12 RADIATION PROTECTION

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 12, "Radiation Protection," of the Design Certification Application (DCA), Part 2, "Final Safety Analysis Report (FSAR) Tier 2 " Revision 1, submitted by NuScale Power, LLC (NuScale) (hereinafter referred to as the applicant).

This chapter of the FSAR Tier 2 provides information on facility and equipment design and programs used to meet the radiation protection requirements in Title 10 of the *Code of Federal Regulations* (CFR), Part 20, "Standards for Protection against Radiation"; 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

The NRC evaluated the information in FSAR Tier 2 Chapter 12 against the guidance in Chapter 12, "Radiation Protection," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition" (SRP); Section 12.2, "Radiation Sources," of the "Design-Specific Review Standard for NuScale SMR Design" (DSRS); DSRS Section 12.3–12.4, "Radiation Protection Design Features"; and DSRS Section 12.5, "Operational Radiation Protection Program." Compliance with these criteria provides assurance that doses to workers will be maintained within the occupational dose limits and public dose limits of 10 CFR Part 20. The occupational dose limits applicable to workers at NRC-licensed facilities restrict the sum of the external whole body dose (deep-dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body (i.e., deposited through injection, absorption, ingestion, or inhalation) to 50 milli-sieverts (mSv) (5 roentgen equivalent man (rem) per year with a provision to extend this dose to 100 mSv (10 rem) per year with a lifetime dose limit of 250 mSv (25 rem) resulting from planned special exposures. The total effective dose equivalent (TEDE) dose limits of 10 CFR Part 20 for individual members of the public from the licensed operation is limited to 1 mSv (0.1 rem) in a year. In addition, as part of maintaining compliance with 10 CFR Part 20, the applicant is required to demonstrate compliance with the U.S. Environmental Protection Agency's (EPA) generally applicable environmental radiation standards in 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

The SRP and DSRS acceptance criteria also provide the guidance for assuring that radiation doses resulting from exposure to radioactive sources both outside and inside the body can be maintained well within the limits of 10 CFR Part 20 and as low as is reasonably achievable (ALARA). The balancing of internal and external exposure necessary to ensure that the sum of the doses is ALARA is an operational concern. An applicant seeking a combined license (COL) must address these operational concerns and programmatic radiation protection concerns.

12.1 Assuring that Occupational Radiation Exposures are As Low As Is Reasonably Achievable

12.1.1 Introduction

ALARA means making every reasonable effort to maintain exposures to radiation as far as practicable below the dose limits of 10 CFR Part 20. This includes accounting for the state of technology and the economics of improvements in relation to benefits to the public health and safety. It also includes using procedures and engineering controls based on sound radiation protection principles.

In addition to providing radiation exposure limits for workers and members of the public, 10 CFR 20.1101, "Radiation Protection Programs," requires that, to the extent practical, procedures and engineering controls based on sound radiation protection principles be used to achieve occupational doses and doses to the public that are ALARA. In addition, 10 CFR 20.1704(a) requires that the respiratory protection program of the licensee is adequate to limit doses to individuals from intakes of airborne radioactive materials consistent with maintaining TEDE ALARA. Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," provides specific guidance and criteria on the design, construction, and operation of a nuclear power plant to meet this regulatory requirement. Programmatic and policy considerations associated with plant operations that are needed to ensure that radiation doses will be ALARA (as discussed in RG 8.8; RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable"; and RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants") are outside the scope of this design certification (DC). The applicant has identified a COL information items COL Item 12.1-1, and COL Item 12.5-1, (see SER Section 12.1.5) to ensure that COL applicants referencing the design will address these issues.

12.1.2 Summary of Application

FSAR Tier 1: There is no Tier 1 information related to radiation protection for this section.

FSAR Tier 2: The applicant has provided FSAR Tier 2, Section 12.1, "Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable," which is summarized, in part, as follows:

- Most nuclear plant worker occupational radiation exposure (ORE) results from the operation and maintenance of systems that contain radioactive material, radioactive waste handling, in-service inspection, refueling, abnormal operations, and decommissioning work activities. The design of the small modular reactor (SMR) addresses and includes these activities through the plant physical layout, selection of materials, shielding, and chemistry control.

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

- Examples of facility design 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 field run piping to the extent practicable. SER Section 12.3 provides a more detailed discussion of design features to 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 OREs are ALARA, as discussed later in this section.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no ITAAC associated with the review of FSAR Tier 2, Section 12.1.

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

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

12.1.3 Regulatory Basis

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

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

- 10 CFR 20.1101 and the definition of ALARA in 10 CFR 20.1003, “Definitions,” as they relate to those 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 DCs 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.

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

The guidance in SRP Section 12.1, “Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable,” 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 1.8, “Qualifications and Training for Nuclear Power Plant Personnel.”

- RG 1.33, “Quality Assurance Program Requirements (Operation).”
- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).”
- 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.”
- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” Revision 1, issued October 2018.

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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML0914906841).
- 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 (ADAMS Accession No. ML0932201780).
- NEI 08-08A, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination,” and the associated NRC SER (ADAMS Accession No. ML093220530).
- NUREG-1736, “Consolidated Guidance: 10 CFR Part 20—Standards for Protection against Radiation,” issued October 2001.
- SECY-04-0032, “Programmatic Information Needed for Approval of a Combined License Application without Inspections, Tests, Analyses, and Acceptance Criteria,” dated February 26, 2004.

12.1.4 Technical Evaluation

The NRC staff reviewed the information in FSAR Tier 2, Section 12.1, to assess adherence to the guidelines in RG 1.206, Revision 1, and the criteria in SRP Section 12.1 for the radiation protection aspects of the reactor design. Specifically, the NRC staff reviewed FSAR Tier 2, Section 12.1, to ensure that the applicant had either committed to adhere to the guidance of the RGs and NRC staff positions referenced in SRP Section 12.1 or had provided acceptable alternatives. As described below, the NRC staff finds that FSAR Tier 2, Section 12.1, conforms to the applicable guidance in these RGs and the applicable NRC staff positions except for the open and confirmatory items discussed below.

