFR 50.36a. Sections 11.2, 11.3, and 11.4 of this SER 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.

- Following the guidance in RG 1.143, Revision 2, the NuScale design conforms to the quality group classifications used for system components and the seismic design applied to structures that house PERMISS subsystems.

- 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 describe the site-specific process and effluent monitoring and sampling system components and address the guidance in ANSI/HPS N13.1-2011, ANSI N42.18-2004, and RGs 1.21, 1.33, and 4.15 under COL Item 11.5-1, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale certified design will develop an ODCM that contains a description of the methodology and parameters used for calculation of 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 compliance with 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 under COL Item 11.5-2, as described in DCA Part 2, Tier 2, Table 1.8-2.

- A COL applicant referencing the NuScale DC will develop a REMP using the guidance in NUREG-1301 and NUREG-0133 that considers local land use census data to identify potential exposure pathways from liquid and gaseous effluents and direct external radiation from SSCs. COL Item 11.5-3, as described in DCA Part 2, Tier 2, Table 1.8-2, specifies the implementation of a REMP.

11.6 Guidance on 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 Items

There are no COL Items for Section 11.6.

11.6.5 Conclusion

SER Section 11.5.5 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.