12 RADIATION PROTECTION

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereafter referred to as the staff) review of Chapter 12, "Radiation Protection," of the NuScale Power, LLC (NuScale) (hereafter referred to as the applicant), Standard Design Approval Application (SDAA), Final Safety Analysis Report (FSAR). The staff's regulatory findings documented in this report are based on Revision 2 of the SDAA, dated April 9, 2025 (Agencywide Documents Access and Management System Accession No. ML25099A237). In the SDAA, the applicant uses the English system of measure to provide the precise parameter values reviewed by the staff in this SER. Where appropriate, the NRC staff converted these values for presentation in this SER to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate.

FSAR Chapter 12 addresses radiation protection policy considerations, design considerations, and operational considerations applied during the design process. Where appropriate, combined license (COL) action items are described. Radiation sources, including the quantities of contained solid and liquid radioactive material and airborne radioactive material, and the associated bases are described. This chapter of the application describes the radiation protection design features provided to protect members of the public, the workers, and the environment, including facility design features; radiation shielding material and quantities; ventilation components and flow rates; area radiation monitoring and airborne radiation monitoring equipment; and features of the design that minimize contamination of the environment, minimize the generation of waste, and facilitate decommissioning. The application describes the methods used and the resultant dose estimates expected for activities anticipated during normal operation, including during refueling, following accidents, and during construction.

12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable

12.1.1 Introduction

As low as is reasonably achievable (ALARA) means making every reasonable effort to maintain exposures to radiation as far as practicable below the dose limits of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation." This includes, in part, accounting for the state of technology and the economics of improvements in relation to benefits to public health and safety. It also includes using procedures and engineering controls based on sound radiation protection principles.

The ALARA principles are used at the design stage to identify and describe the sources of radiation exposure expected to be generated during plant operation, anticipated operational occurrences (AOOs), maintenance and inspection activities, and accidents.

The ALARA principles are applied during the design process to identify and describe design features and specifications intended to limit and minimize the amount of radiation exposure to members of the public from contained radiation sources within the plant during operation; radiation exposure from operating modules to workers constructing or installing additional modules; and radiation exposure to occupational workers during plant operation, AOOs, maintenance and inspection activities, and accidents. Operational program elements are used to complement design features and specifications to limit and minimize radiation exposure.

12.1.2 Summary of Application

FSAR Section 12.1, "Ensuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable," can be summarized as follows:

- Most nuclear plant worker occupational radiation exposure results from the operation and maintenance of systems that contain radioactive material, radioactive waste handling, inservice inspection, refueling, abnormal operations, and decommissioning work activities. The application discusses how the design of the NuScale US460 small modular reactor (SMR) minimizes radiation exposure from these activities through the physical layout of the plant, selection of materials, radiation shielding, chemistry controls to minimize corrosion products, and other design features.

- The design of the facility is important to ensuring that occupational doses and doses to the public remain ALARA. During the design process, ALARA design reviews are periodically conducted. To the extent that operating experience is relevant to the NuScale SMR design, the design is based on experience and lessons learned from operating reactors.

- Examples of facility features in the NuScale SMR design that ensure that the design is ALARA include the separation of radioactive components into individual shielded compartments; the use of remote operating equipment, where possible, to reduce radiation exposure; and the minimization of embedded or underground piping to the extent practicable. SER Section 12.3 discusses in more detail the design features that ensure that exposures to occupational workers and members of the public are ALARA and are within applicable dose limits.

- The COL applicant will provide operational aspects of the radiation protection program to provide reasonable assurance that occupational radiation exposures are ALARA, as discussed later in this section.

Technical Specifications: There are no technical specifications (TS) for this area of review.

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

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

12.1.3 Regulatory Basis

The relevant requirements of the Commission's regulations for ensuring that occupational radiation exposure is ALARA, 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" (the SRP), appear in SRP Section 12.1, "Assuring that Occupational Radiation Exposures Are as Low as Is Reasonably Achievable," Revision 4, issued September 2013 (ML13151A061). The following summarizes the regulatory requirements:

- 10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," as it relates to keeping workers who receive occupational radiation exposure informed as to the storage, transfer, or use of radioactive materials or radiation in such areas and instructed as to the risk associated with occupational radiation exposure, precautions, and procedures to reduce exposures, and the purpose and function of the protective devices used.

- 10 CFR 52.137(a)(5) as it relates to the means to controlling and limiting radioactive effluents and radiation exposure within the limits of 10 CFR Part 20.

- 10 CFR 52.47(b)(1), which requires an FSAR to contain the inspections, tests, analyses and acceptance criteria (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 facility that incorporates the design certification (DC) has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act of 1954, as amended (AEA), and NRC regulations.

- 10 CFR 20.1101, "Radiation protection programs," and the definition of ALARA in 10 CFR 20.1003, "Definitions," as they relate to measures that ensure that radiation exposures resulting from licensed activities are below specified limits and ALARA.

- 10 CFR 20.1406, "Minimization of contamination," which requires that applicants for standard design approvals (SDAs) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," shall describe in the application how the facility design will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste.

The guidance in SRP Section 12.1 lists the acceptance criteria that are adequate to meet the above requirements and review interfaces with other SRP sections. SRP Section 12.1 also references the following:

- Regulatory Guide (RG) 1.8, "Qualification and Training of Personnel for Nuclear Power Plants"

- RG 1.33, "Quality Assurance Program Requirements (Operation)"

- 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 and Public Radiation Exposures as Low as Is Reasonably Achievable"

- RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants"

- NUREG-1736, "Consolidated Guidance: 10 CFR Part 20—Standards for Protection against Radiation," issued October 2001 (ML013330179)

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

- Nuclear Energy Institute (NEI) 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program Description" and the associated NRC SER (ML091490684), dated May 2009

- NEI 07-08A, “Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA),” and the associated NRC SER (ML093220178), dated October 2009

- NEI 08-08A, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination,” and the associated NRC SER, issued October 2009 (ML093220530)

- SECY-04-0032, “Programmatic Information Needed for Approval of a Combined License without Inspections, Tests, Analyses, and Acceptance Criteria,” dated February 26, 2004 (ML040230079) and its associated staff requirements memorandum, dated May 14, 2004 (ML041350440)

12.1.4 Technical Evaluation

The NRC staff reviewed the information in FSAR Section 12.1, in accordance with the review procedures in SRP Section 12.1. The results of the NRC staff’s review are provided below.

<u>Policy Considerations</u>

In FSAR Section 12.1.1, “Policy Considerations,” the applicant described the design, construction, and operational policies that have been implemented to ensure the ALARA considerations of 10 CFR 20.1101(b). The applicant has committed to ensuring that the NuScale SMR plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8 and the requirements of 10 CFR Part 20. FSAR Section 12.1.2, “Design Considerations,” states that the applicant has met this commitment by training designers and engineers on the incorporation of ALARA into the design evolution process. This training included communicating lessons learned from the nuclear power industry, as applicable to the NuScale SMR design.

The requirements of 10 CFR Part 20 specify that all licensees must develop, document, and implement a radiation protection program that encompasses the ALARA concept and includes provisions for maintaining radiation doses and intakes of radioactive materials ALARA for both occupational workers and members of the public. The detailed policy considerations for overall plant operations and implementation of such a radiation protection program are outside the scope of the SDA review. Compliance with 10 CFR Part 19 requires, in part, that workers who receive an occupational exposure be kept informed of and receive instructions with the objective of minimizing exposures to radiation and radioactive materials. COL Items 12.1-1 and 12.5-1 direct the COL applicant to describe the operational radiation protection and ALARA program, which include elements necessary to demonstrate compliance with 10 CFR Parts 19 and 20.

<u>Design Considerations</u>

The applicant used an interdisciplinary team of experienced engineers to identify and evaluate existing operating plant experience for evaluating the guidance in RG 8.8 with respect to meeting the requirements, including ALARA, of 10 CFR Part 20. The application identifies the types of design and operating considerations subsequently used to inform the design and specifications presented.

<u>Operational Considerations</u>

The application states that ALARA was implemented as part of the design process. Evidence of the implementation of these principles includes the use of an interdisciplinary team of experienced engineers to identify and evaluate existing operating plant experience for evaluating the guidance in RG 8.8 with respect to meeting the requirements, including ALARA, of 10 CFR Part 20. Operational considerations for the implementation of a radiation protection program are outside the scope of the SDAA review. NuScale states that a COL applicant that references the NuScale US460 SMR design will address the operational program to maintain radiation exposures to radiation ALARA. The NRC staff does not review operational programs during the SDAA review phase; therefore, it is acceptable for COL applicants to address the operational considerations as described in the COL item applicable to this section.

Radiation Protection Considerations

The COL applicant will provide the operational radiation protection program, as discussed in SER Section 12.5.

12.1.5 Combined License Information Items

Table 12.1-1 lists the COL information item number and description related to radiation protection from FSAR Section 12.1.3, “Operational Considerations.”

Table 12.1-1 NuScale COL Information Items for FSAR Section 12.1

COL Item No.	Description	FSAR Section
12.1-1	An applicant that references the NuScale Power Plant US460 standard design will describe the operational program to maintain exposures to ionizing radiation as far below the dose limits as practical, as low as reasonably achievable (ALARA).	12.1.3

12.1.6 Conclusion

Based on the information supplied by the applicant as described above, the NRC staff concludes that the general NuScale SMR design features described in FSAR Section 12.1 meet the acceptance criteria of SRP Section 12.1 and the applicable requirements of 10 CFR Part 19, 10 CFR Part 20, 10 CFR 52.137(a)(5), and 10 CFR 52.47(b)(1).

12.2 Radiation Sources

12.2.1 Introduction

The determination of projected radiation sources during normal operations, AOOs, and accident conditions in the plant is used as the basis for designing the radiation protection program and for developing shield design calculations. This determination includes defining isotopic composition, identifying the location of sources of radiation in the plant, determining source strength, and determining source geometry. In addition, the airborne radioactive material sources in the plant are considered in the design of the ventilation systems and are used for the design of personnel protective measures and for dose assessment.

12.2.2 Summary of Application

The applicant described onsite radiation sources, primarily in FSAR Section 12.2, "Radiation Sources," which is summarized, in part, as follows:

- FSAR Section 12.2 discusses and identifies the sources of radiation that form the basis for the shielding design calculations, radiation zoning, and dose assessments. This section also describes sources of direct radiation exposure to members of the public. In addition, it describes the sources of airborne radioactivity used to design personnel protection measures. Finally, it provides information on post-accident radiation sources.

- During normal operation, inside containment and near containment, the radiation types of concern consist of neutrons and gamma radiation emitted by the reactor core; gamma radiation from fission, corrosion, and activation products in the reactor coolant; and gamma radiation from activated components. Elsewhere in the facility, the contained sources of radiation include radioactive material found in systems and components (such as demineralizers, filters, and tanks) that treat, process, or otherwise contain reactor coolant. The systems include the chemical and volume control system (CVCS); pool cooling and cleanup system; plant sampling systems; and solid, liquid, and gaseous waste management systems. These sources emit gamma radiation, which requires shielding consideration and assessment of the dose to occupational workers and members of the public.

- Airborne radioactivity material within the reactor building (RXB) consists of evaporation from the ultimate heat sink (UHS) pool and equipment leakage. Airborne radioactive material within the radioactive waste building (RWB) is principally the result of equipment leakage. The design of the ventilation systems in radiological portions of these buildings is used to minimize airborne radioactive material concentrations by providing airflow from regions that are expected to have a lower potential for airborne contaminates to those with a higher potential for airborne contaminates.

- In addition, FSAR Section 12.2 provides information on post-accident source terms in the NuScale design. FSAR Chapter 15, "Transient and Accident Analyses," provides additional information on the post-accident source terms, and the accident source term methodology appears in NuScale TR-0915-17565-NP-A, "Accident Source Term Methodology," Revision 4, dated February 26, 2020 (ML20057G132). The post-accident source terms are used to evaluate the doses in the main control room (MCR), potential post-accident doses to the public, and the doses to equipment important to safety (i.e., FSAR Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," discusses the equipment qualification (EQ) program and the associated analysis, and SER Section 3.11 discusses the NRC staff's evaluation).

- FSAR Chapter 11, "Radioactive Waste Management," also provides some source term information, including the maximum isotopic core inventory and realistic and design-basis coolant source terms. These initial core and primary coolant source terms are used in developing many of the other plant source terms.

ITAAC: There are no ITAAC associated with the review of FSAR Section 12.2.

Technical Specifications: The normal operation design-basis fission product source terms are based on a design-basis failed fuel fraction (DBFFF) of 0.066 percent, consistent with Limiting Condition for Operation 3.4.8.

Technical Reports:

- NuScale TR-123242, “Effluent Release (GALE Replacement) Methodology and Results,” Revision 1, dated August 2023 (ML23304A359)

- Electric Power Research Institute (EPRI) TR-3002000505, “Pressurized Water Reactor Primary Water Chemistry Guidelines,” Volumes 1 and 2, Revision 7, issued April 2014

Topical Reports:

- NuScale TR-0915-17565-NP-A, Revision 4 (The staff’s approval of TR-0915-17565 applies only to the NuScale SMR design. The NuScale SMR design is defined as the design described on Docket Number 52-048 and subsequent revisions to that design that continue to maintain the same fundamental size, geometry, and safety features of the design docketed in 52-048. The design described in Docket Number 52-048 is the US600. The US460 design that is under review is similar to the US600 design in certain respects and maintains the same fundamental size, geometry, and safety feature; hence the staff finds this topical report to be applicable to the US460.)

12.2.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for ensuring that occupational radiation exposure is ALARA, in accordance with dose limits, and other requirements that rely on or use the design-basis source terms information in FSAR Section 12.2 are described below. Section 12.2, “Radiation Sources,” of “Design Specific Review Standard for NuScale SMR Design,” issued June 2016 (ML15350A320), describes the associated acceptance criteria and the review interfaces with other SRP or design-specific review standard (DSRS) sections.