Policy Considerations

In FSAR Tier 2, Section 12.1.1, “Policy Considerations,” the applicant described the design, construction, and operational policies that have been implemented to ensure that ALARA considerations are factored into each state of the NuScale SMR design process. The applicant has committed to ensure that the NuScale SMR plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8. In particular, FSAR Tier 2, 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. Specifically, this program shall encompass the ALARA concept and include 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 DC review. To maintain doses to plant personnel ALARA, the applicant stated in FSAR Tier 2, Section 12.1.3, “Operational Considerations,” that the COL applicant will submit a description of the radiation protection program to provide reasonable assurance that ORE will be ALARA.

Compliance with 10 CFR Part 19 requires, in part, that workers who receive occupational exposure be kept informed of radioactive material and the associated radiation and receive instructions with the objective of minimizing exposures to radioactive materials. COL Item 12.1-1 requires the COL applicant to describe the operational radiation protection program, which includes elements necessary to demonstrate compliance with 10 CFR Part 19.

Design Considerations

The plant radiation protection design should ensure that individual doses and total person-rem doses to plant workers and to members of the public are ALARA and that individual doses are maintained within the limits of 10 CFR Part 20. FSAR Tier 2, Section 12.1.2, describes the objectives for the general design. These design considerations include, but are not limited to, the following:

- Shielding is provided between components.
- Labyrinth entrances are provided to reduce radiation streaming out of cubicle entrances.
- Shield wall penetrations are configured to prevent “line-of-sight” streaming.
- Pipe chases are used for pipes containing significant radioactive material.
- Radiation areas where station personnel spend substantial time are designed to the lowest practical dose rates.
- Curbing and sloped floors direct leakage to local drains or sumps to limit the spread of contamination from liquid systems.

- Tanks containing radioactive liquids are designed with sloped bottoms toward outlets and flushing or cleaning features.

- Spare connections on tanks and other components located in high-radiation areas are provided to allow for greater operational flexibility.

- Pumps are selected to minimize leakage and provide housing drains.

- Radiation sources are separated from occupied areas where practicable (e.g., pipes or ducts that contain high radioactive fluids do not pass through occupied areas).

- When permanent shielding is impractical, provisions for temporary shielding are provided.

- Instrumentation is designed, selected, and located with consideration for long service life, ease and low frequency of calibration, and low accumulation of crud.

- Provisions are included to permit the rapid manipulation of shielding and insulation for equipment that requires periodic inspection or service.

- Adequate space is provided for moveable or temporary shielding for sources.

- The means to control contamination and to facilitate decontamination of potentially contaminated areas is provided.

- Piping for "clean services" (e.g., station air, potable water, nitrogen) is located separately from piping for contaminated systems to avoid cross-contamination.

- Features that permit remote removal, installation, inspection, or servicing of radioactive components are provided.

- Design features, such as ventilation isolation or filtration and heating ventilation and air conditioning, are provided such that air flows from areas of lower radioactivity to areas of higher radioactivity to protect against airborne contamination.

Operational Considerations

Operational considerations on the implementation of a radiation protection program are outside the scope of the DC review. The applicant has stated that a COL applicant who references the NuScale SMR certified design will address operational and maintenance requirements while satisfying the guidance of RG 1.33, RG 1.8, RG 8.8, and RG 8.10. The applicant also listed other RGs that the COL applicant will need to address in its application; these RGs are discussed in the policy considerations section above. The NRC staff does not review programs during the design 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 radiation program, as discussed in SER Section 12.5.

12.1.5 Combined License Information Items

Table 12.1.3 lists the relevant COL information item numbers and descriptions from FSAR Tier 2, Table 12.1.3, that have the COL applicant describe the operational radiation protection program.

Table 12.1.3 NuScale COL Information Items for Section 12.1

COL Item No.	Description	FSAR Tier 2 Section
12.1-1	A COL applicant that references the NuScale Power Plant design certification will describe the operational program to maintain exposures to ionizing radiation as far below the dose limits as practical, 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 meet the criteria of SRP Section 12.1. These design features are intended to maintain individual doses and total person-rem doses within the limits of 10 CFR Part 20. Therefore, the NRC staff finds that that NuScale SMR design features, in conjunction with the radiation protection program elements and ALARA elements described in COL Items, provides reasonable assurance that the COL Licensee can build and operate the plant consistent with the requirements of 10 CFR Part 20. The COL applicant will address the operational considerations for the NuScale SMR. The NRC staff does not review programs during the design phase, therefore it is acceptable for COL applicants to address the operational considerations as described in the COL item applicable to this section. The NRC staff will determine compliance with the requirements of 10 CFR Part 20 in these areas during the COL review.

12.2 Radiation Sources

12.2.1 Introduction

The determination of projected radiation sources during normal operations, anticipated operational occurrences (AOOs), and accident conditions in the plant is used as the basis for designing the radiation protection program and for shield design calculations. This includes the defining of isotopic composition, location of sources of radiation in the plant, source strength, and 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

FSAR Tier 1: There is no Tier 1 information related to radiation protection for this section.

FSAR Tier 2: The applicant has described onsite radiation sources primarily in FSAR Tier 2, Section 12.2, “Radiation Protection,” which is summarized, in part, as follows:

- FSAR Tier 2, Section 12.2, discusses and identifies the sources of radiation that form the basis for the shielding design calculations, radiation zoning, and the performance of 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 cleanup system (PCUS); plant sampling systems (PSSs); 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, this section of the FSAR Tier 2 provides information on post-accident source terms in the NuScale design. FSAR Tier 2 Chapter 15 provides additional information on the post-accident source terms, and the accident source term methodology appears in NuScale Topical Report (TR)-0915-17565, “Accident Source Term Methodology Topical Report,” Revision 3. The post-accident source terms are used to evaluate the doses in the main control room (MCR); the doses to take post-accident radiation liquid and gas samples, including sample analysis; and the doses to equipment important to safety (i.e., FSAR Tier 2 Section 3.11 discusses the equipment qualification (EQ) program and the associated analysis, and SER Section 3.11 discusses the NRC staff’s evaluation).

ITAAC: There are no ITAAC associated with the review of FSAR Tier 2, 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 (see the discussion related to Request for Additional Information (RAI) 8759, Question 12.02-1, below).

Technical Reports:

- NuScale TR-1116-52065, “Effluent Release (GALE Replacement) Methodology and Results,” Revision 0.

- NuScale TR-0116-20781-P, “Fluence Calculation Methodology and Results,” Revision 0.

- NuScale TR-0915-17565, Revision 3.

12.2.3 Regulatory Basis

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, as it relates 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.

- General Design Criterion (GDC) 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, 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.

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

- GDC 4, "Environmental and Dynamic Effects Design Bases," which requires 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 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.

- GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to systems that may contain radioactive materials.

- 10 CFR 52.47(b)(1), which requires that a DCA contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria 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.

- 10 CFR 52.47(a)(5), as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits established in 10 CFR Part 20.

- 10 CFR 52.47(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, “Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries 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 structures, systems, and components (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 1.143, “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 ORE and the classification of structures that house radioactive waste systems based on potential exposure to site personnel.

- 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 1.183, as it relates to the assumptions used in evaluating the concentrations of radionuclides in containment and plant systems following a loss-of-coolant 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.

The applicant must modify the methods and data in ANSI/ANS Std. 18.1-1999 to reflect NuScale-specific design attributes, which were developed using relevant industry operating experience.

12.2.4 Technical Evaluation

The NRC staff reviewed the descriptions of the radiation sources given in FSAR Tier 2, Section 12.2, to assess the applicant’s conformance with the guidance in the criteria in DSRS Section 12.2. The specific areas of review include contained sources and airborne radioactive sources. The NRC staff uses the sources of radiation described in FSAR Tier 2, Section 12.2, to determine the kinds and quantities of radioactive material to be present during operation and resulting from accumulation in plant systems. The NRC staff uses the kinds and quantities of radioactive materials present to evaluate the design features described in FSAR Tier 2, Sections 3.11 and 12.3 that are provided to protect equipment, workers, and members of the public from the effects of radiation. This review process ensures that the design is compliant with the regulatory requirements described in SER Section 12.2.3.

12.2.4.1 Contained Sources

The NRC staff reviews FSAR Tier 2, Section 12.2, to verify that it accurately describes contained sources, including byproduct, source, and special nuclear materials, and radiologically significant sources. The NRC staff ensures that the application describes radiation sources generated during normal operations, AOOs, and accident conditions in the plant. In FSAR Tier 2, Section 12.2.1, “Contained Sources,” the applicant described the source terms used for determining the radiation shielding, facility design features, and zoning during normal full-power operation, including AOOs, and evolutions such as refueling. FSAR Tier 2, Section 12.2.1, describes all large contained sources of radiation that are used as the basis for designing the radiation protection program and completing shield design calculations. These sources include the reactor core, reactor coolant system (RCS), CVCS, liquid system, gaseous system, solid radioactive waste system (SRWS), and other miscellaneous sources. For each of these contained sources, the applicant provided either the source strength by energy group or the associated activity levels listed by isotope (in most cases, the source terms are provided by activity level).

The sources of radiation during normal full-power operation are direct core radiation, coolant activation processes, the leakage of fission products from defects in the fuel rod cladding, and the activation of the reactor coolant corrosion and erosion products. The applicant is currently using contained source terms based on a DBFFF of 0.066-percent as the basis for the radiation shielding design calculations, ventilation system design calculations, and personnel dose assessment.

The applicant used industry standard application packages, such as ORIGEN, to develop the core source terms. Applicant-specific analytical calculations were used to distribute the core source terms through the 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. The NRC staff conducted audits to review the applicant's implementation of these source terms.

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 technical specification limits for halogens (iodine (I)-131 dose equivalent) and noble gases (xenon-133 dose equivalent). However, FSAR Tier 2 Section 11.1, Revision 0, Table 11.1-2, "Parameters Used to Calculate Coolant Source Terms," showed that the DBFFF value was 0.028 percent (which was proposed as the basis for determining plant radiation shielding, zoning, ventilation design, and EQ dose calculations). FSAR Tier 2, Revision 0, stated that the shielding design basis for primary coolant source term is based on NuScale SMR-specific design inputs and a 0.028-percent DBFFF. Because the NuScale SMR FSAR Tier 2 submittal was not consistent with the acceptance criteria of DSRS Section 12.2 with respect to the DBFFF, the NRC staff issued RAI 8759, Question 12.02-01 (ML17116A011), to ask the applicant to explain why adopting a technical specification limit that bounds the newly proposed DBFFF is not warranted as discussed in the DSRS Section 12.2 acceptance criteria stated above. The applicant's response to RAI 8759, Question 12.02-1, dated June 19, 2017 (ADAMS Accession No. ML17170A366), stated that, based on its review of past SRPs, the 0.25-percent and 1.0-percent DBFFFs cited in the SRP regulatory documents have not changed since the 1978 editions. The applicant further stated that individual nuclear power plants in the 1970s experienced fuel failure greater than 0.25 percent. The applicant stated that the NRC selected 0.25 percent to conservatively encompass most expected operation fuel failure rates at that time. In a supplemental response dated March 21, 2018 (ADAMS Accession No. ML18080A162), NuScale revised the value for the DBFFF from 0.028-percent failed fuel fraction to 0.066-percent failed fuel fraction. As stated by the applicant, this supplemental response did not, nor was it intended to, capture all the changes to the FSAR Tier 2 resulting from this change in the DBFFF. Based on a need for further understanding of the impacts on the FSAR Tier 2 and associated underlying calculations, the NRC staff audited the DBFFF change and the associated RAI responses, as described in the "Audit Plan for the Phase II Regulatory Audit of the Design Basis Failed Fuel Fraction for NuScale Power, LLC Design Certification Application," dated September 6, 2018 (ADAMS Accession No. ML18243A296). Because of the DBFFF change and other changes, the applicant revised many of the source terms and radiation zones in FSAR Tier 2 Chapter 12. Many of the different RAI responses discussed throughout the remainder of this section include these changes. RAI responses that provide information related to source term revisions include those to RAI 9161, RAI 9256, RAI 9257, RAI 9264, RAI 9265, RAI 9270, RAI 9281, and RAI 9302. The NRC discussed the status of RAI 8759, Question 12.02-1, with the applicant as part of the Phase II regulatory audit of the DBFFF “Audit Summary for The Phase II Regulatory Audit of The Design Basis Failed Fuel Fraction for NuScale Power, LLC Design Certification Application,” (ADAMS Accession No. ML18348A966). The NRC staff is still waiting for resolution of these issues. Therefore, t**he NRC staff is tracking RAI 8759, Question 12.02-01, as an open item.**