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

- 10 CFR 20.1101, 10 CFR 20.1201, “Occupational dose limits for adults,” 10 CFR 20.1202, “Compliance with requirements for summation of external and internal doses,” and 10 CFR 20.1206, “Planned special exposures,” as they relate to limiting occupational radiation doses

- 10 CFR 20.1203, “Determination of external dose from airborne radioactive material,” and 10 CFR 20.1204, “Determination of internal exposure,” as they relate to limiting average concentrations of airborne radioactive materials to protect individuals and control the intake (inhalation or absorption) of such materials

- 10 CFR 20.1207, “Occupational dose limits for minors,” as it relates to limiting exposure to minors to one-tenth of the annual limits for adults

- 10 CFR 20.1208, “Dose equivalent to an embryo/fetus,” as it relates to limiting exposure to declared pregnant workers

- 10 CFR 20.1301, “Dose limits for individual members of the public,” and 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” as they relate to the determination of radiation levels and radioactive material concentrations within the components of the plant that could affect direct radiation exposure to members of the public

- 10 CFR 20.1406 and 10 CFR 52.137(a)(6), as they relate to the identification of systems that contain radioactive material for which the applicant should describe how the design minimizes contamination of the facility and environment, minimizes the generation of waste, and facilitates decommissioning

- 10 CFR 20.1801, “Security of stored material,” as it relates to securing licensed materials against unauthorized removal

- 10 CFR 50.34(f)(2)(vii), which requires radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive material, and design as necessary to permit adequate access and to protect safety equipment from the radiation environment

- 10 CFR 50.49(e)(4), which requires the determination of the radiation environment expected during normal operation and the most severe design-basis accidents (DBAs) and requires electric equipment that can be relied on to remain functional during and following design-basis events (DBEs), including AOOs

- General Design Criterion (GDC) 4, “Environmental and dynamic effects design bases,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” which requires systems, structures, and components (SSCs) important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents

- GDC 19, “Control room,” as it relates to the acceptable radiation conditions in the plant under accident conditions and the source term release assumptions used to calculate those conditions

- GDC 61, “Fuel storage and handling and radioactivity control,” as it relates to systems that may contain radioactive materials

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

- 10 CFR 52.137(a)(22), as it relates to ensuring that the application includes information necessary to demonstrate how the plant design incorporates operating experience insights

The guidance in DSRS Section 12.2 lists the acceptance criteria that are adequate to meet the above requirements and review interfaces with other SRP sections, and it references the following:

- RG 1.7, “Control of Combustible Gas Concentrations in Containment,” as it relates to radionuclides in systems used for determining gaseous concentrations in containment following an accident

- RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants”; RG 1.29, “Seismic Design Classification for Nuclear Power Plants”; and RG 1.117, “Protection against Extreme Wind Events and Missiles for Nuclear Power Plants,” as they relate to the radiological criteria for classification and protection of nonradioactive waste SSCs that contain radioactive material

- RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” and RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” as they relate to the determination of radiation dose to certain electrical equipment important to safety as described in 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants”

- RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” as it relates to complying with NRC regulations under 10 CFR 20.1301 concerning the calculation of realistic radiation levels and radioactive material source terms for the evaluation of waste treatment systems

- RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” as it relates to design features that are provided to minimize occupational radiation exposure and the classification of structures that house radioactive waste systems based on potential exposure to site personnel

- RG 1.183, as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following an accident

- NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980, Task Action Plan Item II.B.2, using NuScale-specific source term values, as it relates to the identification of specific post-accident sources of radiation in the facility

The following document also provides additional criteria or guidance in support of the SRP acceptance criteria to meet the above requirements:

- American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard (Std.) 18.1-1999, “Source Term Specification,” as it relates to methods and data used to estimate typical long-term concentrations of principal radionuclides in fluid streams of light-water-cooled nuclear power plants

DSRS Section 12.2, “Radiation Sources,” specifies that the applicant modify the methods and data in ANSI/ANS Std. 18.1-1999 to reflect NuScale-specific design attributes. The methods and data in ANSI/ANS Std. 18.1-1999 were developed using relevant industry operating experience.

12.2.4 Technical Evaluation

The NRC staff evaluated the information in FSAR Section 12.2 against the applicable regulations and the guidance in DSRS Section 12.2, to verify that the FSAR accurately described contained sources, including byproduct, source, and special nuclear materials, and other radiologically significant sources. The staff reviewed the radiologically significant radiation sources described in the application that were expected to be generated during normal operations, during potential AOOs, and as a result of potential accidents. The staff reviewed the methods, models, and assumptions used as the basis for establishing the kinds and quantities of radioactive materials or a radiation environment presented in the application. The specific areas of review include contained sources and airborne radioactive sources. The staff used the kinds and quantities of radioactive materials or the radiation environment present, as described in FSAR Section 12.2, to evaluate the SSCs and design features described in FSAR Section 3.11 and Section 12.3, "Radiation Protection Design Features," and other FSAR sections that describe the protection of equipment, workers, and members of the public from the effects of radiation. This section of the SER discusses many of the radiation sources and significant staff findings related to the radiation sources.

Radiation sources and storage areas described in the SDAA are located within the restricted area, which addresses the requirement of 10 CFR 20.1801. A COL applicant's radiation protection program must ensure that the requirements of 10 CFR 20.1801 and 10 CFR 20.1802 are met for any sources in controlled or unrestricted area (See COL Item 12.5-1 and FSAR Chapter 13 for radiation protection program and procedure requirements).

12.2.4.1 Contained Sources

In FSAR Section 12.2.1, "Contained Sources," the applicant described the source terms used for determining the radiation shielding, facility design features, and radiation zoning during normal full-power operation, including AOOs, and evolutions such as refueling, as well as sources of radiation exposure to equipment following potential accidents. FSAR Section 12.2.1 describes the radiologically significant contained sources of radiation that are used as the basis for designing the radiation protection program and shield design calculations. For each of these contained sources, the applicant provided either the source strength by energy group, the associated activity levels listed by isotope, or both. Most of these radiological source terms during normal operation are found in FSAR Section 12.2, based on the initial core and reactor coolant system (RCS) source term activities found in FSAR Section 11.1, "Source Terms."

The NRC staff reviewed the methods, models, and assumptions used by the applicant to determine the radionuclide concentrations in the RCS, the connected systems, and the downstream SSCs. The staff used a combination of the information in FSAR Section 12.2 and other FSAR sections; calculations performed by the NRC staff; and audit material to review the methods, models, and assumptions used by the applicant to derive the source terms given in FSAR Section 12.2. The NuScale US460 facility design includes simultaneous operation of six NuScale Power Modules (NPMs). SSCs, such as the spent resin storage tanks (SRSTs) and the phase separator tanks (PSTs), are designed to receive radioactive material from multiple modules. In addition, to allow for greater operational flexibility, some SSCs, like the SRSTs and PSTs, are relatively larger than the corresponding SSCs in the current operating fleet.

Using a risk-informed approach, the NRC staff based its review of the radioactive material content of only a single NPM operating at the TS coolant specific activity limit, with the other five units operating with realistic coolant activity concentrations. As a result, the methods the NRC staff used to evaluate the radioactive material content of these shared SSCs was adjusted to account for the dilution from other waste streams and radiological decay of the contents, as the

SSCs were filled. The NRC staff determined that the applicant used the appropriate methods, models, and assumptions to identify the kinds and quantities of radioactive material in contained sources resulting from operation of one NPM at the TS coolant specific activity limit for one operating cycle, consistent with the guidance in DSRS Section 12.2.

12.2.4.2 Sources Inside of Reactor Containment

The applicant used industry standard application packages, such as Oak Ridge Isotope Generation (ORIGEN), to develop the core source terms. Applicant-specific analytical calculations were used to distribute postulated leakage of the core source terms into the reactor coolant and through the connected plant systems. The applicant relied on industry guidance documents to develop applicant-specific analytical packages to describe the quantities and distributions of source terms for corrosion and activation products.

During the NuScale US600 design review, the staff reviewed information during an audit (see ML19203A043) and performed calculations using ORIGEN, Standardized Computer Analyses Licensing Evaluation (SCALE), and Monte Carlo N-Particle Transport Code (MCNP) and evaluated the neutron dose and the subsequent contribution of photons from the neutron activation of the containment vessel and reactor pressure vessel (RPV). Based on this review, the staff was able to determine that the gammas due to neutron activation of structural materials were only a minor contributor to the total integrated dose (TID) to the equipment at the top of the RPV and containment vessel and under the bioshield. Based on this confirmatory analysis, the NRC staff determined that the neutron and gamma radiation environments in the upper RPV, the upper containment vessel, and under the bioshield have been adequately described by the applicant and accounted for in FSAR Appendix 3C, “Methodology for Environmental Qualification of Electrical and Mechanical Equipment.” The staff compared the similarities between NuScale US460 and US600 design, including the similarities in the reactor and core design. The staff reviewed the neutron and gamma doses in the US460 design and with consideration of changes in reactor power level and core operating cycle length, the neutron and gamma doses in and around the containment vessel and RPV were similar to doses in the US600 design, as expected. The staff also audited the applicant’s calculation packages related to core and reactor coolant source terms in the US460 design. Based on the review, the staff found the neutron and gamma doses in and around the containment vessel and RPV to be acceptable in the US460 design.

The applicant provided source terms and assumptions for the in-core instrumentation (found in 12 fuel assemblies distributed in the reactor core) and control rod assembly tips in FSAR Tables 12.2-22, “In-Core Instrument Source Term Input Assumptions,” through 12.2-25, “Control Rod Assembly Tip Gamma Spectra (End of Cycle 1).” The applicant stated that the source terms for these components peak early; therefore, the source terms at the end of the first cycle are provided. For the in-core instrumentation, the source term is also provided after 40 cycles. This method of listing the radioactive content provides the maximum values expected to be present during use. Therefore, the NRC staff finds that the applicant adequately identified the kinds and quantities of radioactive material associated with the in-core neutron detectors, consistent with the requirements of 10 CFR 52.137(a)(5).

Reactor Coolant Source Terms

The RCS accumulates radionuclides from neutron activation of the coolant, from corrosion products in the coolant that become activated, and from fission products that leak into the reactor coolant from the reactor fuel. The reactor coolant is circulated through the reactor coolant loop and into other plant systems. The reactor coolant source term is the initial source term, which contributes to the radionuclide concentrations in most of the other plant systems that process and contain radioactivity.

The acceptance criteria in DSRS Section 12.2 state, in part, that the shielding and ventilation design fission product source terms will be acceptable if they are developed using the bases of 0.25 percent fuel cladding defects for pressurized-water reactors (PWRs) or the RCS isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to the operation for a full fuel cycle at the TS limits for halogens (iodine (I)-131 dose equivalent) and noble gases (xenon (Xe)-133 dose equivalent).

FSAR Table 11.1-2, "Parameters Used to Calculate Coolant Source Terms," specifies that NuScale used a DBFFF of 0.066 percent as the basis when calculating the radioactive material content in the RCS. The design basis reactor coolant source term is provided in FSAR Table 11.1-4, "Primary Coolant Design Basis Source Term." This is the starting source term to determine the radionuclide concentrations in most contained source inventories for SSCs, such as liquid tanks, demineralizers, filters, and charcoal beds, that are subsequently used as inputs for the normal operation radiation shielding design calculations, ventilation system design calculations, and normal operation personnel dose assessment. The NRC staff independently confirmed that the 0.066 percent failed fuel fraction for the DBFFF gives radionuclide concentrations that are consistent with the information in TS 3.4.8, "RCS Specific Activity," which provides RCS coolant concentration limits on dose-equivalent I-131 and dose-equivalent Xe-133. Since the RCS specific activity concentration limits of TS 3.4.8 correspond to the 0.066 percent failed fuel fraction value referenced by the applicant in FSAR Chapter 12, the NRC staff finds the use of this failed fuel fraction value to be acceptable in determining the design basis fission product inventory in plant systems and components.

As part of its review of the applicant's compliance with 10 CFR 52.137(a)(5), the NRC staff examined how the applicant addressed the generation, distribution, and collection of activated corrosion products. EPRI TR-3002000505, which is referenced in FSAR Section 12.2, states that the occurrence of crud-related phenomena has negatively impacted plant operations and core performance, such as anomalous crud releases and elevated radiation fields during refueling outages, crud-induced power shifts (formerly called axial offset anomaly), and crud-induced fuel failures. Crud source terms in the US460 design are calculated similarly to the crud source terms in the approved NuScale US600 certified design. The design basis crud source term in the reactor coolant system for normal operation is based on ANSI/ANS-18.1-1999 modified for NuScale specific design parameters, as discussed in TR-123242 for the US460 design. NuScale DSRS Section 12.2 states that the methodology in ANSI/ANS-18.1-1999 modified for NuScale-specific design attributes is an acceptable method for developing source terms. The crud source term in the reactor coolant is the basis for the crud source term in other plant systems. The staff also compared the crud source terms per individual reactor in the US460 design to those in the US600 design. The staff found that when adjusting the crud source terms for power level and other design parameters, the US460 design crud source terms were consistent with the US600 design crud source terms. Since the crud source terms in the US460 design are consistent with the source terms in the US600 design, when accounting for the changes in the design, and since the crud source terms are based on ANSI/ANS-18.1-1999, as modified for the NuScale US460 specific design parameters, the staff finds that the applicant has appropriately considered crud in the plant source terms.

The NRC staff reviewed information in FSAR Section 12.2 regarding the generation and transport of nitrogen (N)-16 through the RCS. Large quantities of N-16 are generated as reactor coolant passes through the neutron field present in the core during normal operation. The half-life of N-16 is about 7 seconds, so the transit time around the RCS flow loop has a significant impact on the amount of N-16 present at different points within the RCS. N-16 decays through the emission of a gamma photon with an energy of approximately 1.1 x 10^{-18} joules (7 megaelectronvolts). Because of their abundance and their energy, these photons can be a significant contributor to the TID of equipment and potentially to personnel if radioactive fluid with N-16 makes it outside of containment, to areas near personnel before significant radioactive decay has occurred.

FSAR Appendix 3C, Table 3C-6, “Normal Operating Environmental Conditions,” provides the 60-year integrated gamma dose from normal operation. FSAR Table 12.2-2, “Primary Coolant at the Steam Generator Entrance Gamma Source Term,” and Table 12.2-3, “Primary Coolant at the Core Exit Gamma Source Term,” provide the gamma spectra and fluence from the reactor coolant at two locations that are briefly described in FSAR Section 12.2.1.2, “Reactor Coolant System.” FSAR Table 12.2-4, “Nitrogen-16 Primary Coolant Concentrations at Full Power,” provides the applicant’s calculated N-16 concentration at several locations. The staff noted that the reactor coolant transit time identified in TR-123242 was significantly faster in the US460 design than it was in the previously approved US600 design.

Therefore, the staff audited information related to the N-16 source term calculations and the N-16 concentrations in the CVCS. Through the audit (ML23304A480) and as provided in FSAR Section 12.2.1.2, Table 12.2-7, “Chemical and Volume Control System Component Source Terms—Source Strengths,” and Table 12.2-11, “Liquid Radioactive Waste System Component Source Term Inputs and Assumptions,” the staff found that the applicant accounted for N-16 based on its decay time in the CVCS vertical pipe chase and in CVCS components until it is decayed to 10 half-lives. At 10 half-lives, N-16 contribution to dose is insignificant and it no longer needs to be considered. By the time the coolant reaches the liquid radioactive waste system (LRWS) degasifier vessel, N-16 has decayed 10 half-lives. FSAR Table 12.2-7 provides the photon source strengths at several locations; these source strengths include N-16, as appropriate. The column “CVCS Letdown—71.3 second decay” provides the concentrations in the letdown, after N-16 decays to the point it no longer needs to be considered. Based on the information provided by the applicant in the FSAR and information audited by the staff, the staff finds that N-16 is adequately accounted for in dose calculations.

Sources Outside of Containment

The NRC staff confirmed that the applicant provided source terms for the radiologically significant contained sources of radioactivity in FSAR Section 12.2. Outside of containment, the staff focused its review primarily on the most significant sources, which include, but are not limited to, the CVCS, pool cooling and cleanup system, LRWS, gaseous radioactive waste system, solid radioactive waste system, and spent fuel.

The NRC staff reviewed the geometries, concentrations, branching ratios, daughter product concentrations and emitted radiation, decay half-lives, radiation emission spectra, and attenuation coefficients. The staff verified that the applicant was accounting for the changes in radionuclide concentrations resulting from the decay and buildup of radionuclides (such as cesium (Cs)-137 and barium (Ba)-137m). The staff reviewed calculation packages and input and output files and performed confirmatory calculations to verify that the applicant either properly determined the concentration of radionuclides or, where it did not explicitly determine the radionuclide concentrations resulting from decay, that the applicant applied NRC-approved methods to compensate.

However, the NRC staff identified several areas in the FSAR where the Ba-137m concentrations were not commensurate with the stated Cs-137 concentrations:

- the "PCWS Surge Tank" column of Table 12.2-9, "Pool Cooling and Cleanup System Component Source Terms—Radionuclide Content"
- the columns "LCW Collection Tank (Ci)" and "HCW Collection Tank (Ci)" of Table 12.2-12a, "Liquid Radioactive Waste System Component Source Terms—Radionuclide Content"
- the column "LCW Reverse Osmosis Unit" in Table 12.2-12b, "Liquid Radioactive Waste System Component Source Terms—Radionuclide Content"
- the column "LCW Processing Skid" in Table 12.2-12c, "Liquid Radioactive Waste System Component Source Terms—Radionuclide Content"
- Table 12.2-15a, "Degasifier Radiological Content"
- the column "Phase Separator Tank" in Table 12.2-18, "Solid Radioactive Waste System Component Source Terms"

The principal gamma emission from the decay of Cs-137 is actually emitted from the Ba-137m daughter. Based on the 2.55-minute half-life of Ba-137m, the concentration of Ba-137m should be approximately 95 percent of the Cs-137 concentration. The staff identified a small change in dose rate calculations if Ba-137m was increased to be consistent with the Cs-137 concentration. While increasing the Ba-137m to be consistent with the Cs-137 may result in an increase in dose rate of a few percent, it would not result in significant changes to radiation zoning or to worker dose estimates.