FSAR Tier 2, Revision 0, Figure 12.3-2a, “Radioactive Waste Building Radiation Zone Map—71' Elevation,” shows the area provided for the storage of high-integrity containers (HICs) that contain radioactive material (i.e., the Class A, B, and C HIC storage area, Room 030-034 in FSAR Tier 2 Figure 1.2-28, “Radioactive Waste Building 71'-0" Elevation”) as Radiation Zone VII. FSAR Tier 2, Revision 0, Table 12.3-1, “Normal Operation Radiation Zone Designations,” shows that areas designated as Radiation Zone VII have dose rates greater than or equal to 500 radiation absorbed doses per hour (rad/h) (5 grays per hour (Gy/h), with no upper limit specified. FSAR Tier 2, Revision 0, Section 12.2.1.7, “Solid Radioactive Waste System,” states that Table 12.2-18, “Solid Radioactive Waste System Component Source Term Inputs,” lists the assumed values used to develop the SRWS source terms. FSAR Tier 2 Section 12.2.1.7 also states that Table 12.2-19, “Solid Radioactive Waste System Component Source Terms—Radionuclide Content,” lists the radionuclide inventory of the major SRWS components and that Table 12.2-20, “Solid Radioactive Waste System Component Source Terms—Source Strengths,” lists the SRWS component source strengths. However, FSAR Tier 2 Table 12.2-18, Table 12.2-19, and Table 12.2-20 do not mention the Class A, B, and C HIC storage area. Therefore, on June 6, 2017, the NRC staff issued RAI 8860, Question 12.02-02 (ML17157B493), asking the applicant to describe the radionuclide content of the material expected to be stored in the Class A, B, and C HIC storage area; explain how it arrived at the stated concentrations; describe the storage configuration of the packages; and clarify whether drums from the drum dryer facility were to be stored in the room. In its response to RAI 8860, Question 12.02-02, dated July 10, 2017 (ADAMS Accession No. ML17191B211), the applicant provided revised FSAR Tier 2 Table 12.2-18; Table 12.2-19; Table 12.2-20; and Section 12.2.1.7, “Solid Radioactive Waste System.” The essence of these proposed changes was to indicate that, for the shielding analysis, the determination of the contained sources assumed that five HICs, 5.83 feet high by 4.92 feet in diameter loaded with resin at the activity listed in FSAR Tier 2 Table 12.2-19, column “SRST Ci,” that has been decaying for 2 years, were stored in a single layer. In the discussion part of the response, the applicant stated that this was a conservative analysis because it assumed that all 12 modules were operating at the DBFFF for 2 years. Although the amount of resin represented 1 year of resin generation, the applicant stated that 1 year was sufficient time to arrange for offsite disposal of additional HICs. The RAI response discussion noted that the HICs were assumed to be located in the middle of the room in a single layer. The discussion further noted that operational plant programs are provided to control the amount of radioactive material in the Class A, B, and C HIC storage area to limit the radiation levels in the surrounding areas to be compliant with the designated radiation zones. Based on the NRC staff review, as discussed above, the NRC staff considers RAI 8860, Question 12.02-2 (ADAMS Accession No. ML17191B211), closed/unresolved. In a follow up RAI to RAI 8860, Question 12.02-2 (RAI 9269, Questions 12.02-15, 12.02-16, 12.02-17, and 12.02-18 ADAMS Accession No. ML18008A157the NRC staff asked the applicant to provide additional justification for the assumptions related to the source term and geometry of the material to be stored in the room. As part of its Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power

Design," (ADAMS Accession No. ML18124A182)), the NRC staff reviewed the calculations used to determine the radionuclide content of the demineralizer beds and subsequently the spent resin storage tank (SRST). The NRC staff identified errors that resulted in non-conservative underestimations of resin radioactivity and, therefore, the gamma source strength. The NRC staff noted that the amount of barium (Ba)-137m contained in the SRST was under estimated by three orders of magnitude. The NRC staff also noted that the amount of Ba-137m contained in the CVCS mixed bed and other CVCS system demineralizers was zero after 2 days of decay; however, because Ba-137m is in secular equilibrium with cesium (Cs)-137 and because Ba-137m has a 2.552-minute half-life, the amount of Ba-137m after 20 minutes should be within approximately 6 percent of the amount of Cs-137. The NRC staff concluded that because the principal gamma considered for shielding from the decay of Cs-137 actually originates from the subsequent decay of Ba-137m, the errors could result in higher than estimated dose rates and the need for additional shielding or the changing of radiation zone designations. Therefore, the NRC staff asked the applicant to revise the calculations as necessary. The applicant also stated in the response to RAI 8860 Question 12.2-02, (ADAMS Accession No. ML17191B211), that operational controls would be used to limit the amount of radioactive material allowed to be stored with the room. The NRC staff noted that the documents referenced by the applicant as part of the RAI response did not contain criteria for the types of controls alluded to by the applicant. Therefore, in RAI 9269 the NRC staff asked the applicant to provide additional information about the controls that the COL applicant is expected to implement (i.e., COL item).