In addition, the staff notes that during operation of the NuScale plant, a licensee's radiation protection program would have to ensure that doses throughout the plant are appropriately characterized and controlled. The radiation protection program can control minor differences in actual dose rates from calculations. Furthermore, because dose rates are calculated at the TS failed fuel fraction, the reactors would likely be operating with a lower failed fuel fraction and therefore a source term lower than the maximum source terms. Based on the low radiological significance of the total dose from the differences in reported radioactive concentration of Ba-137m, the NRC staff determined that the radiological assessment of the source terms in these SSCs is acceptable.

The NRC staff reviewed the geometric description of sources described in FSAR Section 12.2 to ensure that sources were appropriately distributed within the SSC that was expected to contain the source (e.g., in a volume the size and shape of a demineralizer within a room, versus the entire volume of the room containing the demineralizer bed). The staff also reviewed the assumed contents of the SSC versus the stated capacity of the SSC. The geometric arrangement of the sources may have a significant impact on the resultant dose rates outside of the SSC. Based on the review of the source term data, data on the capacities of some of the equipment and rooms based on information in other FSAR sections, and data of the sizes of some similar equipment at operating plants, the NRC staff determined that the applicant has appropriately considered the geometrical arrangement of SSCs containing radioactive material.

FSAR Table 12.2-11 provides decontamination factors (DFs) for the liquid radioactive waste granulated activated charcoal (GAC) unit. In the absence of regulatory guidance for the application of DFs for GAC, the applicant used DFs that were discussed in the paper "The Volume Reduction of Liquid Radioactive Waste by Combined Treatment Methods," referenced in the International Atomic Energy Agency (IAEA) document IAEA-TECDOC-1336, "Combined Methods for Liquid Radioactive Waste Treatment," dated February 2003. The applicant noted that the GAC filter is neither designed nor intended to collect radionuclides; however, because it could collect radionuclides, it was conservatively (for the purpose of the local radiation shielding assessment) assumed to collect radionuclides at the rate indicated by the DFs. In addition, the radionuclide concentration in the outlet stream from the GAC filter was assumed to not be reduced by the GAC filter. Therefore, the activity calculated as collected in the GAC filter is also available for collection on the downstream components. This ensures that shielding requirements and radiation zone designations of the downstream components are also conservatively established. The NRC staff concludes that the methodology, as described by the applicant, offers a reasonable way to estimate the local dose rates and shielding requirements, while also providing a reasonable process for estimating potential activity accumulation in downstream components.

The NRC staff reviewed the radionuclide concentrations listed in purification media, such as filters, resins, and charcoal beds, to ensure that the radioactive material content listed was based on the stated DFs and the flow rates through the components. As previously noted, some of the components are shared by as many as six NPMs, so the NRC staff adapted the review process to consider that only one NPM at a time would be operating at the TS coolant specification limit. The staff considered information provided by the applicant to assess the concentrations in the other waste streams that are expected to contribute to the radioactive material content of the purification media. The staff reviewed the DFs provided for purification media in the FSAR and the source terms provided for selected filtration media based on the DFs. The staff determined that the source terms and the methods, models, and assumptions used to determine that the radionuclide content of the purification media was appropriate.

With respect to the accumulation of radioactive material in demineralizers, the applicant did not consider the accumulation of crud-burst-related activity in FSAR Section 12.2. Consistent with this position, FSAR Table 12.2-6, "Chemical and Volume Control System Component Source Terms—Radionuclide Content," and Table 12.2-7 reflect only the radioactive material content of the SSCs resulting from normal operation and not the accumulation of radioactive material resulting from a crud burst.

While FSAR Table 12.2-6 and Table 12.2-7 do not contain crud-burst-related activity, FSAR Section 12.2.1.3, "Chemical and Volume Control System," does describe the potential accumulation of crud-related activity in demineralizers from crud burst cleanup. Therefore, the NRC staff concludes that the applicant has adequately identified the kinds and quantities of radioactive material expected to be produced during operation, including refueling outages, and collected in the CVCS demineralizers, consistent with the requirements of 10 CFR 52.137(a)(5). Further, the staff finds that the method used to establish crud burst factors employed in refueling dose calculations is consistent with the requirement of 10 CFR 52.137(a)(22) to demonstrate how operating experience insights have been incorporated into the plant design.

The NRC staff reviewed assumptions related to calculating the amount of radioactive material that could be present in resin transfer lines, as discussed in FSAR Section 12.2.1.3. The applicant assumed that the resin transfer line is modeled as 100 percent obstructed by resin beads from the CVCS mixed-bed demineralizer using a bulk dry resin density. The source term used for the spent resin transfer line is the CVCS mixed-bed source term decayed for 48 hours. However, as noted in the previous discussion of the crud burst content of resins, the source terms for the mixed-bed demineralizer are only representative of the source term accumulated in the listed components from normal operation without a crud burst. For resins that have collected radioactive material from a crud burst, the crud burst conditions outlined in FSAR Section 12.2.1.3 need to be considered by the facility operator when performing resin transfer operations. Since the radioactive material content of the resin may be higher than that assumed during the design of the shielding around the transfer lines, the staff determined that the radiation protection program may need to consider additional actions for the protection of equipment or personnel should the resin transfer occur before the radioactive material has decayed to values commensurate with those listed in FSAR Table 12.2-6 and Table 12.2-7. However, implementation and specification of items for inclusion in a radiation protection program are beyond the scope of review conducted for an SDA and will have to be addressed at the COL stage, in accordance with COL Items 12.1-1 and 12.5-1. Since the facility design does consider the activity in the CVCS mixed-bed demineralizer and since it is reasonable to assume that access to areas impacted by high doses during resin transfer can acceptably be controlled by a radiation protection program that addresses COL Items 12.1-1 and 12.5-1, the staff finds the design acceptable.

During the NRC staff review of the source term used as the basis for the high-integrity container (HIC) storage room, an area identified as containing a sufficient amount of radioactive material to result in dose rates of up to 5 grays per hour (Gy/h) (500 radiation absorbed doses per hour (rad/h)), the applicant noted in FSAR Section 12.2.1.7, “Solid Radioactive Waste System,” that the Class A/B/C HIC storage area in the RWB contains five HICs loaded with Class B/C dewatered resins from the SRST, which are decayed for approximately 2 years. The staff determined that the method used by the applicant to estimate the radiological significance of the radioactive material in the HIC storage room, while not considering the full storage volume of the room, was sufficient for the purpose of performing the required shielding analysis because HICs can be stored in a manner to limit radiation exposure to outside areas (for example, high activity HICs can be stored behind lower activity HICs to provide self-shielding) or other programmatic controls can be used to limit the dose to areas outside of the HIC storage area. Based on this, the staff determined that the description of the source term in the HIC storage room is consistent with the requirement of 10 CFR 52.137(a)(5) to identify the kinds and quantities of radioactive material sufficiently well to ensure that design features for controlling radiation exposure to within the limits of 10 CFR Part 20 can be implemented. Therefore, the staff finds this method to be acceptable.

The NRC staff reviewed the radioactive material content of the UHS pool provided in FSAR Section 12.2. Because of the depth of the UHS pool relative to those in the existing light-water reactor fleet, the staff determined that the radionuclide concentration in the UHS pool water is the dominant source of radiation exposure during refueling outages. FSAR Section 12.2.3, “References,” includes a reference to EPRI TR-3002000505. Volume 2 of this EPRI report covers startup and shutdown chemistry and states that deposition of particulates released during the shutdown evolution can lead to increased shutdown dose rates, elevated smearable activity levels in low-flow regions, and increases in personnel contamination risks. The document further notes that without operating reactor coolant pumps, the flow forces will be reduced, which could result in increased deposition of suspended material, less solubilization of system deposits, and an increased rate of deposition in low flow rate areas.

Using the information provided in the application and information audited by the staff (ML24211A089), the NRC staff was able to determine how the application factored aspects of the NuScale design, such as efficiencies and flow rates of cleanup systems, into the estimated amounts of radioactive material projected to be initially present in the RCS; the estimation of the effectiveness of the processes used to clean up the RCS; the amount of radioactive material that may be present inside NPM components at the time of disassembly; the subsequent amount of radioactive material added to the UHS pool water; and ultimately, the impact on radiological conditions (e.g., dose rates, airborne activity) in the area of refueling activities.

FSAR Section 12.2.1.8, “Reactor Pool Water,” specifies that the post crud burst cleanup of the primary coolant in the NPM by the CVCS will operate until the projected dose rate (after disassembly of the NPM) to an operator on the refueling bridge is less than 0.025 millisieverts per hour (mSv/h) (2.5 millirem per hour (mrem/h)). The proposed criterion of less than 0.025 mSv/h (2.5 mrem/h) is consistent with the criteria in ANSI/ANS Std. 57.2-1983, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” that the NRC staff uses. Thus, the licensee will have to operate the CVCS during each outage until the projected dose rate to workers on the bridge is less than 0.025 mSv/h (2.5 mrem/h), consistent with ANSI/ANS Std. 57.2-1983. In addition, the staff reviewed the methods, models, and assumptions that were used for determining the radiological conditions of the UHS pool water and subsequently can be used to determine the radiological conditions to an operator on the refueling bridge during the staff audit. The review by the NRC staff confirmed that the crud burst factors assumed by the applicant were consistent with industry operating experience and were therefore acceptable. Controlling the dose on the refueling bridge to less than 0.025 mSv/h (2.5 mrem/h) is consistent with ANSI/ANS 57.2-1983 and is acceptable. Based on the above, the NRC staff finds the design and proposed operation to be ALARA and therefore acceptable.

The normal operation source terms described in FSAR Section 12.2 are used as the basis for determining the radiation dose rates for demonstrating compliance with the requirements of GDC 4 and 10 CFR 50.49(e)(4). The NRC staff reviewed information provided in the FSAR to determine how the applicant complied with the requirements of 10 CFR 50.49(e)(4) and GDC 4. The staff reviewed information in FSAR Section 12.2 to verify that it is consistent with the FSAR TID for SSCs listed in FSAR Appendix 3C, Table 3C-1, “Environmental Qualification Zones—Reactor Building,” Table 3C-6. This table provides the normal operation integrated dose in and around the NuScale module containment vessel and throughout the facility caused by normal operations.

The applicant indicates that they are using the methodology in TR-0915-17565-NP-A, Revision 4, for accident EQ dose, which is based on the TS RCS specific activity limits and the methodology for calculating the design-basis iodine spike source terms in TR-0915-17565. FSAR Section 12.2.1.13, “Post-Accident Sources,” states that the maximum post-accident activity concentrations for the iodine spike design-basis source term used for EQ evaluation are provided in FSAR Table 12.2-31, “Maximum Post-Accident Radionuclide Concentrations.” The design-basis iodine spike source term was found acceptable for DBA EQ dose in the DCA design. The applicability of the design-basis iodine spike source term for determining the limiting EQ dose following a DBA in the NuScale SDA is discussed in SER Section 3.11.

The NRC staff reviewed the methods, models, and assumptions used by the applicant to determine the source terms for the RG 1.143 classification of SSCs for the radioactive waste processing components. The staff concludes that the applicant has appropriately determined the source content used to establish the RG 1.143 classifications and has appropriately classified radioactive waste SSCs. Sections 11.2, 11.3, and 11.4 of this SER present additional information on the NRC staff review of how the applicant addressed the guidance of RG 1.143.

Based on the above, the staff finds that the methods, models, and assumptions for developing the source terms for contained sources are acceptable and that the source terms provided are consistent for the assumptions described. These source terms are used elsewhere in the FSAR, such as in FSAR Section 3.11 and FSAR Section 12.3, “Radiation Protection Design Features,” to evaluate doses to equipment, occupational workers, and the public.

12.2.4.3 Airborne Radioactive Material Sources

The NRC staff reviewed the description of airborne radioactive material sources in the plant that are considered in the design of the ventilation systems, which are used for the design of personnel protective measures and for dose assessment. The NRC staff’s review verified that the applicant has provided a tabulation of the calculated concentrations of radioactive material, by nuclides, expected during normal operation. In FSAR Section 12.2, the applicant provided tables that, when used with information in FSAR Section 11.1 and Chapter 15, facilitate the calculation of potential airborne activity concentrations in the RXB following an accident.

The NRC staff reviewed concentrations of particulate, iodine, and noble gas airborne radioactive material within the RXB, as discussed in FSAR Section 12.2. The staff review included comparing the staff’s estimates of airborne particulate activity to those in FSAR Section 12.2. Table 12.2-30, “Reactor Building Airborne Concentrations,” provides the airborne concentrations in the CVCS pump room, degasifier room, and the air space above the reactor pool area. For the pool water, the pool surface temperature was assumed to be 100 degrees Fahrenheit for the calculations. The staff review showed that, based on the concentrations of radionuclides in the UHS pool water and the resuspension rates of particulates due to the evaporation of the UHS pool water, the principal contributor to dose from airborne activity within the RXB was airborne tritium.

DSRS Section 12.2 states, in part, that for nuclear power plants designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations should be based on a primary coolant concentration of 130 kilobecquerels per gram (3.5 microcuries per gram (μCi/g)) or an alternate value for which the application provides the methods, models, and assumptions. This value is an important consideration in evaluating the concentration of airborne tritium. The RCS coolant tritium concentration depends on how much tritium is produced in the coolant, the amount of tritium removed via letdown, and the amount of tritium added as a result of recycling processed RCS wastewater, as makeup feed water for the CVCS, during boron dilution. This is the amount of tritium that will be added to the pool water when the containment vessel is disassembled for refueling. Using recycled RCS water as makeup water for the UHS pool will increase the concentration of tritium in the UHS pool. Tritium is removed from the UHS pool primarily through evaporation. The amount of evaporation from the UHS pool depends on the air flow rate across the surface of the pool water, the surface area of the pool water, and the temperature difference between the pool water and the building air temperature. The NRC staff determined that the method used by the applicant to calculate RCS tritium concentration included the use of tritiated water for all RCS makeup. FSAR Table 11.1-8, “Tritium Concentration versus Primary Coolant Recycling Modes,” contains the primary coolant average concentration of tritium, and a footnote to the table notes the maximum calculated peak primary coolant tritium activity of 3.3 microcuries per gram. Because the staff’s review of the applicant’s calculations verified that the RCS tritium activity was at this peak value for a short period of time, and because the value used by the applicant for RCS tritium calculations was reasonable, the method proposed to calculate tritium concentrations is acceptable to the staff.

The NRC staff reviewed the sources of tritium that contribute to the equilibrium concentration of tritium in the UHS pool, provided in FSAR Table 12.2-9. One of the potential sources of tritium in the UHS pool is direct activation of water by neutrons escaping the containment vessel. Other major sources of tritium in the UHS pool water include neutron absorption by lithium, which is used for pH control, and boron, which is used for reactivity control. Through independent analysis performed using physical plant parameters available in the FSAR, the NRC staff compared the tritium concentrations in the FSAR for the NuScale US460 to those in the previously approved NuScale US600 design and found that the tritium concentrations were consistent between the two designs, given changes in the reactor power level, number of reactors, and size of the UHS pool. The staff finds that the FSAR contains sufficient information about the concentration of tritium in the UHS pool water and is therefore acceptable.

The NRC staff evaluated the amount of airborne tritium that could be present in the RXB atmosphere and the methods, models, and assumptions used by the applicant to determine the concentration of airborne tritium. The NRC staff and the applicant recognized that the amount of airborne tritium was related to how much tritium was in the UHS pool water, how much tritium evaporated from the pool, and, finally, how much tritium was removed by the RXB ventilation system. Since the processes used to clean wastewater do not remove tritium, the concentration of tritium in any water removed from the RCS and recycled back to the RCS will remain unchanged. The staff reviewed FSAR Section 12.2.2.1, “Reactor Building Atmosphere,” FSAR Table 12.2-29, “Input Parameters for Determining Facility Airborne Concentrations,” and information made available for staff audit and determined that the applicant adequately accounted for these processes and that the applicant adequately considered tritium in the airspace above the UHS pool. The airborne concentrations are expected to be below concentrations that would result in an airborne radioactivity area. As a result, the staff finds the airborne activity concentrations above the UHS pool to be acceptable.

Other than the UHS pool area airspace, the NRC staff review indicates that the locations of most of the major potential sources of airborne radioactive material, such as filters, operating pumps, high-pressure fluid systems, or systems directly connected to the RCS (e.g., CVCS), were contained within the RXB. Although the NuScale design minimizes the potential for leakage of radioactive fluids, the applicant assumed that there was leakage in the CVCS pump/valve rooms on the 35-foot, 8-inch elevation of the RXB and in the degasifier rooms on the 24-foot elevation of the RXB and calculated airborne radioactive material concentrations in these areas.

The regulations at 10 CFR 20.1003 use, as described in the following, two criteria to define an airborne radioactivity area: (1) the derived air concentration exceeding the values of Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure: Effluent Concentrations: Concentrations for Release to Sewerage,” to 10 CFR Part 20 or (2) the presence of airborne radioactivity concentrations in the area such that an individual without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the Annual Limits on Intake or 12 Derived Air Concentration hours. As a result of the review of the RXB airborne source terms and associated information, the NRC staff finds that the particulate activity concentration contained within the RXB atmosphere is unlikely to result in the area being classified as an airborne radioactivity area, in accordance with 10 CFR 20.1003. Since there are not any areas that are likely to meet the definition of an airborne radioactivity area and since the areas that would be expected to contain significant airborne radioactivity are limited, the concentration of airborne particulate material is unlikely to be a significant contributor to the occupational radiation exposure of workers in the RXB. Therefore, the NRC staff finds the probable concentrations of particulate, iodine, and noble gas airborne radioactive material within the RXB to be acceptable.

The NRC staff reviewed how the applicant considered sources of airborne radioactive material within the RWB. FSAR Section 12.2.2, "Airborne Radioactive Material Sources," does not discuss the sources of airborne radioactivity within the RWB, but FSAR Chapter 12 does identify assumptions relevant to the determination of airborne radioactivity concentrations in the RWB. For example, tanks and vessels, including resin storage tanks, located within the RWB are vented to the building ventilation system. There are also area and airborne radiation monitors in selected areas (as provided in FSAR Tables 12.3-8, "Fixed Area and Airborne Radiation Monitors Post-Accident Monitoring Variables" through 12.3-10, "Fixed Area Radiation Monitors") that can be used to identify potential airborne radioactivity that could be present, such as airborne radioactivity that could occur as a result leakage from equipment. An example is an airborne radiation monitor near the gaseous waste management system. The monitors will alert operators, which result in operators taking appropriate actions to limit airborne radioactivity concentrations. The staff found that the design of the RWB and systems within the RWB provide reasonable assurance that airborne activity concentrations within the RWB, from those sources, would be maintained within the limits of 10 CFR Part 20. The staff determined that the applicant adequately described the potential sources of airborne radioactive material within the RWB. Therefore, the staff finds the description of the sources of airborne radioactive material in the RWB acceptable.

Based on the above, the staff finds that the methods, models, and assumptions for developing the source terms for airborne sources are acceptable and that the source terms provided appear consistent with the assumptions described. These source terms are used elsewhere in the FSAR, such as in Section 3.11 and Section 12.3, to evaluate doses to equipment and occupational workers.

12.2.5 Combined License Information Items

Table 12.2-1 lists the COL information item number and description related to radiation sources from FSAR Table 1.8-2, "Combined License Information Items."

Table 12.2-1 NuScale COL Information Items for FSAR Section 12.2

COL Item No.	Description	FSAR Section
12.2-1	An applicant that references the NuScale Power Plant US460 standard design will describe additional site-specific contained radiation sources that exceed 100 millicuries (including sources for instrumentation and radiography) not identified in Section 12.2.1.	12.2.1.14

12.2.6 Conclusion

NuScale described contained and airborne radioactivity sources used as inputs for the dose assessment and for shielding and ventilation designs. The applicant also included the assumptions used in arriving at quantitative values for these contained and airborne source terms based on the guidance of DSRS Section 12.2 or justified appropriate alternative methodologies. NuScale TR-0915-17565-NP-A, Revision 4, provides the methodology and source terms for accident sources. The SER for TR-0915-17565 provides more information on the NRC staff evaluation of these source terms. FSAR Section 3.11 and SER Section 3.11 provide more information on source terms and doses for equipment qualification.

During power operation, the greatest potential for personnel external dose is from neutron and gamma shine from the NPM bays, fission products and corrosion and activation products contained in individual module and facility liquid and gaseous processing systems, and contaminated and irradiated NPM components during refueling evolutions. The applicant provided methods, models, assumptions, and tabulated data related to the NRC staff's evaluation of the kinds and quantities of radioactive material for contained sources of direct radiation exposure to occupational workers and members of the public.

In the RXB, the main sources of airborne radioactivity are from evaporation from the UHS pool and leakage from system components located in the equipment compartments. The applicant has tabulated the maximum expected routine radioactive airborne concentrations for areas where airborne radioactive material is more likely to be present (i.e., in the CVCS pump/valve room, the degasifier room, and the airspace above the reactor pool).

The NRC staff focused its review particularly on the aspects of the NuScale application that were radiologically different in concept or implementation from currently licensed large light-water PWRs and in areas with larger source terms and potentially greater radiological consequence. Accordingly, the staff deemphasized its review of sections of the application for which aspects of the NuScale design were less radiologically significant than the currently licensed fleet.

As described above, the NRC staff has reviewed the applicant's submittal against the requirements of 10 CFR Part 20, as it relates to occupational dose limits and ALARA requirements for occupational workers and members of the public, the generally applicable environmental radiation standards in 40 CFR Part 190; 10 CFR 50.34(f)(2)(vii); 10 CFR 50.49(e)(4); 10 CFR 52.137(a)(5); 10 CFR 52.137(a)(22) and GDCs 4, 19, and 61, as they relate to the information on radiation sources provided by the applicant; and 10 CFR 20.1406(a) and 10 CFR 52.137(a)(6), as they relate to the identification of sources of radioactive material that could lead to the contamination of the facility, contamination of the environment, or the generation of radioactive waste. The staff determined that the applicant provided adequate data on the kinds and quantities of radioactive material expected to be present in the plant and that the applicant adequately described the methods, models, and assumptions used to determine the quantities of radioactive material expected to be present, in accordance with the regulatory requirements listed above.

12.3 <u>Radiation Protection Design Features</u>

This section covers both Section 12.3 and Section 12.4, "Dose Assessment," of the FSAR because NuScale DSRS Section 12.3–12.4, "Radiation Protection Design Features," combines both sections.

12.3.1 Introduction

This section focuses on radiation protection design features, including the equipment used for ensuring that occupational radiation exposure will be ALARA. This section also considers dose rates during normal operation, AOOs, and accident conditions. Radiation zones are defined for various modes of plant operation. Design features to control personnel radiation exposures include the physical layout of equipment, shielding, and barriers to high-radiation areas (HRAs); fixed area radiation; and continuous airborne radioactivity monitoring instrumentation, including instrumentation for accident conditions. The estimated annual personnel doses associated with major functions, such as operation, handling of radioactive waste, normal maintenance, special maintenance (e.g., steam generator (SG) tube plugging), refueling, and inservice inspection, provide a measure of the effectiveness of the proposed design features in reducing overall area dose rates.

12.3.2 Summary of Application

ITAAC: SDAA Part 8 includes ITAAC associated with this section including in Section 2.4, “Equipment Qualification—Module-Specific”; Section 2.7, “Radiation Monitoring—Module Specific”; Section 3.3, “Reactor Building Heating Ventilation and Air Conditioning System”; Section 3.9, “Radiation Monitoring—Shared-Systems”; Section 3.11, “Reactor Building”; Section 3.12, “Radioactive Waste Building”; and Section 3.14, “Equipment Qualification—Shared Equipment.” These sections describe design features that demonstrate compliance with the occupational and public radiation safety requirements of 10 CFR Part 20; the requirements to protect SSCs from the effects of the potential radiation environment that may be present during normal operation and potential accidents; the monitoring of radiation levels in the plant following potential accidents; and monitoring of radiation levels in areas where fuel is stored or handled, including those sections that address radiation shielding and zoning for radiological areas of the plant and radiation monitors, including the under-the-bioshield accident radiation monitors and the MCR ventilation accident radiation monitors. The staff findings related to ITAAC are made in Chapter 14 of this SER.

FSAR Sections 12.3 and 12.4: The applicant described radiation protection design features in FSAR Sections 12.3 and 12.4, which are summarized as follows:

- The RXB, which provides shielding for the protection of MCR operators and members of the public, contains six individual reactor modules in a common pool of water. The water provides radiation shielding during normal operation, refueling, and accident conditions.

- The RPV contains an integral pressurizer and SGs. RCS fluid is circulated through the core and SGs through natural convection. The RPV is located inside a steel containment vessel that is evacuated to near 0 pascals (absolute) (0 pounds per square inch, absolute) pressure. The containment vessel is partially immersed in a pool of water. The water serves as the UHS and as the primary biological shielding.

- Borated high-density polyethylene (HDPE) shielding material is used for neutron attenuation for the portion of the containment vessel located above the UHS pool water level.

- Shielded compartments are provided for module-specific CVCS components located inside the RXB, outside the NPM bays that contain the individual reactor modules.

- Shielding and controls are provided for common liquid and gaseous waste processing equipment contained within the RXB and the RWB.

- Ventilation provisions to protect workers from airborne radioactive material include air pressure gradients from low potential airborne contamination areas to areas of higher potential airborne contamination and then the exhaust of the air through filters.

- Radiation monitoring equipment is provided for monitoring areas where fuel is stored or handled (including the movement of NPMs during refueling). Radiation monitoring is also used to monitor area and airborne radiation levels at various locations throughout the plant.

Technical Specifications: SDAA Part 4, Chapter 16, “Technical Specifications,” Section 5.7, “High Radiation Area,” addresses TS for the control of HRAs.

Technical Reports: There are no technical reports associated with this section.

Topical Reports: NuScale TR-0915-17565, Revision 4

12.3.3 Regulatory Basis

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

- 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003, as they relate to licensees making every reasonable effort and ensuring engineering controls to maintain radiation exposures ALARA

- 10 CFR 20.1201, as it relates to occupational dose limits for adults

- 10 CFR 20.1201; 10 CFR 20.1202; 10 CFR 20.1203; 10 CFR 20.1204; 10 CFR 20.1701, “Use of process or other engineering controls”; and 10 CFR 20.1702, “Use of other controls,” as they relate to design features, ventilation, monitoring, and dose assessment for controlling the intake of radioactive materials

- 10 CFR 20.1301 and 10 CFR 20.1302, “Compliance with dose limits for individual members of the public,” as they relate to the facility design features that affect the radiation exposure to a member of the public from noneffluent sources associated with normal operations and AOOs

- 10 CFR 20.1406 and 10 CFR 52.137(a)(6), as they relate to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the generation of radioactive waste

- 10 CFR 20.1601, “Control of access to high radiation areas”; 10 CFR 20.1602, “Control of access to very high radiation areas”; 10 CFR 20.1901, “Caution signs”; 10 CFR 20.1902, “Posting requirements”; 10 CFR 20.1903, “Exceptions to posting requirements”; and 10 CFR 20.1904, “Labeling containers,” as they relate to the identification of potential sources of radiation exposure and the controls of access to work within areas of the facility with a high potential for radiation exposure

- 10 CFR 20.1801, as it relates to securing licensed materials against their unauthorized removal from the place of storage

- 10 CFR 50.34(f)(2)(vii) (using the NuScale-specific source term), which requires the performance of radiation shielding design reviews to ensure that the design permits adequate access to important areas and provides for protection of safety equipment from radiation following an accident

- 10 CFR 50.34(f)(2)(xvii) (using the NuScale-specific source term), which requires the applicant to provide instrumentation to monitor containment radiation intensity (high level)

- 10 CFR 50.34(f)(2)(xxvi), as it relates to minimizing leakage from systems outside of containment

- 10 CFR 50.49(e)(4), which requires the determination of the radiation environment expected during normal operation and the most severe DBAs and requires electric equipment relied on to remain functional during and following DBEs, including AOOs

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

- 10 CFR 50.34(f)(2)(vii), which requires radiation and shielding design reviews of spaces around systems that may, as the result of an accident, contain accident source term radioactive material, and to design as necessary to permit adequate access and to protect safety equipment from the radiation environment

- 10 CFR 50.68, “Criticality accident requirements,” or 10 CFR 70.24, “Criticality accident requirements,” as they relate to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled

- GDC 14, “Reactor coolant pressure boundary,” and GDC 30, “Quality of reactor coolant pressure boundary,” as they relate to the ability to detect RCS pressure boundary leakage with radiation detectors

- GDC 19, as it relates to the provision of adequate radiation protection to permit access to areas necessary for occupancy after an accident without personnel receiving radiation exposures in excess of the 50 mSv (5 rem) total effective dose equivalent, as defined in 10 CFR 50.2, “Definitions,” to the whole body or the equivalent to any part of the whole body for the duration of the accident

- GDC 61, as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems that may contain radioactivity designed to ensure adequate safety during normal and postulated accident conditions with suitable shielding and appropriate containment and filtering systems

- GDC 63, “Monitoring fuel and waste storage,” as it relates to detecting excessive radiation levels in the facility

- 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” Section VI.2(a)(i), which requires radiation monitoring systems for reactor coolant radioactivity, containment radiation level, condenser air removal radiation level, and process radiation monitor levels

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

- 10 CFR 52.137(a)(22), as it relates to ensuring that the application includes information necessary to demonstrate how the plant design incorporates operating experience insights

- 10 CFR 52.47(b)(1), which requires an FSAR to contain the ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the AEA, and NRC regulations

The guidance in DSRS Section 12.3–12.4 lists the acceptance criteria that are adequate to meet the above requirements and review interfaces with other DSRS or applicable SRP sections, and it references the following:

- RG 1.7, as it relates to protection from radionuclides in systems used for determining gaseous concentrations in containment following an accident

- RG 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” as it relates to minimizing occupational radiation exposure through the selection of locations for installing seismic monitoring equipment and the selection of equipment design specifications that reduce the frequency or duration of testing, inspection, or maintenance of seismic monitoring equipment

- RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” as it relates to the detection capabilities of radiation monitors described in DSRS Section 12.3-12.4 that are provided for RCS pressure boundary leakage detection to the extent that they are not addressed in other sections of the DSRS

- RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” as it relates to radiation protection considerations for engineered-safety-feature atmosphere cleanup systems that are operable under postulated DBA conditions to be designated as “primary systems”

- RG 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” as it relates to the requirements and recommended practices acceptable for construction of facilities that apply to occupational radiation protection shielding structures for nuclear power plants

- RG 1.89, as it relates to the determination of radiation dose to certain electrical equipment important to safety as described in 10 CFR 50.49

- RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” the SRP; DSRS Section 11.6, “Guidance on Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring,” and a memorandum from D.G. Eisenhut, Office of Nuclear

Reactor Regulation, to Regional Administrators, dated August 16, 1982 (ML103420044), as they relate to a method acceptable to the NRC staff for complying with NRC regulations that require the licensee to provide and calibrate radiation monitoring instrumentation and as they relate to monitoring plant variables and systems that are important to safety during and following an accident