The applicant provided responses to RAI 9269, Questions 12.02-15, 12.02-16, 12.02-17, and 12.02-18, dated August 22, 2018 (ADAMS Accession No. ML18234A508). In RAI 9269, Questions 12.02-15 and 12.02-16, the NRC staff sought to understand the source term and the configuration of the HICs within the HIC storage room. As part of the NRC staff's DBFFF Phase II audit "Audit Summary for The Phase II Regulatory Audit of The Design Basis Failed Fuel Fraction for NuScale Power, LLC Design Certification Application," (ADAMS Accession No. ML18348A966), the NRC staff reviewed the calculations used to determine the radionuclide content of the demineralizer beds and subsequently the SRST. Because the contents of the SRST are a major contributor to the radionuclide concentrations in the HIC, the NRC staff asked the applicant to provide additional information about the sources of radioactivity in the SRST. The applicant answered the NRC staff's questions by clarifying that it assumed that, during 2 years of operation, five HICs of waste (assuming the source term from the 2-year decay of the SRST inventory) and one drum of waste from the drum dryer are within the room for the dose analysis. In addition, the applicant confirmed that the current design does not allow for HIC stacking. In response to RAI 9269, Question 12.02-17 (ML18324A508), asking the applicant to revise the calculations used to determine the activity in the SRST and update the application, the applicant provided the updated source term for the SRST, phase separator tank (PST), and HICs. The response included revisions that updated the amount of Ba-137m contained in the source terms found in FSAR Tier 2 Table 12.2-19. In this response, the applicant also described changes to the radiation zone mapping to be consistent with the updated source terms provided. In RAI 9269, Question 12.02-18, the NRC staff requested details about the controls that the applicant would have in place to maintain the facility within the design basis and to include the necessary COL items to ensure that it is maintained. In response to this question, the applicant detailed the assumptions used to generate the five HICs of waste and discussed the COL items used to establish programs and procedures for the design (COL Items 12.1-1, 12.3-1, 12.3-2, 12.3-3, 12.5-1, 13.5-1, and 13.5-3).

Using the information in the responses to RAI 9269, Questions 12.02-15, 12.02-16, 12.02-17, and 12.02-18, the NRC staff performed a confirmatory calculation to verify the reported radiation zoning using the assumptions described in the responses. Although the applicant's response stated that HIC stacking was not allowed, the NRC staff performed open item calculations to assume that the HIC storage room was filled with HICs and developed calculations to determine the effect of HIC stacking on the areas of the RWB in adjacent rooms, including the truck bay, which is located above the HIC storage room. The results of the NRC staff's analysis showed that the zoning designation reported by the applicant were reasonable. The NRC staff determined that, when compared to the assumptions made by NuScale, the NRC staff's analysis in the HIC filling the room with HICs was reasonably close. The NRC staff's analysis confirmed the radiation zoning listed in FSAR Tier 2 Figure 12.3-2a and Figure 12.3-2b, "Radioactive Waste Building Radiation Zone Map—100' Elevation," for the HIC storage room and adjacent areas. The response to RAI 9269 did not contain FSAR Tier 2 markup pages; however, information in other RAI responses, such as RAI 9280, Question 12.03-4; RAI 9264, Question 12.02-4; and RAI 9267, Question 12.02-7, in conjunction with the confirmatory calculations performed by the NRC staff, allowed the NRC staff to determine that the response to RAI 9269 is reasonable. Therefore, the NRC staff is tracking RAI 9269, Questions 12.02-15, 12.02-16, and 12.02-18, as confirmatory items, pending the incorporation of the proposed changes in a future FSAR, Tier 2 revision. RAI 9269, Question 12.02-17, is closed.

FSAR Tier 2 Table 11.1-4, "Primary Coolant Design Basis Source Term," lists the concentration of cobalt (Co)-60 as 1.0100E-04 micro-curies per gram (µCi/g) (1.01E-10 curies per gram). However, using the information cited above and the decontamination factor (DF) for the granular activated charcoal (GAC) units in FSAR Tier 2 Table 12.2-12, "Liquid Radioactive Waste System Component Source Term Inputs and Assumptions," the NRC staff was unable to derive a Co-60 activity at or below the 1.09E-01 curies (Ci) (1.09E+5 micro-curies) of Co-60 listed for the low-conductivity waste (LCW) GAC media in FSAR Tier 2 Table 12.2-13a, "Liquid Radioactive Waste System Component Source Terms—Radionuclide Content." Based on information made available to the NRC staff during its Radiation Protection and Accident Consequences (RPAC) Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff was not able to understand the methods, models, and assumptions that resulted in the noted difference. Therefore, the NRC staff issued RAI 9256, Question 12.02-09 (ADAMS Accession No. ML18008A099), asking the applicant to provide additional information about the methods, models, and assumptions used to calculate the amount of radioactive material in the LCW GAC bed. In its response to RAI 9256, Question 12.02-09, dated August 12, 2018 (ADAMS Accession No. ML18225A249), the applicant noted that the input process stream that enters the GAC filters is not direct primary coolant, as assumed by the NRC staff, and is composed of the outlet streams such as those from the liquid radioactive waste system (LRWS) LCW collection tank. This stream receives various waste streams, including outlet from the CVCS. The waste stream into the GAC is therefore partially cleaned and diluted by other processes. Because the explanation allowed the NRC staff to determine that the source term within LCW GAC is consistent with the use as described, the NRC staff finds the response acceptable. Therefore, the response to RAI 9256, Question 12.02-09, is closed.

FSAR Tier 2 Table 12.2-12 provides DFs for the liquid radioactive waste GAC unit. As stated in the FSAR Tier 2, the source for the stated DFs is International Atomic Energy Agency (IAEA)-TECDOC-1336, "Combined Methods for Liquid Radioactive Waste Treatment." The DFs referenced were discussed in the paper, "The Volume Reduction of Liquid Radioactive Waste by Combined Treatment Methods," referenced in IAEA-TECDOC-1336. Based on information in the FSAR Tier 2 and information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No.

ML18124A182)), the NRC staff was not able to understand the bases for the selection for the stated DFs. Therefore, in RAI 9256, Question 12.02-10 (ADAMS Accession No. ML18008A099), the NRC staff asked the applicant to provide additional information about the GAC bed DF values used and relevant input stream process variables that could affect the assumed DF values. In its response to RAI 9256, Question 12.02-10, dated August 13, 2018 (ADAMS Accession No. ML18225A249), the applicant noted that the GAC filter is not designed or intended to collect radionuclides; however, because it could collect radionuclides, it was conservatively assumed to do so, as opposed to assuming that the GAC filter did not collect radionuclides. In the absence of regulatory guidance for DFs for GAC, the applicant located the IAEA document (IAEA-TECDOC-1336) that contained DFs for a GAC filter. In the RAI response, the applicant stated that compared to the assumption of no retention by the GAC filter, these DF values are conservative for estimating the amount of radioactive material retained in the GAC filter. However, to ensure that the amount of radioactive material sent downstream is not underestimated, the radionuclide concentration in the outlet stream from the GAC filter is 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 effectively double counts these radionuclides to ensure the radiation zones and shielding requirements of the downstream components are also conservatively established. The NRC staff agrees that the methodology, as described by the applicant, provides reasonable methods for estimating the local dose rates and shielding requirements, while also providing reasonable process for estimating potential activity accumulation in downstream components. Therefore, the response is acceptable. **The NRC staff is tracking RAI 9256, Question 12.02-10, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9256, Question 12.02-10 in the same RAI response document. Specifically, the applicant modified information in FSAR Tier 2 Table 11.2-13; Table 12.2-13a; Table 12.2-13b, “Liquid Radioactive Waste System Component Source Terms—Radionuclide Content”; Table 12.2-14a; and Table 12.2-14b, which included changes not relevant to the GAC filter. These changes resulted from changes to the DBFFF, changes in crud burst assumptions, and other changes. These changes are not related to the review and tracking of RAI 8759, RAI 9161, and RAI 9257.

FSAR Tier 2, Revision 0, Section 12.2.1.3, “Chemical and Volume Control System,” states that, at the end of the fuel cycle, a crud burst is assumed, and the mixed-bed demineralizers are loaded with the entire radionuclide inventory increase resulting from the crud burst. FSAR Tier 2 Table 12.2-6, “Chemical and Volume Control System Component Source Term Inputs and Assumptions,” lists the resulting crud burst peaking factors. Electric Power Research Institute (EPRI) TR-3002000505, “Pressurized Water Reactor Primary Water Chemistry Guidelines,” Volumes 1 and 2, Revision 7, issued April 2014, 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 (AOA)), and crud-induced fuel failures. The term “high duty cores” is frequently used to differentiate some PWRs with more aggressive core designs from other PWRs, which can give rise to enhanced corrosion product deposition in the core. The high duty core index (HDCI) methodology was developed and incorporated into EPRI Technical Report (TR)-1008102, “PWR Axial Offset Anomaly (AOA) Guidelines,” Revision 1. Corrosion and wear products that spend longer periods in high neutron fluxes, such as material deposited on fuel surfaces, will have much higher specific radioactivity. Using the methodology described in Appendix F, “Definition of High Duty Core (A Means for Evaluating the Propensity to Deposit Crud on Fuel Assemblies),” to EPRI TR-1008102, the NRC staff determined that, based on the core power density, heat flux, and coolant flow rates described in the relevant sections of the FSAR Tier 2, the NuScale plant could also be classified as a high duty plant.

Based on information known to the staff at the time, the crud burst peaking factors listed in FSAR Tier 2 Table 12.2-6 appeared to be based on operational data from plants without an HDCI and as a result may understate the estimated crud burst. Therefore, the NRC staff issued RAI 9257, Question 12.02-14 (ADAMS Accession No. ML18008A156), asking the applicant to provide additional information related to the potential impact on the assumed radiation source terms from operating with an HDCI core and any compensatory design features provided. In its response to RAI 9257, Question 12.02-14, dated August 8, 2018 (ADAMS Accession No. ML18220B407), the applicant indicated that, during the development of the NuScale design and license application, radionuclide activity from an assumed end-of-cycle crud burst was included in addition to the crud radionuclide activity assumed using the methodology presented in ANSI/ANS Std. 18.1-1999. In addition, the applicant indicated that it developed the crud burst model using relevant industry operating information from EPRI TR-1011106, "Proceedings of the June 2004 EPRI PWR Shutdown Workshop." This EPRI report uses data from several large PWRs, including some with HDCIs, from which NuScale selected the highest reported values on which to base its model. However, in the later part of the response, NuScale removed the additional radionuclide activity from an assumed crud burst transient condition from the CVCS design-basis evaluation stating that NuScale believed it to be unnecessarily conservative. Instead, NuScale proposed using the guidance of ANSI/ANS Std. 18.1-1999 for crud isotopes. The applicant proposed updating the corresponding FSAR sections to incorporate this change. The NRC staff agrees that, based on the explanation in the first part of the RAI response, the original crud burst factors adequately addressed the NRC staff's concerns about the HDCI-induced crud burst. However, because of the applicant's subsequent deletion of the use of the crud burst factors, the RAI response does not address the NRC staff's concerns about crud bursts as explained below. Based on its review of NUREG/CR-1992, "In-Plant Source Term Measurements at Four PWR's," to assess the applicability of ANSI/ANS Std. 18.1-1999 for evaluating the contributions from crud bursts on coolant radioactivity concentrations and the subsequent accumulation of radioactive material in systems, the NRC staff disagrees with the applicant's assertion that ANSI/ANS Std. 18.1-1999 already addresses the accumulation of activity resulting from crud bursts in plant systems. Therefore, the applicant's response does not consider a crud burst and does not consider the peak activity that may routinely be present within the system components (e.g., the CVCS demineralizers and waste purification media) and the resultant dose rate. As a result, the NRC staff considers RAI 9257, Question 12.02-14, as closed/unresolved. On October 25, 2018, the NRC staff held a public meeting with the applicant (ADAMS Accession No. ML18327A104), to discuss the applicant's response to RAI 9257, Question 12.02-14 (ADAMS Accession No. ML18220B407). During the public meeting, the NRC staff clarified that ANSI/18.1-1999 does not include activity accumulation as a result of crud burst clean up. The NRC staff stated that the applicant needed to identify the kinds and quantities of radioactive material involved (consistent with 10 CFR 52.47(a)(5)). Once the source term is developed, then the applicant can design the radiation shielding, and the radiation protection program elements necessary to protect equipment, and to maintain worker exposure within the limits of 10 CFR Part 20. Therefore, in RAI 9607 Question 34 (ADAMS Accession No. ML19016A269), the staff asked the applicant to identify the kinds and quantities of radioactive material to be contained within the CVCS Mixed Bed Demineralizer following crud burst clean up, and to identify how the COL Licensee is to apply the information contained in FSAR Tier 2 Tables 12.2-7 and 12.2-8. **The NRC staff is tracking RAI 9607, Question 12.02-34 as an open item**.