- RG 1.97, DSRS Chapter 7, "Instrumentation and Controls—Introduction and Overview of Review Process," and a memorandum from D.G. Eisenhut, Office of Nuclear Reactor Regulation, to Regional Administrators, dated August 16, 1982, as they relate to methods acceptable to the NRC staff for complying with NRC regulations to provide and calibrate, or verify the calibration of, safety-related instrumentation for radiation monitoring following an accident in a nuclear power plant

- RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," as it relates to actions taken to address the guidance in RG 8.8, Regulatory Position C.2(d), during facility design, engineering, construction, and decommissioning to maintain occupational radiation exposure ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 with regard to the radiation protection information provided in FSAR Chapter 12

- RG 1.143, as it relates to design features provided to minimize occupational radiation exposure and classification of structures that house radioactive waste systems based on potential exposure to site personnel

- RG 1.183, as it relates to the assumptions and methods for evaluating doses to individuals who access the facility during and following an accident in accordance with NUREG-0737, Task Action Plan Item II.B.2

- RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," as it relates to the design features provided to minimize the contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste

- RG 8.2, "Administrative Practices in Radiation Surveys and Monitoring," as it relates to general information on radiation monitoring programs for administrative personnel

- RG 8.8, as it relates to actions taken during facility design, engineering, construction, operation, and decommissioning to maintain occupational radiation exposure ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 concerning the radiation protection information to be included in FSAR Chapter 12

- RG 8.10, as it relates to the commitment by management and vigilance by the radiation protection manager and NRC staff to maintain occupational radiation exposure ALARA in accordance with 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003

- RG 8.15, "Acceptable Programs for Respiratory Protection," as it relates to methods acceptable to the NRC staff for ensuring the safety of personnel who use an installed breathing air system provided for radiological respiratory protection

- RG 8.19, “Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants—Design Stage Man-Rem Estimates,” as it relates to a method acceptable to the NRC staff for assessing collective occupational radiation doses as part of the ongoing design review process to ensure that such exposures will be ALARA

- RG 8.25, “Air Sampling in the Workplace,” as it relates to a method acceptable to the NRC staff for continuous monitoring of airborne radioactive materials in plant spaces

- RG 8.38, “Control of Access to High and Very High Radiation Areas of Nuclear Plants,” as it relates to the physical controls for personnel access to HRAs and very high radiation areas (VHRAs)

- SRP Branch Technical Position 11-3, “Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants” (ML070730202), and SECY-94-198, “Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste” (ML071640462), dated August 1, 1994, as they relate to design features provided to minimize occupational radiation exposure for the radioactive waste storage facilities described in the application

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

- ANSI/ANS HPSSC-6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors,” as it relates to criteria for the establishment of locations for fixed continuous area gamma-radiation monitors and for design features and ranges of measurement

- ANSI/Health Physics Society N13.1-2011, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities,” as it relates to the principles that apply in obtaining valid samples of airborne radioactive materials and the acceptable methods and materials for gas and particle sampling

- ANSI/ANS 6.4-2006, “Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants,” as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures

- Memorandum from L.W. Camper to D.B. Matthews and E.E. Collins, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing,” dated October 10, 2006 (ML062620355), and NUREG/CR-3587, “Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors,” issued June 1986 (ML081360413), as they relate to the design issues that licensees need to address to meet the requirements of 10 CFR 20.1406

- NEI 97-06, “Steam Generator Program Guidelines,” as it relates to the leakage detection capabilities of the radiation monitoring equipment described in FSAR Chapter 12, which are provided to detect SG tube leakage in accordance with the criteria in the EPRI bases documents to the extent that other DSRS sections do not address them

12.3.4 Technical Evaluation

The NRC staff reviewed the radiation protection design features, dose assessment, and minimization of contamination design considerations in FSAR Sections 12.3 and 12.4 and in other related sections of the FSAR for consistency with the guidance in DSRS Section 12.3–12.4. The purpose of this review was to ensure that the applicant had either committed to following the guidance of the RGs and applicable NRC staff regulatory positions or offered acceptable alternatives. In areas where the FSAR is consistent with the guidance in these RGs and NRC staff regulatory positions, the staff can conclude that the relevant requirements of 10 CFR Part 20 and 10 CFR Part 50 have been met. The sections below present the staff's findings.

Several aspects of the radiation protection design features of the NuScale SMR design differ from those traditionally found in PWRs. The RXB contains up to six reactors that are each enclosed in a separate containment vessel, which is submerged in a common pool of water that is used as the UHS. The UHS, instead of concrete, provides the primary shielding (i.e., the shielding immediately around the reactor vessel). During refueling, the containment vessel, including the contained reactor vessel and all the fuel for that reactor, is moved as an integral unit to the attached refueling pool. In the refueling portion of the pool, the containment and reactor vessels are disassembled, and fuel is removed and placed in the attached spent fuel pool (SFP). The three connected pools (the reactor cooling pool, the refueling pool, and the SFP are interconnected and enclosed by a single RXB. While one reactor is being refueled, up to five other reactors, located in the contiguous reactor cooling pool, may continue to operate. The SGs utilize helical coil tubes, with the secondary coolant on the inside of the tubes and reactor coolant on the outside of the tubes. Control rods and the associated drive mechanisms are all fully contained within the reactor vessel. Spent resin and liquid waste storage tanks and related processing SSCs are in a separate building adjacent to the RXB.

Radiation sources and storage areas described in the design application are located within the restricted area, which addresses the requirement of 10 CFR 20.1801. A COL applicant's radiation protection program must ensure that the requirements of 10 CFR 20.1801 and 10 CFR 20.1802 are met for any sources in controlled or unrestricted area (See COL Item 12.5-1 and FSAR Chapter 13 for radiation protection program and procedure requirements).

12.3.4.1 Radiation Protection Design Features

The facility design incorporates features to help maintain occupational radiation exposure ALARA in accordance with the guidance in RG 8.8 and the requirements of 10 CFR 20.1101(b), and to maintain radiation exposures for workers and members of the public to within the limits of 10 CFR Part 20. The facility design includes features for minimizing contamination of the facility and the environment, minimizing the amount of waste generated, and minimizing the cost of decommissioning. Design features are provided to monitor radiation fields within the facility for the protection of workers, the assessment of potential accident conditions, and the radiation fields where fuel is stored or handled. These design features include facility design, shielding, ventilation, and area and airborne radiation monitors.

12.3.4.1.1 Facility Design Features

Because the containment vessel is not accessible by personnel during operation, the sources of radiation inside the containment during operation do not present a hazard to personnel. The shielding provided by the water in the UHS and the concrete structure of the NPM bay reduces radiation levels to workers and SSCs from the reactor and irradiated reactor components.

In its review of FSAR Section 12.3.1, "Facility Design Features," the NRC staff found that the NuScale design incorporates many features of large light-water reactors (LWRs) that have been shown to be effective in reducing radiation exposures to workers (consistent with ALARA), minimizing contamination of the facility, minimizing the generation of waste, and facilitating decommissioning.

Based on information in FSAR Section 12.3.1, the types of materials used in the construction of the facility are specified to reduce corrosion rates and improve equipment reliability. Stainless steel or stainless steel clad is used for components and piping in contact with primary coolant and reactor pool water. Thermally treated Alloy 690 base metal is used for SG tubing material. Stainless steels and thermally treated Alloy 690 metal are used to reduce the possibility of intergranular stress-corrosion cracking, which could reduce equipment failure rates and, therefore, reduce worker dose resulting from maintenance activities. Low alloy steels, stainless steel clad, and austenitic stainless steel is used for the containment vessel, which is in continuous contact with the UHS pool water. In FSAR Section 12.3.1, the applicant stated that the use of cobalt and nickel is minimized to reduce the quantity of activation products to the extent practicable. FSAR Table 12.3-3, "Typical Cobalt Content of Materials," lists the maximum weight percent of cobalt for different components. The staff finds that, based on the information in the FSAR, including Table 12.3-3, the design appropriately limits cobalt, consistent with the guidance in RG 8.8 and RG 4.21, and is thereby consistent with the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406.

Tanks containing radioactive material have bottoms that slope toward outlets and, where practicable, provide built-in spray features, spargers and eductors for mixing tank contents and reducing sedimentation in tanks, thereby reducing the local source term. Tanks containing radioactive material have a smooth interior finish that minimizes crud traps. Remotely actuated valves are used to minimize personnel exposure, where practicable. Many valves are located in valve galleries to provide additional shielding. Double isolation valves are used at the interface between contaminated and noncontaminated systems to prevent cross-contamination. Piping is designed to exclude crevices and crud traps and includes smooth internal surfaces to the extent practicable, and many systems are designed with flushing capabilities. These are just a few design features identified in FSAR Section 12.3 to reduce radiation exposure and minimize contamination in the NuScale design. These design features are consistent with the guidance of RG 8.8. Section 12.3.4.1.5 of this SER discusses additional design features related to minimizing contamination and the staff's evaluation of these features. Since the design is consistent with the requirement of 10 CFR 52.137(a)(22) to demonstrate how operating experience insights have been incorporated into the plant design, the requirements of 10 CFR 20.1406(a) to provide design features for minimizing the amount of waste generated, and the requirements of 10 CFR 20.1101(b) to provide design features for maintaining occupational radiation exposure ALARA, the NRC staff finds this approach acceptable.

The NRC staff reviewed how the design limits doses to members of the public from direct sources (i.e., contained sources) of radioactive material. The applicant has a large pool surge control storage tank located outside of the RXB. The pool surge control storage tank is designed to temporarily store cleaned up pool water that is displaced during dry dock operations. This tank would likely be the largest potential source of direct radiation exposure to members of the public from a NuScale facility during normal operation. The staff confirmed that the design of the facility includes the ability to remove radioactive material from the water before it is pumped to the pool surge control storage tank and from water in the pool surge control storage tank. This provides reasonable assurance that the design will allow operators to adequately control doses to members of the public from the pool purge control storage tank and is therefore acceptable.

Consistent with the guidance in RG 8.8, the applicant considered the radiation levels, the access frequency, and the duration of access of personnel when establishing radiation zone maps. The applicant based the radiation zones on the maximum dose rate in the area, which is consistent with the guidance in RG 8.8. Based on this approach, the applicant provided normal operation radiation zone maps in FSAR Section 12.3 for portions of the RXB and RWB. The applicant also provided radiation zones related to EQ (see Section 3.11 of this SER for the NRC staff's evaluation of EQ) in FSAR Section 3.11. The applicant provided airborne radioactivity zones in FSAR Table 12.3-2, "Airborne Radiation Zone Designations," for those portions of the facility with the potential for airborne radioactivity. Based on the absence of identified vital area missions (i.e., no expectation for personnel to be in the area) for non-core-damage accidents, the applicant did not provide post-accident radiation zone maps. There are no direct requirements for radiation zone maps in 10 CFR Part 20, 10 CFR Part 50, or 10 CFR Part 52. As such, the NRC staff evaluated the proposed use of radiation zone maps to help keep occupational radiation exposure ALARA and to support meeting other regulatory requirements. Some of the radiation zone designations are based on areas that have unrestricted access, areas that are required to be posted as radiation areas, high radiation areas, locked high radiation areas, and very high-radiation areas in accordance with 10 CFR Part 20 requirements and locked high radiation area technical specifications requirements specified in NuScale US460 technical specification 5.7.2. The staff also observed that the radiation zone maps show that corridors and areas that would be expected to be routinely accessed are low dose areas. Based on this, the staff determined that the radiation zoning is consistent with the requirements of 10 CFR 20.1101(b).

FSAR Section 12.3.1.3.1, "Normal Conditions," provides information on controls and design features for HRAs and VHRAs. It specifies that HRAs either are locked or have alarmed barriers and that VHRAs are locked. It also states that positive control is exercised, and egress from the area is not impeded for each HRA and VHRA. The applicant stated in FSAR Section 12.3.1.3.1 that, based on design and calculation, no VHRAs are identified in the NuScale US460 design. However, the staff recognizes that areas inside of containment and near reactor fuel would likely be VHRAs, but these areas cannot be accessed during operation and irradiated fuel remains under significant water coverage. While the applicant indicated that it has not identified any VHRAs, it did provide COL Items 12.3-1 and 12.3-2, which specify that the COL applicant will develop administrative controls for access to HRAs and VHRAs in accordance with the guidance of RG 8.38. The acceptance criteria in DSRS Section 12.3–12.4 for VHRAs state that the facility design should ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 5 Gy (500 rad) or more in 1 hour at 1 meter (3.3 feet) from a radiation source or any surface through which the radiation penetrates (e.g., those adjacent to operating reactors or irradiated portions of reactors or containment vessels of shutdown reactors). Given that the applicant provided information on access and egress from VHRAs and provided a COL item for the COL applicant to develop the administrative controls for access to VHRAs, the NRC staff concludes that the identification of the VRHAs and the description of the design features specified for VHRAs are consistent with the requirements of 10 CFR 52.137(a)(5) for providing controls to maintain radiation exposures within the limits of 10 CFR Part 20. Further, this is consistent with the requirements of 10 CFR Part 20, Subpart G, "Control of Exposure from External Sources in Restricted Areas," and 10 CFR 20.1101(b) to maintain occupational radiation exposure ALARA. Therefore, the staff finds the information provided on controls and design features for VHRAs to be acceptable.

12.3.4.1.2 Shielding

The objective of the plant's radiation shielding is to minimize plant personnel and public exposures to radiation during normal operation (including refueling and maintenance), AOOs, and accident conditions while maintaining a program of controlled personnel access to, and occupancy of, radiation areas. In addition to protecting workers and members of the public, the design also includes shielding, where necessary, to mitigate the possibility of radiation damage to materials (see SER Section 3.11 for the NRC staff's evaluation of EQ) from radiation resulting from normal operation, from DBEs, and from core damage events (see Section 19.2 and Section 15.0.3 of this SER for the NRC staff's equipment survivability evaluation). Shielding is provided to attenuate direct and scattered radiation through walls and penetrations to less than the upper limit of the radiation zone for each area in the RXB and the RWB, as discussed in FSAR Section 12.3.1.2.3, "Penetrations."

Using a risk-informed approach, the NRC staff's evaluation of radiation shielding focused on areas of the facility that could contain high concentrations of radioactive materials during normal operation or following accidents. Using this approach, the staff focused the shielding review on areas where neutron and gamma radiation are experienced from the NPMs as a result of operation; gamma radiation emitted from the NPMs following accidents; plant components, such as demineralizers, filters, and charcoal beds, that concentrate radioactive material; irradiated components, such as the self-powered neutron detectors; large masses of radioactive material, such as the UHS pool; and areas of the facility where high concentrations of radioactive material may accumulate, such as SRSTs, PSTs, and HIC storage areas. For the selected areas, the staff compared the amount of radioactive material in the SSCs to the shielding provided and the resultant dose rates. The staff used computer programs such as MicroShield to perform confirmatory and scoping calculations and audited calculation packages associated with the applicant's calculations. The staff also notes that it used knowledge gained as part of the NuScale US600 review to help inform the US460 review, as appropriate.

FSAR Table 12.3-5, "Reactor Building Shield Wall Geometry," and Table 12.3-6, "Radioactive Waste Building Shield Wall Geometry," provide concrete shielding thicknesses for rooms and cubicles containing significant radiation sources, which require shielding. These shielding thicknesses are based on the source terms provided in FSAR Chapter 11 and Section 12.2 and consider accident source terms consistent with TR-0915-17565, where appropriate. The applicant used SCALE and MCNP to perform source term and radiation shielding and zoning dose calculations. For most shielding applications in the NuScale design, concrete shielding is designed in accordance with NRC-endorsed ANSI/ANS 6.4-2006. The NRC staff finds that methods described by the applicant for performing shielding evaluations are consistent with those identified in NRC guidance and are, therefore, acceptable.