FSAR Tier 2, Revision 0, Section 12.2.1.3, states that, for the specific activity content of the resin transfer pipe, a lower density value for the resin than that stated in the SRST is assumed. Based on operating experience at commercial nuclear power plants (see Report No. DE2004826015, "Studsvik Processing Facility Update," issued 2003, available at https://ntrl.ntis.gov/NTRL/), a relatively high ratio of water to resin is required to ensure unobstructed flow of resin slurries through the resin transfer pipes. However, based on that same operational experience, despite efforts by plant personnel, resin transfer lines do become clogged. Therefore, the NRC staff issued RAI 9258, Question 12.02-29 (ADAMS Accession No. ML18028A009), asking the applicant to provide additional information related to the assumed density in the resin transfer line.

In its response to RAI 9258, Question 12.02-29, dated August 13, 2018 (ADAMS Accession No. ML18225A285), the applicant revised its source term for the resin transfer pipe. The revised source term assumes that CVCS mixed bed demineralizer resins decayed for 48 hours. The applicant indicated that the CVCS mixed bed demineralizer resin was more limiting than the SRST. The NRC staff agrees that the CVCS mixed bed demineralizer source term would be expected to be the highest radioactivity concentration of resin because it removes radioactivity directly let down from the RCS. The NRC staff finds the information and the proposed FSAR Tier 2 changes in the response to be acceptable. However, the response indicates that the applicant's response to RAI 9257, Question 12.02-14, included a revised source term for the resin transfer pipe. As indicated above, the NRC staff has issued follow-up RAI 9607, Question 12.02-34 requesting the applicant to provide supplemental response with additional justification on the source terms provided in the response to RAI 9257, Question 12.02-14, and the appropriate accounting of crud burst activity. Therefore, because the source term is being evaluated, **the NRC staff is currently tracking RAI 9258, Question 12.02-29, as an open item** until it has confirmed that crud burst activities and other changes made to the resin transfer pipe source term are acceptable.

In its response to RAI-9258 Question 12.02-29, dated (ADAMS Accession No. ML18225A285), the applicant stated that that it modeled the size of the resin transfer line using the parameters described in FSAR Tier 2 Table 12.2-6. This table lists the resin transfer line as a 2-inch inside-diameter pipe with a 0.154-inch-thick wall (i.e., 2-inch schedule 40 pipe). During the audit of the DBFFF change and the associated RAI responses, as described in the DBFFF audit report (ADAMS Accession No. ML18348A966), the NRC staff noticed that the "RXB Dose Rates and Shielding Calculations" package specified the use of a 3-inch schedule 40 pipe as the basis for the resin transfer line. During the audit, the NRC staff noted that, because of the relative location of the CVCS mixed bed demineralizers and the PCUS demineralizers, the NRC staff had a reasonable expectation that these two lines would join in the RXB before going to the SRST in the RWB. The applicant stated that it was unable to determine the location of that juncture because it had not yet designed the resin transfer lines. While trying to ascertain the potential impact of having a transfer line clogged with resin, the NRC staff looked at the radiation zone designations in FSAR Tier 2 Figure 12.3-1a, "Reactor Building Radiation Zone Map—24' Elevation"; Figure 12.3-1b, "Reactor Building Radiation Zone Map—35'-8" Elevation"; and Figure 12.3-1c, "Reactor Building Radiation Zone Map—50' Elevation," provided in response to RAI-9281 (ADAMS Accession No. ML18235A654) and in FSAR Tier 2 Table 12.3-1. The NRC staff noted in RAI-9258 (ML18028A009), dated January 29, 2018 (see the applicant's response at ADAMS Accession No. ML18225A285), that, based on operating experience at commercial nuclear power plants (see Report No. DE2004826015), despite efforts by plant personnel, resin transfer lines do become clogged. The NRC staff's confirmatory calculations show that the dose rates from a 3-inch schedule 40 pipe filled with resin that contains CVCS mixed bed source terms would be 60 percent higher than that from a 2-inch schedule 40 pipe filled with resin at the same concentration. Based on the location of the CVCS demineralizers, and the absence of any horizontal shielded pipe chases described in FSAR Tier 2 Table 12.3-6, "Reactor Building Shield Wall Geometry," there does not appear to be a shielded path from the location of the demineralizers to the point in the RXB wall that is common to the RWB. RAI-9258 Question 12.02-29 is being used by the NRC staff to address resin transfer line source term concerns. **The NRC staff is tracking RAI-9258 Question 12.02-29, as an open item.**