FSAR Section 12.3.2.2, "Design Considerations," states that while concrete is the material used for a significant portion of plant shielding, other types of materials, such as steel, water, tungsten, and polymer composites, are considered for both permanent and temporary shielding. FSAR Section 12.3.2.3, "Calculation Methods," states that Tables 12.3-5 and 12.3-6 provide the radiation shielding for the RXB and RWB in terms of nominal concrete attenuation thicknesses. The only radiation shielding specified in Tables 12.3-5 and 12.3-6 that is not concrete is the high-density borated polyethylene shielding specified for the vertical bioshield. FSAR Chapter 3, "Design of Structures, Systems, Components and Equipment," describes in more detail the high-density borated polyethylene shielding design, which is used primarily to shield neutron radiation.

The applicant stated in FSAR Section 12.3.2.3 that shielding materials used in place of the specified concrete provide the equivalent radiation attenuation prescribed for the applicable gamma and neutron radiation sources. The attenuation is to be demonstrated by achieving the radiation zones depicted in FSAR Figures 12.3-1a, “Reactor Building Radiation Zone Map—25' Elevation,” through Figure 12.3-2c, “Radioactive Waste Building Radiation Zone Map—100' Elevation.” Alternative shielding is also to be verified to maintain compliance with 10 CFR 50.49, GDC 4, GDC 61, 10 CFR 50.34(f)(2)(vii), and other relevant requirements. Therefore, if using an alternative material to the specified concrete, the radiation shielding will have to ensure that radiation zoning is not affected. If other requirements, such as EQ, are impacted, the COL applicant must ensure that equipment is still appropriately zoned and qualified and make any necessary adjustments under the EQ program. The staff notes that based on the information audited, alternative shielding specified in NuScale calculation packages that is expected to be used in place of the concrete values specified in Tables 12.3-5 and 12.3-6 would provide more radiation attenuation than the specified concrete values.

In addition, based on the audited information, the staff noted that uses of alternative material are expected to be in selected areas and not widespread throughout the plant. The staff also notes that, the areas in the NuScale design expected to be radiation areas (in excess of 0.05 millisieverts (5 mrem/h)) contain significant radiation sources or areas that only need to be accessed infrequently. Shielding ensures that areas that do not contain significant radiation sources are generally less than 0.05 millisieverts (5 mrem/h) (radiation Zone 3 or less), and most walkways and areas that are routinely accessed are less than 0.025 millisieverts (2.5 mrem/h) and ALARA. In addition, the COL applicant will be required to include a radiation protection program that ensures that doses to workers and members of the public are ALARA, consistent with COL Items 12.1-1 and 12.5-1. For these reasons, the use of alternative shielding materials should not result in an increase in occupational or public dose, and if any increase in dose were to occur from using alternatives to concrete, the overall increase in radiation dose would be expected to be minimal. For these reasons, the staff finds the general radiation shielding approach provided by the applicant, including the approach for using alternative shielding to concrete in some cases, to be acceptable.

FSAR Section 12.3.2.2 states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. The acceptance criteria in DSRS Section 12.3–12.4 state that an assessment of design features provided to protect shielding material subject to degradation, such as through the effects of radiation, temperature extremes (e.g., degradation of concrete caused by high temperature), and density changes (e.g., due to drying), should be provided. The guidance in RG 1.69 discusses the use of American Concrete Institute (ACI) 349-06, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary”; ACI 349.1R-07, “Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures”; and the associated environmental constraints on concrete shielding material. The NRC staff confirmed that the applicant has committed to following the guidance of RG 1.69, which endorses ACI 349, for the design of radiation shields. Therefore, the staff finds that the specifications for the design of the concrete shielding walls provide adequate shielding for the life of the plant and are consistent with the requirements of 10 CFR 52.137(a)(5) to provide controls to maintain radiation exposures within the limits of 10 CFR Part 20 and thus are acceptable.

In FSAR Section 12.3.2.4.1, “NuScale Power Module,” the applicant stated that since degradation of the borated HDPE radiation shielding material used on the face of the NPM bioshield bay could potentially occur if the exhaust ventilation provided for the reactor module bays does not maintain air temperatures under the bioshield of less than 82.2 degrees

Celsius (°C) (180 degrees Fahrenheit (°F)) (e.g., due to damper failure). Therefore, the application states that conditions in which the air temperature under the bioshield exceeds 82.2°C (180°F) require an evaluation of the continued efficacy of the bioshield polyethylene material's radiation shielding properties. The staff reviewed the acceptability of 82.2°C (180°F) for the borated polyethylene shield material in the US600 design, which used HDPE material with 5% boron content, which is the same as the US460 design, as specified for the shielding for the module bays in FSAR Table 12.3-5. Based on the staff's review and information available for similar materials, the staff finds the temperature specification to be sufficient to ensure the integrity of the HDPE shielding material. Since a COL applicant will have to evaluate the continued efficiency of the bioshield shielding if the temperatures reach 82.2°C (180°F), because of a damper failure or any other reason, the NRC staff finds that the specifications for the design of the shielding are consistent with the requirements of 10 CFR 52.137(a)(5) to provide controls to maintain radiation exposures within the limits of 10 CFR Part 20 and are consistent with the requirements of 10 CFR 20.1101(b) to maintain occupational radiation exposure ALARA for the life of the plant. Therefore, the staff finds this specification to be acceptable.

The NRC staff also verified that the shielding panels include vents to allow the release of gases that will be generated by the boron adsorption of neutrons. As such, the staff finds that the material characteristics of the borated HDPE shielding material are compatible with the radiation environment expected during normal operation over the life of the plant, consistent with the requirements of 10 CFR 52.137(a)(5) to provide controls to maintain radiation exposures within the limits of 10 CFR Part 20, consistent with the requirements of GDC 4 to ensure that SSCs are compatible with the environmental conditions, and consistent with the requirements of 10 CFR 20.1101(b) to maintain occupational radiation exposure ALARA for the life of the plant. Thus, the staff finds the design specifications for the borated HDPE related to boron content to be acceptable.

In FSAR Section 12.3.1.2.3, Section 12.3.2.2, and Section 12.3.2.4.1, the applicant discussed penetrations, including general design features for minimizing radiation streaming through penetrations. The applicant stated that penetrations through radiation shield walls are minimized as much as practicable. When penetrations are necessary, the penetrations are designed to minimize streaming (e.g., with an offset) from a radiation source to accessible areas. If penetration offsets are not practical, then penetrations are either shielded or elevated above floor level to minimize occupational radiation exposure. The applicant also stated that penetrations are compensated to comply with the radiation zone maps during normal operation. The NRC staff review identified the presence of large penetrations, such as the main steamlines, main feedwater lines, and NPM bay heating, ventilation, and air conditioning (HVAC) lines in the radiation shield wall between the NPM bay and the RXB steam gallery area.

FSAR Section 12.3.2.4.1 indicates that pool wall penetrations into the reactor module bay are modeled using MCNP6, with the shield voids occupied by the piping, HVAC ducting, cabling, and insulation, as designed for the penetrations. The applicant stated that the model includes filler materials between the void spaces following design standard specification for shielding penetrations. During the audit, the staff reviewed calculation packages and other information related to the radiation dose on the other side of the reactor module bay walls due to the larger penetrations. The penetrations are located high above the level of the reactor core, and there is no direct streaming path for the significant radiation sources under the bioshield. The applicant's calculations modeled the dose rates through the penetrations based on the penetration locations in the design. The calculations showed low dose rates through the penetrations. Based on the staff review of the applicant's calculations, as part of the audit, the staff finds that with the assumed filler materials, the penetrations do not result in a significant increase to the dose rate on the other side of the reactor module bay wall. Based on its review, the staff finds that radiation shine through the penetrations is adequately accounted for in the radiation zoning on the other side of the module bay wall. The staff also finds the general design features described in FSAR Section 12.3.1.2.3 for penetrations in other areas, such as minimizing penetrations through shield walls as much as practicable and minimizing direct line of sight through penetrations and providing shielding for penetrations to be acceptable for keeping radiation exposure ALARA.

The NRC staff reviewed the radiation zone designations for areas in the RXB near the UHS pool. The radiation zone designations are based on the NPM as a neutron and gamma source term. The concrete mass of the bioshield cover provides shielding above the NPM. The borated HDPE that is encapsulated in steel plates provides shielding for the front of the NPM bay. The staff determined that, based on the inaccessibility of the area of the RXB corresponding to the UHS pool level near the NPM bay entrances by personnel during normal operation (i.e., the need to use a boat or a personnel basket suspended from a crane), the radiation zone designations for those portions of the RXB provide reasonable assurance that occupational radiation exposure will be controlled by the radiation protection program, will be maintained ALARA, and will be kept within the limits of 10 CFR Part 20. Therefore, the staff finds these radiation zone designations to be acceptable.

FSAR Table 12.3-7, “Radioactive Waste Building Radiation Shield Doors,” shows the applicant’s specifications for radiation shield doors in the RWB. The applicant does not credit any shield doors in the RXB, and all door openings were modeled as opened doorways. Some of the radiation shield doors provided in the RWB, as shown in FSAR Table 12.3-7, do not provide shielding equivalent to that of the radiation shield walls. Staff confirmatory calculations indicate that the shielding provided by the doors are sufficient to keep dose rates within those specified in the radiation zone maps, when controls from a radiation protection program are in place. For example, if necessary, the operational radiation protection program may ensure that high-activity sources, such as high-activity HICs, are not stored in a location in a direct line of sight to a shield door, if the HIC could result in elevated dose rates to areas outside the door. The NRC staff concludes that the doors are consistent with the requirements of 10 CFR 52.137(a)(5) for providing controls to maintain radiation exposures within the limits of 10 CFR Part 20 and are consistent with the requirement of 10 CFR 20.1101(b) to maintain occupational radiation exposure ALARA. Therefore, the staff finds that the type of radiation material identified is acceptable.

The acceptance criteria in DSRS Section 12.3–12.4 state that accessible portions of the facility that are capable of having radiation levels greater than 1 Gy/h (100 rad/h) should be shielded and clearly marked with a sign stating that potentially lethal radiation fields are possible. The staff identified several areas (e.g., CVCS resin demineralizers, CVCS filters, SRSTs) that may contain quantities of radioactive material resulting in radiation dose rates near 1 Gy/h (100 rad/h) or potentially greater. The CVCS filters and HIC storage area are accessed from above through floor shield plugs and the CVCS ion exchangers are accessed through knockout panels. As discussed in FSAR Section 12.3.2.2, shield floor plugs and knockout panels provide radiation attenuation equivalent to that of the shield floor that contains the plug and the wall containing the knockout panel. The staff evaluated the information provided and determined that the use of shield plugs and knockout panel that have a shielding value equivalent to the floor or wall in which they are installed provides reasonable assurance that radiation exposure to workers in the areas around the shield plugs will be maintained within the limits of 10 CFR Part 20 and occupational radiation exposure will be ALARA. Therefore, the staff finds this to be acceptable. The staff also reviewed the normal concrete radiation shielding specified for the potential 1 Gy/h (100 rad/h) sources. Based on staff confirmatory calculations, the staff finds the shielding in these areas to be acceptable.

The information (source term, shielding geometries, shield thicknesses) provided by the applicant allowed the staff to perform confirmatory analyses of the applicant's radiation zone designations. The staff's analyses concluded that the radiation zoning specified in the figures in FSAR Section 12.3 is accurate and appropriate with the specified shielding. The staff has reasonable assurance that the radiation zone designations will allow workers to support the operation of the facility within the radiation exposure limits of 10 CFR Part 20 and will allow workers to maintain occupational radiation exposure ALARA. While the Chapter 12 radiation zoning is fundamentally different than the FSAR Section 3.11 EQ zones, the radiation source terms, radiation shielding, and radiation zone designations shown in the Chapter 12 figures provide relevant information for FSAR Section 3.11 regarding the radiation environments for equipment with the specified shielding.

12.3.4.1.3 Ventilation

RG 8.8 and DSRS Section 12.3–12.4 provide guidance on acceptable ventilation design features to control airborne radioactivity levels and maintain personnel doses ALARA. The ventilation systems are designed to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA and is within the applicable limits of 10 CFR Part 20.

In general, for the NuScale design, ventilation pathways in radiologically controlled areas flow from areas anticipated to have lower levels of airborne activity to areas expected to contain higher levels of radioactivity. FSAR Table 12.2-29 provides the ventilation air change rates for the pool airspace, the CVCS pump/valve rooms, and degasifier rooms. SER Section 12.2 documents the NRC staff's assessment of the airborne activity concentrations.

FSAR Section 12.3.3, "Ventilation," describes ventilation system design features provided to minimize occupational radiation exposure. In areas subject to airborne activity, the ventilation systems are designed to collect, process, and exhaust airborne radioactive material, including directing airflow to processed exhausts; the building ventilation systems are designed to maintain an airflow inside the buildings from areas of low airborne potential to areas of higher airborne potential and to maintain a negative pressure with respect to the outside environment, to control the release of airborne radioactivity to the environment. FSAR Table 12.3-30, "Regulatory Guide 4.21 Design Features for Reactor Building Heating, Ventilation, and Air Conditioning System," and Table 12.3-33, "Regulatory Guide 4.21 Design Features for Radioactive Waste Building Heating, Ventilation, and Air Conditioning System," indicate that smooth finished materials are used as much as practicable for the RXB and RWB ventilation system surfaces to minimize contamination of equipment and facilitate decommissioning. This information gives the NRC staff reasonable assurance that the design of the HVAC system will be consistent with the requirements of 10 CFR Part 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas," regarding the use of engineering controls to limit radiation exposure and to provide assurance that radiation exposure will be ALARA. Chapter 11 of this SER discusses radiation detection, alarms, and actions taken as a result of high radiation levels in the plant ventilation systems.

The NRC staff reviewed how the applicant complied with the requirements of 10 CFR 50.34(f)(2)(xxvi) for the design of components to minimize leakage from systems outside of containment. Specifically, 10 CFR 50.34(f)(2)(xxvi) requires applicants to provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term (i.e., a core damage source term) radioactive materials following an accident. The staff review of 10 CFR 50.34(f)(2)(xxvi), within the context of this chapter, is discussed in Section 12.5 of this SER.

12.3.4.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

All plant radiation-monitoring equipment is designed to alert operators and other station personnel to changing or abnormally high radiation conditions in the plant to prevent possible personnel overexposures, to aid health physics personnel in keeping worker doses ALARA, and to limit releases to the environment and public. The area radiation monitors supplement the personnel and area radiation survey provisions of the health physics program, which FSAR Section 12.5, “Operational Radiation Protection Program,” directs the COL applicant to describe (see COL Item 12.5-1). The area radiation monitors must comply with the applicable requirements of 10 CFR Part 20 and 10 CFR Part 50 and should conform to the personnel radiation protection guidelines in RGs 1.97, 8.2, and 8.8 and the guidance of DSRS Section 12.3–12.4.

Radiation indications from the fixed airborne and area monitors can be read locally and in the MCR. Alarms are also provided both locally and in the MCR; some monitors also alarm in the waste management control room. ANSI/ANS HPSSC-6.8.1-1981, referenced in FSAR Section 12.3.4.2, “Fixed Area Radiation Monitoring Instrumentation,” and DSRS Section 12.3–12.4, provide examples of appropriate locations for radiation monitors in PWRs. The NuScale design includes radiation monitors in areas consistent with those identified in ANSI/ANS HPSSC-6.8.1-1981, including areas where radiation levels may change significantly during plant operation and where airborne activity levels can alert operators to unexpected leaks.

FSAR Section 12.3.4, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation,” discusses the fixed airborne and area radiation monitors. Additional information related to the monitors appears in FSAR Table 12.3-9, “Fixed Airborne Radiation Monitors,” and Table 12.3-10. FSAR Table 12.3-8 provides information on radiation monitors credited to monitor post-accident radiation levels in accordance with the guidance in RG 1.97. In large LWRs, containment high range radiation monitors are located inside of containment, which satisfy the requirements of 10 CFR 50.34(f)(2)(xvii)(D). The NuScale NPMs include a very small containment compared to large LWRs, so in the NuScale design, the under-the-bioshield radiation monitors are high-range radiation monitors that are designed to provide radiation dose information under the bioshield following accident conditions. Since NuScale has a small containment and the under-the-bioshield monitors are located near the outside of containment, under the bioshield, the under-the-bioshield monitors provide an adequate means for providing radiation intensity levels inside of containment. Therefore, the under-the-bioshield monitors meet the intent of 10 CFR 50.34(f)(2)(xvii) to monitor radiological conditions during an accident, despite being outside containment. These monitors meet the range and placement criteria specified in NUREG-0737, Task Action Plan Item II.F.1.3.

The NRC staff verified that the applicant identified these monitors as environmentally qualified for source terms corresponding to non-core-damage accidents and are post-accident monitoring Type B, Type C, and Type F variables in accordance with the guidance in Institute of Electrical and Electronics Engineers (IEEE) 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” which is referenced in RG 1.97, Revision 5, issued April 2019. The design-basis analysis source term adopted by NuScale is consistent with the methodology outlined in SECY-19-0079, “Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application,” dated August 16, 2019 (ML19107A455). The applicant stated that these radiation monitors are relied on to assess the presence of core damage. The source terms to be used for equipment survivability and the criteria to be applied to equipment expected to be able to perform a function following a core damage event are not included in the criteria for the NRC staff review of Chapter 12 of the application. Chapter 15 of this SER discusses the NRC staff’s evaluation of the radiological conditions associated with core damage source terms that are used to assess equipment survivability. Section 19.2 and Section 15.0.3 of this SER contain the staff’s evaluation of the survivability of equipment, as discussed in SECY-19-0079.

Section 3.11 of this SER and the SER for TR-0915-17565 contain more information on compliance with 10 CFR 50.49(e)(4), including for the high-range containment radiation monitors. The monitors are qualified for 720 hours following the start of a non-core-damage DBA. Therefore, the staff finds that the containment high-radiation monitors comply with the requirements of 10 CFR Part 20, Subpart F, “Surveys and Monitoring,” and the requirement of 10 CFR 50.34(f)(2)(xvii) to provide radiation monitoring instrumentation for normal operation and non-core-damage events. As discussed in Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation,” of the FSAR, the under-the-bioshield monitors are designed to survive 48 hours following an event that results in-core damage.

In its February 6, 2024, letter (ML24037A134), NuScale indicated that conditions in the core are stable after 48 hours, and at that point, changing atmospheric conditions are the relevant input to operational decision-making. The applicant also indicated that the accident dose rate information from the under-the-bioshield monitors is necessary to detect the onset of core damage, the extent of core damage, and failures of containment integrity under accident conditions. Finally, the applicant noted that various other monitors, such as the refueling pool area and airborne radiation monitors and the RXB exhaust stack airborne monitor, would also be available to indicate the radiological conditions. Many of these radiation monitors are Type F variables, in accordance with IEEE 497-2016. Type F variables provide primary information to accident management personnel to indicate fuel damage and the effects of fuel damage.

The staff notes that these other monitors are not analyzed for surviving core damage conditions. However, doses in these areas would not be as high as under the bioshield and the various process, effluent, area, and airborne radiation monitors throughout the facility should be able to provide additional information on radiological conditions, if the under-the-bioshield monitors were unavailable. Based on this, the staff noted that there should be a means of determining radiological conditions and identifying radiation releases for longer than 48 hours, if necessary, such as if needed for potential emergency planning purposes (while the staff considered this information for potential impacts on emergency planning, a COL applicant will be responsible for determining the details of how emergency planning requirements will be met). Chapter 19 of this SER discusses the staff review of the survivability of equipment following an accident that involves core damage.

The FSAR states that the NuScale design complies with the requirements of 10 CFR 50.68(b) in lieu of 10 CFR 70.24, as allowed by 10 CFR 70.24(d)(1), with respect to criticality accident monitoring. FSAR Section 9.1.2.3.5, “Monitoring,” states that radiation monitors are provided in the SFP area to detect both general area radiation levels and airborne contamination levels as described in FSAR Section 12.3. In addition, a local area radiation monitor is mounted on the refueling bridge with a local and MCR alarm function that monitors refueling activities. In FSAR Section 12.3.4.1, “Design Bases,” the applicant stated that the radiation monitors provided in fuel storage and handling areas when fuel is present are intended to detect excessive radiation levels and to initiate appropriate safety actions if excessive radiation levels occur, consistent with the requirements of 10 CFR 50.68(b)(6).

FSAR Table 12.3-10 shows that there is a reactor pool gamma radiation monitor, a refueling bridge monitor, a module maintenance center monitor, as well as 12 radiation monitors inside the bioshield (two for each module). FSAR Section 12.3.4.2 states that the placement of fixed area radiation monitors conforms to the criteria for selection and placement of the area radiation monitoring instrumentation in ANSI/ANS HPSSC-6.8.1-1981. Section 4.2.3 of that standard requires detectors to be located such that inadvertent shielding by structural materials is minimized. Based on the information in the FSAR about the purpose and location of radiation monitors, as described above, and on the commitment to ANSI/ANS HPSSC-6.8.1-1981 with regard to the placement of the radiation monitors, the NRC staff concludes that the design is consistent with the requirements of 10 CFR 50.68(b)(6) for monitoring radiation levels in areas where fuel is handled or stored, including during its transit from the NPM bay to the refueling area, and the design is therefore acceptable.

FSAR Section 12.3.4.1 states that the radiological monitoring equipment is designed to monitor plant area and airborne radiation levels for use in the emergency response data system (ERDS), consistent with the requirements of 10 CFR Part 50, Appendix E, Section VI.2(a). FSAR Section 12.3.4.2 states that fixed area radiation monitoring data are capable of being supplied to the NRC Operations Center through the ERDS through a secure direct electronic data link in the event of an emergency. FSAR Section 12.3.4.3, “Airborne Radioactivity Monitoring Instrumentation,” states that fixed continuous airborne monitoring data can be supplied to the NRC Operations Center through the ERDS through a secure direct electronic data link in an emergency and that FSAR Section 7.2, “System Features,” discusses the ERDS connection.

The emergency plan will identify and describe the specific instruments, including radiation monitors, that will be used to satisfy the requirements of 10 CFR Part 50, Appendix E, Section VI.2(a)(i). Section I of Appendix E to 10 CFR Part 50 states that each applicant for a COL under 10 CFR Part 52, Subpart C, “Combined Licenses,” is required by 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” to include in the application plans for coping with emergencies. The NRC staff finds this acceptable because the COL applicant will describe the comprehensive emergency plan, including radiation monitors that will be relied upon in the emergency plan.

Area radiation monitors are also located in the control room and technical support center, and there is an airborne radiation monitor in the control room. Other monitor locations include areas that could have significant changes in radiation dose or airborne radioactivity or in areas where it is important to know the radiological conditions, such as area and airborne radiation monitors in the hot lab, an area monitor in the safety instrument room, an area monitor in the primary sampling area, airborne radiation monitors in the degasifier room, and in numerous other areas of the RXB and RWB near radiation sources, access areas, and other areas where identifying the radiological conditions is warranted or useful. The monitors will alarm locally and in the MCR if high radiation levels are detected.

DSRS Section 12.3–12.4 states that GDC 14 and GDC 30 are acceptance criteria for this SER section as they relate to the ability to detect RCS pressure boundary leakage with radiation detectors. The area and airborne radiation monitors discussed in Chapter 12 of the FSAR are not relied on to detect RCS pressure boundary leakage, but the containment evacuation system

(CES) process and effluent monitors discussed in FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System,” are relied on as a method for detecting and identifying RCS pressure boundary leakage. These process and effluent radiation monitors are discussed in the following paragraph as they relate to meeting the requirements of GDC 14 and GDC 30.

FSAR Section 5.2.5.1, “Leakage Detection and Monitoring,” discusses three methods for detecting and, to the extent practicable, identifying the source of leakage into the containment vessel. These three methods are containment vessel pressure monitoring, CES sample tank level change monitoring, and CES vacuum pump discharge process radiation monitoring. FSAR Chapter 5 discusses these three methods in more detail. Radiation monitors for the CES vacuum pump discharge process are also discussed in more detail in FSAR Section 11.5 and evaluated in Section 11.5 of this SER. FSAR Section 9.3.6, “Containment Evacuation System,” discusses the CES. The system establishes and maintains a vacuum in the containment vessel during operation by removing non-condensable gases in the containment vessel.

The CES radiation monitors discussed in FSAR Table 11.5-1, “Process and Effluent Radiation Monitoring Instrumentation Characteristics,” and Table 11.5-4, “Effluent and Process Monitoring Off Normal Radiation Conditions,” monitor radiation levels in the gas removed from the containment vessel and, depending on the radiation level in the gas, either filter or discharge the gas through the RXB HVAC system plant exhaust stack or transfer the gas to the gaseous radioactive waste system, which will result in more radioactive removal and decay before release. Since these radiation monitors monitor gas removed directly from the containment vessel, it is appropriate to use them as one of the methods for identifying RCS pressure boundary leakage. Therefore, their use is acceptable as one of the methods for meeting the requirements for detecting and identifying RCS leakage in accordance with GDC 30, and as it relates to ensuring that the reactor coolant pressure boundary is appropriately maintained, in accordance with GDC 14.

Based on the above, the staff finds that the area and airborne radiation monitoring is consistent with the guidance in DSRS Section 12.3-12.4 and thus acceptable and that the containment vacuum pump monitors are appropriate for use in monitoring and potentially identifying RCS leakage.

12.3.4.1.5 Minimization of Contamination

Under 10 CFR 20.1406, the NRC requires each applicant to describe in the application, in part, how facility design will minimize, to the extent practicable, contamination of the facility, contamination of the environment, and the generation of radioactive waste. The regulation also requires applicants to describe how facility design will facilitate decommissioning. RG 4.21 contains a basis acceptable to the NRC staff for complying with the requirements of 10 CFR 20.1406.

FSAR Section 12.3.6, “Minimization of Contamination and Radioactive Waste Generation,” describes a design philosophy of prevention and early detection of leaks such that occupational doses are maintained ALARA, contamination is minimized, and decommissioning is facilitated.

FSAR Section 12.3.6.1, “Facility Design Objectives for 10 CFR 20.1406,” describes four design objectives and two operational program objectives used by the applicant during the design phase and specified for use by COL applicants using the approved design. As described in FSAR Section 12.3.6.1, the design and operational measures address the following objectives:

- Objective 1—Minimize the potential for leaks and spills to prevent the spread of contamination
- Objective 2—Provide sufficient leak detection capability to support timely leak identification from appropriate SSC
- Objective 3—Reduce the likelihood of cross-contamination, the need for decontamination and waste generation
- Objective 4—Facilitate eventual decommissioning through design practices
- Objective 5—Operational and programmatic considerations
- Objective 6—Site Radiological Environmental Monitoring

Objectives 1 through 4 are associated with the facility design. Objectives 5 and 6 are operational and programmatic measures to be addressed by a COL applicant (COL Item 12.3-6 is associated with objectives 5 and 6 and is discussed below). The NRC staff determined that the general design features described by the applicant are in accordance with this design philosophy and demonstrate compliance with the requirements of 10 CFR 20.1406. These features include measures to minimize facility contamination and contamination of the environment and features to facilitate decommissioning. FSAR Table 12.3-11, “NuScale Power Plant Systems with Nuclear Regulatory Commission RG 4.21 Evaluation,” lists systems that were evaluated using the guidance in RG 4.21. FSAR Tables 12.3-12, “Regulatory Guide 4.21 Design Features for Auxiliary Boiler System,” through 12.3-40, “Regulatory Guide 4.21 Design Features for Utility Water System,” list many of the specific features in the NuScale design consistent with the guidance in RG 4.21 and the requirements in 10 CFR 20.1406. Much of the information in FSAR Tables 12.3-12 through 12.3-40 summarizes material found elsewhere in the FSAR.

In addition, the FSAR includes COL Item 12.3-6, which directs the COL applicant to develop a plantwide RG 4.21 program to address the operational and programmatic considerations and site radiological environmental monitoring aspects of the minimization of contamination program, as provided in objectives 5 and 6, in accordance with 10 CFR 20.1406 and the guidance in RG 4.21. This will ensure that the program will meet the requirements in 10 CFR 20.1406 for life-cycle minimization of contamination. It is acceptable for COL applicants to address the operational considerations as described in COL Item 12.3-6.

The NRC staff reviewed the design features of SSCs provided for minimization of contamination described in FSAR Tables 12.3-12 through 12.3-40, and supporting information found in the text of the Chapter 12 FSAR and in the sections of the FSAR that provide the details of the specific plant systems. These tables discuss the specific features in the NuScale design that show how the applicant addressed the guidance in RG 4.21 and the requirements in 10 CFR 20.1406. The staff also reviewed the layouts for these systems to verify the containment and control of radioactive materials. The following paragraphs offer a few specific examples of the minimization of contamination design features identified in the NuScale design that meet the requirements of 10 CFR 20.1406 and are consistent with the guidance of RG 4.21.

As indicated in FSAR Section 11.4.1.2, “Wet Solid Waste,” the SRST and PST include tank vent piping that terminates below a vent hood and directs air into the RWB ventilation system to prevent nongaseous radioactive material from entering and contaminating the ventilation system. The vent pipe has a screen to prevent solid contamination from escaping and contaminating the area. Liquid overflow would flow into a shielded cubicle lined with stainless steel to prevent radioactive contaminated liquids from escaping the cubicle in the event of a leak, failure, or tank overflow.

As discussed in FSAR Table 12.3-34, “Regulatory Guide 4.21 Design Features for Radioactive Waste Building,” the RWB equipment rooms are designed with curbs, sump pits, and stainless steel liners in tank cubicles. This table also states that stainless steel liners in solid and liquid radwaste tank rooms are a sufficient height to contain the failure of any single vessel or piece of equipment. In addition, structural surfaces with the potential for radioactive contamination are epoxy coated, stainless steel lined, or otherwise treated to minimize absorption of contaminants.

The FSAR Section 12.3.6.1.1, “Design Considerations to Minimize Leaks and Contamination - Objective 1,” and Section 12.3.6.1.4, “Design Considerations for Decommissioning - Objective 4,” specify that buried and embedded piping for piping containing radioactive material is minimized throughout the design, and if buried or embedded piping is used in piping that may contain radioactive material, double-wall piping is used.

The NRC staff also reviewed the design features of the pool leakage detection system (PLDS). The staff examined this area of the plant design based on the guidance in RG 4.21, which addresses leakage from SFPs. As part of the review, the staff also considered information contained in the “Liquid Radioactive Release Lessons Learned Task Force Final Report,” dated September 1, 2006 (ML062650312), related to releases from SFPs and the causes for those releases. The report notes that the potential exists for unplanned and unmonitored releases of radioactive liquids to migrate off site undetected, including those portions of SFPs that are not visible to operators. The task force identified leakage from SFPs as one of the main components resulting in ground water contamination. The report also describes an event in which the liner leakage detection system became clogged with boric acid precipitate.

As discussed in FSAR Section 9.1.3.2.4, “Pool Leakage Detection System,” the PLDS is provided to detect leakage from the UHS and the other connected pools, such as the SFP. The PLDS consists of floor and wall leakage channels, perimeter leakage channels, drainage lines, small pool leakage detection sumps, leakage test lines, and valves. The PLDS works in conjunction with the radioactive waste drain system to collect and quantify leakage from the UHS pool routed from the floor and wall leakage channels. The sumps in the radioactive waste drain system are monitored, and the LRWS treats any leakage. In addition to the information in FSAR Chapter 12, FSAR Section 9.1.3, “Pool Cooling and Cleanup System,” and especially Section 9.1.3.2.4 discuss the PLDS.

Based on its review, as discussed above, and the information provided by the applicant, the NRC staff concludes that there is reasonable assurance that the design features described by the applicant will ensure compliance with the design requirements of 10 CFR 20.1406 and are therefore acceptable.

12.3.4.2 Dose Assessment

This section provides information on the dose assessment for both normal operations, including refueling, and post-accident actions. The NRC staff reviewed FSAR Section 12.4 for completeness against the criteria in DSRS Section 12.3–12.4. The staff ensured that the applicant had either committed to following the guidance of the applicable RGs and NRC staff positions in DSRS Section 12.3–12.4 or had provided acceptable alternatives. In areas where the FSAR adheres to these RGs and NRC staff positions, the staff can conclude that the relevant requirements of 10 CFR Part 20 and other applicable regulations have been met. In addition, the NRC staff selectively compared, based on the radiological significance of the task, the applicant's dose assessment for specific functions and activities against the experience of operating PWRs. Radiation exposures to operating personnel shall not exceed the occupational dose limits specified in 10 CFR 20.1201, and doses shall be maintained ALARA in accordance with 10 CFR 20.1101.

The NRC staff reviewed the radiation protection design features, dose assessment, and minimization of contamination design considerations in FSAR Sections 12.3 and 12.4 and other related sections of the FSAR for consistency with the guidance in DSRS Section 12.3–12.4. The purpose of this review was to ensure that the applicant had either committed to following the guidance in the RGs and applicable NRC staff positions or had offered acceptable alternatives. In areas where the FSAR is consistent with the guidance in these RGs and NRC staff positions, the staff concludes that the relevant requirements of 10 CFR Part 20 and 10 CFR Part 50 have been met. The sections below present the staff's findings.

12.3.4.2.1 Post-Accident Sampling:

SDAA Part 7, Section 16, "10 CFR 50.34(f)(2)(viii) Post-Accident Sampling," includes an exemption request from 10 CFR 50.34(f)(2)(viii). Section 9.3.2 of this SER evaluates NuScale's request for an exemption to 10 CFR 50.34(f)(2)(viii). As discussed in Section 9.3.2 of this SER, as NuScale requested an exemption from 10 CFR 50.34(f)(2)(viii), the NRC staff did not assess the radiological dose consequences to a worker obtaining and analyzing RCS and containment atmosphere samples following an accident.

12.3.4.2.2 Post-Accident Vital Area Mission Dose:

The purpose of this section is to discuss potential vital missions that may be necessary outside the MCR and technical support center following a potential accident, including an accident that results in core damage, and the evaluated dose to workers performing such actions. The evaluation of the radiation dose from potential post-accident missions is required by 10 CFR 50.34(f)(2)(vii) and is discussed in NUREG-0737, item II.B.2. In power reactors, vital actions that require evaluation typically include, but are not limited to, actions associated with manually manipulating valves that provide emergency core cooling or support emergency core cooling functions, actions associated with the operation of post-accident emergency ventilation systems, actions associated with emergency diesel generators, and actions associated with post-accident sampling or radiation monitoring. FSAR Section 12.4.1.8, "Post-Accident Actions," states that there are no vital areas (other than the MCR and the technical support center) in the NuScale design. The NuScale design provides passive cooling, there are no safety-related emergency diesel generators, and neither NuScale nor the staff identified any manual actions required outside the MCR and technical support center during DBAs, including a potential core damage accident.

As discussed in Section 12.3.4.2.1 of this SER, NuScale has requested an exemption from post-accident sampling. The NRC staff notes that in the NuScale US600 design, combustible gas monitoring was required and the potential dose to workers performing actions associated with post-accident combustible gas monitoring was discussed in the SER for the NuScale US600. SDAA Part 7, Section 2, "10 CFR 50.44(c)(4) and 10 CFR 50.34(f)(2)(xvii) Combustible Gas Monitoring," includes a request for exemption from combustible gas monitoring requirements. Chapter 6 of this SER evaluates this exemption. As a result, this type of post-accident hydrogen and oxygen monitoring is not required in the US460 design; therefore, there are no such required vital actions in the US460 design. Other required monitoring does not require operating actions outside of the MCR.

Based on this review, the staff finds NuScale's conclusion that there are no vital missions outside the MCR and the technical support center to be acceptable in the US460 design. Therefore, no vital missions are required to be analyzed in accordance with 10 CFR 50.34(f)(2)(vii). Chapter 15 of this SER evaluates the dose to the MCR and technical support center.

12.3.4.2.3 Operations and Maintenance Exposure Estimates:

The NRC staff reviewed the applicant's estimates for radiation exposures to plant personnel who perform work activities involving normal operations, maintenance and inspections, refueling activities, and waste handling and whether the method of estimating those doses used the guidance in RG 8.19. RG 8.19 notes that the dose assessment process should establish an objective to develop a systematic process for considering and evaluating possible dose-reducing design changes and associated operating procedure changes as part of the comprehensive ongoing design review and should identify principal ALARA-related changes resulting from the dose assessment. To allow for the expected buildup of radioactive material on and in SSCs, the occupational dose assessment should be based on anticipated radiation conditions after at least 5 years of plant operation. Analysis of the elements of the dose estimate (e.g., radiation levels, task duration, and frequency), treated qualitatively, can be significantly valuable in making engineering judgments on design changes for ALARA purposes.

RG 8.19 states that plant experience, which is available from industry groups like EPRI, provides useful information for performing the dose assessment. Using data from operating experience is consistent with 10 CFR 52.137(a)(22), which requires applicants to demonstrate how the plant design incorporates operating experience insights. The staff considered NuScale's specific design features and relevant operating experience (e.g., EPRI TR-1015119, "Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction Final Report," issued November 2007) when performing its review. Although the staff cannot quantitatively assess the change in dose rates associated with the smaller NuScale design, it did qualitatively consider the implications of the relative size of the plant for the assumed dose rates.

To estimate the occupational radiation exposures for the NuScale facility, the applicant identified various work activities and work durations along with the expected significant (greater than 0.001 mSv/h (0.1 mrem/h)) radiation fields that would be encountered. The applicant discussed the various activities in Section 12.4 of the FSAR.

The NRC staff reviewed the dose estimates provided in FSAR Section 12.4 and compared them to what would be anticipated given the staff's operating experience with large light-water reactors. As stated in RG 8.19, an analysis of the elements of the man-rem estimate (e.g., radiation levels, task duration, and frequency), treated qualitatively, can be of significant value in making engineering judgments on design changes for ALARA purposes. An expected result of the dose assessment process described in the guidance is that various dose-reducing design changes and innovations will be incorporated into the design. Some of the dose estimates, such as refueling doses in FSAR Table 12.4-7, "Occupational Dose Estimates from Refueling Activities," are very low compared to those that would be expected in a large light-water reactor. This is due to the ability to perform many refueling activities remotely in the

US460 design. The NRC staff finds that the applicant followed the guidance of RG 8.19 in determining the occupational radiation exposure estimates to plant personnel. The staff notes that actual occupational doses may differ from those estimated on the basis of actual plant conditions. Based on the information above, the staff finds the proposed exposure estimates acceptable.

As stated in COL Item 12.4-1, a COL applicant that references the NuScale Power Plant US460 standard design will estimate doses to construction personnel from a co-located existing operating nuclear power plant. The staff finds this COL item to be acceptable because the dose from a co-located nuclear power plant would be site-specific.

12.3.5 Combined License Information Items

Table 12.3-1 lists COL information item numbers and descriptions related to radiation protection design features from FSAR Table 1.8-2. COL Items 12.3-1, 12.3-2, 12.3-6, and 12.4-1 were discussed previously in Sections 12.3.4.1.1, 12.3.4.1.5, and 12.3.4.2.3. COL Items 12.3-3 through 12.3-5 require a COL applicant to provide additional information on radiation protection related hardware and radiation protection processes and programs. These COL items provide assurance that the requirements of 10 CFR Part 20 will be met by a COL applicant.

Table 12.3-1 NuScale COL Information Items for FSAR Section 12.3

COL Item No.	Description	FSAR Section
12.3-1	An applicant that references the NuScale Power Plant US460 standard design will develop the administrative controls regarding access to high radiation areas per the guidance of Regulatory Guide 8.38.	12.3.1.3.1
12.3-2	An applicant that references the NuScale Power Plant US460 standard design will develop the administrative controls regarding access to very high radiation areas per the guidance of Regulatory Guide 8.38.	12.3.1.3.1
12.3-3	An applicant that references the NuScale Power Plant US460 standard design will specify personnel exposure monitoring hardware, specify contamination identification and removal hardware, and establish administrative controls and procedures to control access into and exiting the radiologically controlled area.	12.3.1.3.1
12.3-4	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs necessary for the implementation of 10 CFR 20.1501 related to conducting radiological surveys, maintaining proper records, calibration of equipment, and personnel dosimetry.	12.3.4.2
12.3-5	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs necessary for the use of portable airborne monitoring instrumentation, including accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.	12.3.4.4
12.3-6	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary	12.3.6.1.6

COL Item No.	Description	FSAR Section
	to demonstrate compliance with 10 CFR 20.1406, and the guidance of Regulatory Guide 4.21.	
12.4-1	An applicant that references the NuScale Power Plant US460 standard design will estimate doses to construction personnel from a co-located existing operating nuclear power plant.	12.4.1.9

Note: For COL Item 12.3-6, Objectives 5 and 6 are applicant-defined terms and are described in SER Section 12.3.4.1.5 regarding features for compliance with 10 CFR 20.1406 and 10 CFR 52.137(a)(6).

12.3.6 Conclusion

As described above, the NRC staff has reviewed the SDAA against the following requirements:

- 10 CFR Part 20 as it relates to limits on doses and ALARA requirements for occupational workers and members of the public from sources of radiation exposure
- 10 CFR 20.1406 and 10 CFR 52.137(a)(6), as they relate to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the generation of radioactive waste
- 10 CFR 50.34(f)(2)(vii); 10 CFR 50.34(f)(2)(xxvi), as it relates to minimizing leakage from systems outside of containment
- 10 CFR 50.49(e)(4)
- 10 CFR 52.137(a)(5)
- 10 CFR 52.137(a)(22)
- GDC 4, 19, and 61, as they relate to the information on radiation sources provided by the applicant
- 10 CFR 50.34(f)(2)(xvii), as it relates to radiation monitoring
- 10 CFR 50.68 and 10 CFR 70.24, as they relate to radiation monitoring where fuel is stored or handled
- GDC 63 and 10 CFR Part 50, Appendix E, as they relate to monitoring for excessive radiation levels in the facility
- GDC 14 and 30, as they relate to RCS pressure boundary radiation monitoring
- 10 CFR 52.47(b)(1), as it relates to the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that a facility that incorporates the SDA can be constructed and operated in conformity with the SDA, the provisions of the AEA, and NRC regulations

- 10 CFR 20.1406(a) and 10 CFR 52.137(a)(5), as they relate to the identification of sources of radioactive material that could lead to the contamination of the facility, contamination of the environment, or the generation of radioactive waste

Based on the above, the NRC staff concludes that with the COL items that address programmatic, procedural, and site-specific aspects that a COL applicant is to address, the NuScale US460 FSAR adequately addresses the requirements described above.

12.4 Dose Assessment

Section 12.3.4.2 of this SER documents the staff's review of this section of the FSAR.

12.5 Operational Radiation Protection Program

12.5.1 Introduction

The operational radiation protection program for a nuclear power facility ensures that exposures of plant personnel to radiation are controlled and minimized. The administration of the radiation protection program, the qualifications of the personnel responsible for conducting various aspects of the radiation protection program, and the procedures for handling and monitoring radioactive material are important components of the program. Adequate equipment, instrumentation, and facilities must also be provided for (1) performing radiation and contamination surveys, (2) monitoring and sampling in-plant airborne radioactivity, (3) monitoring area radiation, and (4) monitoring personnel. Procedures and methods of operation, including those used to ensure that occupational radiation exposure will be ALARA, must be in place. These procedures and methods include those used in normal operation, refueling, inservice inspections, handling of radioactive material, handling of spent fuel, routine maintenance, and sampling and calibration activities related to radiation safety.

12.5.2 Summary of Application

FSAR: The applicant has provided COL Item 12.5-1, which directs the COL applicant to develop the radiation protection program in accordance with 10 CFR 20.1101.

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

Technical Specifications: SDA Part 4, Section 5.7, addresses TS for the control of HRAs.

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

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

12.5.3 Regulatory Basis

The relevant requirements of NRC regulations for the operational radiation protection program and the associated acceptance criteria are incorporated by reference from DSRS Section 12.5, "Operational Radiation Protection Program" (ML15350A341). The guidance in DSRS Section 12.5 and the applicable regulatory requirements will be addressed by the staff during the review of a potential future COL application.

12.5.4 Technical Evaluation

NuScale FSAR Section 12.5 states that the COL applicant must provide the radiation protection program. FSAR Section 12.1 states that the COL applicant must provide the ALARA program. FSAR Section 12.3 states that the COL applicant must provide programs to minimize contamination of the facility. The review of these programs is beyond the scope of review conducted for an SDA.

As described in DSRS Section 12.5, the COL applicant is also responsible for providing the description of the operational program and proposed implementation milestones for the leakage control program required by 10 CFR 50.34(f)(2)(xxvi) and the ground water protection program and procedures required by 10 CFR 20.1406.

Regarding compliance with 10 CFR 50.34(f)(2)(xxvi), design requirements associated with leakage control for systems outside of containment that may contain accident source term, are provided in the specific SER sections that evaluate the systems, including in Chapter 5 and Chapter 9 of this SER. In addition to the design information provided in the FSAR, FSAR Section 9.3.2.1, “Design Basis,” includes COL Item 9.3-1, which states the following:

> An applicant that references the NuScale Power Plant US460 standard design will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. The leakage control program will include an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.

Therefore, since the design includes features to limit leakage from systems outside of containment and a COL applicant will provide a leakage control program, as described above, the staff finds that the applicant has adequately addressed 10 CFR 50.34(f)(2)(xxvi).

NuScale addresses the requirements of 10 CFR 20.1406 through a combination of the design features described in the FSAR and the programmatic considerations in COL Item 12.3-6 that will be addressed by a COL applicant referencing the US460 SDA, as discussed in Section 12.3 of this SER. These programmatic considerations include programs and procedures for the protection of ground water. Therefore, the staff concludes that the applicant has adequately addressed the requirements of 10 CFR 20.1406.

12.5.5 Combined License Information Items

Table 12.5-1 lists COL information item numbers and descriptions related to the operational radiation protection programs from FSAR Table 1.8-2.

Table 12.5-1 NuScale COL Information Items for FSAR Section 12.5

COL Item No.	Description	FSAR Section
12.5-1	An applicant that references the NuScale Power Plant US460 standard design will describe elements of the operational radiation protection program to ensure that occupational and public radiation exposures are as low as reasonably achievable in accordance with 10 CFR 20.1101.	12.5

In addition, in FSAR Chapter 13, NuScale provides COL items related to the radiation protection program including COL Item 13.4-1, which requires, in part, that a COL applicant will provide site specific information for the radiation protection program and COL Item 13.5-6, which requires that a COL applicant will describe the site-specific operating procedures including the plant radiation protection procedures. The staff will review the information required by these COL items for the radiation protection operating programs and procedures as part of a COL application referencing the US460 design.

12.5.6 Conclusion

The NRC staff does not review operational programs during the design phase; therefore, it is acceptable for COL applicants to address the operational considerations as described in the COL Item 12.5-1. The staff will determine compliance with the requirements of 10 CFR Part 20, 10 CFR 50.34(f)(2), and other regulations applicable to these areas during the review of a COL application that references the US460 SDA.