Based on the NRC staff assessment, the methodology used by the applicant to develop the photon source strength from the post-accident fluid does not account for some principal photon radiation-emitting isotopes in the fluid stream. For example, Ba-137m is in secular equilibrium with the parent Cs-137 radionuclide; the specific activity of Ba-137m should be within 94 percent of the Cs-137 specific activity within 20 minutes. However, FSAR Tier 2 Table 12.2-7, "Chemical and Volume Control System Component Source Terms—Radionuclide Content," shows no Ba-137m activity in the resin transfer line. Because the photon source strength that the applicant used to perform the analysis of dose resulting from the sample fluid apparently did not properly account for Ba-137m, the analysis underestimated the dose rate from the sample fluid. Therefore, the NRC staff issued RAI 9258, Question 12.02-30 (ADAMS Accession No. ML18028A009), asking the applicant to provide additional information on the assumed radiation emitted from the resin. In its response to RAI 9258, Question 12.02-30, dated August 13, 2018 (ADAMS Accession No. ML18225A285), the applicant indicated that it updated the source terms for the resin transfer line in Tables 12.2-7 and 12.2-8 in its response to RAI 9257 to properly account for Ba-137m in secular equilibrium with Cs-137. The NRC staff's review of the Cs-137 to Ba-137m ratios in the applicant's response to RAI 9257 and the information related to the Cs-137 to Ba-137m ratios in the documents reviewed by the NRC staff during the DBFFF audit (ADAMS Accession No. ML18243A296) support the applicant's assertion that it now correctly accounts for Ba-137m. Because the applicant has provided information that demonstrates the applicant's compliance with the requirement of 10 CFR 52.47(a)(5) to identify the kinds and quantities of radiation present in the plant**, the NRC staff is currently tracking RAI 9258, Question 12.02-30, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9258, Question 12.02-30 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.Cs-137 has a 30-year half-life. It decays to Ba-137m with a 2.5-minute half-life. Because Ba-137m is in secular equilibrium with the parent Cs-137 radionuclide, the specific activity of Ba-137m should be within 94 percent of the Cs-137 specific activity within 20 minutes. The significant 662-kiloelectronvolt photon associated with the decay of Cs-137 is actually emitted from the decay of Ba-137m; therefore, if Ba-137m is omitted in an analysis, the results would be a significant underestimation of the photon source strength and, therefore, the resultant dose rate. FSAR Tier 2, Revision 0, Table 12.2-7, did not include Ba-137m in the column listing the radionuclide content of the resin transfer line. In addition, based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff observed that several calculations did not properly account for the equilibrium condition between Cs-137 and Ba-137m. The radionuclide content of systems and components is used to determine the photon source strength, which, in turn, is used to determine dose rates. The calculated dose rates are then used to establish shielding requirements, radiation zones, doses to equipment, and doses to personnel during normal operations and AOOs and following accidents. The NRC staff's analysis confirmed that some of the photon source strength information in documents made available to the NRC staff during the RPAC Chapter 12 Phase I audit were underestimated. Therefore, the NRC staff issued RAI 9264, Question 12.02-04 (ADAMS Accession No. ML17356A001), asking the applicant to provide additional information on the treatment of parent-daughter radionuclide activities, the resulting photon emission rates, and the impact on radiation zone designations and shielding requirements within the plant. In its response to RAI 9264, Question 12.02-04, dated August 23, 2018 (ADAMS Accession No. ML18236A618), the applicant indicated that it revised the source term calculations to appropriately account for Ba-137m in secular equilibrium with Cs-137. The applicant also indicated that FSAR Tier 2 Table 3C-2; Table 3C-6, “Normal Operating Environmental Conditions”; and Table 3C-8, “Accident EQ Radiation Dose,” which provide EQ information, appropriately accounted for all decay chains. During the DBFFF audit, the NRC staff confirmed that NuScale was appropriately implementing parent-daughter isotopic mixes. **The NRC staff is currently tracking RAI 9264, Question 12.02-04, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9264, Question 12.02-04 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.

FSAR Tier 2, Revision 0, Table 11.1-2, lists the value of the RCS mass as 117,400 pound-mass. FSAR Tier 2 Section 12.2.1, “Contained Sources,” states that the contained radiation sources are developed for normal operation and shutdown conditions and are based on the design-basis primary coolant activity concentrations from FSAR Tier 2 Section 11.1. Based on the review of material made available to the NRC staff during the Chapter 12 audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the NRC staff noticed that the different calculation packages that it had reviewed often used different values for the amount of water in the RCS. In several cases, the NRC staff observed that the changes made by the applicant were not in a conservative direction for the particular analysis and that the FSAR Tier 2 did not identify the application of the change. Therefore, the NRC staff issued RAI 9265, Question 12.02-05 (ADAMS Accession No. ML17356A002), asking the applicant to provide additional information on the treatment of the RCS mass in different analyses and the impact on the determination of source terms in components, radiation zone designations, and shielding requirements within the plant. In its response to RAI 9265, Question 12.02-5, dated July 11, 2018 (ADAMS Accession No. ML18192C178), the applicant provided proposed revisions to FSAR Tier 2 Table 11.1-2 and Table 11.1-3 that changed the mass values for the RCS. The applicant also stated that it used this revised mass value in the revised source term calculations. The NuScale calculation packages reviewed by the NRC staff confirmed that the applicant was using the revised RCS mass value, which the NRC staff considers to be an appropriate and reasonable value given the information contained in the RAI response. **The NRC staff is currently tracking RAI 9265, Question 12.02-05, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9265, Question 12.02-05 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.

FSAR Tier 2, Revision 0, Section 11.4.2.5.1, “Tanks,” states that NuScale provided two permanently installed SRSTs to receive spent resins from the CVCS and the PCUS demineralizers. FSAR Tier 2, Revision 0, Table 12.2-18, describes the dimensions of the SRST. FSAR Tier 2 Table 12.2-19 lists the radionuclide inventory of the SRST. FSAR Tier 2 Table 12.2-20 provides the SRST gamma emission rate in photons. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)) and information in the FSAR Tier 2, the NRC staff was unable to understand the basis for the stated concentrations of radioactive material in the SRST. Therefore, the NRC staff issued RAI 9267, Question 12.02-07 (ADAMS Accession No. ML17356A004), asking the applicant to provide additional information on the actual input streams to the SRST and volumes of material associated with each input stream.

The source size, magnitude, and configuration are elements of the model used to establish the effects of a contained source on areas adjacent to the contained source. The geometry of the source described in the FSAR Tier 2 did not appear to model the analytical method used to evaluate the radiation effects. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the model of the SRST contained in the analytical package reviewed by the NRC staff appeared to be significantly different from the model described in FSAR Tier 2 Table 12.2-18. The dimensions of the source material container affect the geometry of the radiation source and the subsequent dose rates. Therefore, the NRC staff issued RAI 9267, Question 12.02-08 (ADAMS Accession No. ML17356A004), asking the applicant to provide additional information on the analytical model used to describe the radiation effects from the SRST.

In its response to RAI 9267, Questions 12.02-07 and 12.02-08, dated February 1, 2018 (ADAMS Accession No. ML18032A763), the applicant provided additional information on the waste streams used as inputs into the analysis for determining SRST tank contents. In addition, the applicant updated the information in the FSAR Tier 2 tables to reflect the source geometry to be used in the NRC staff's confirmatory analysis. The NRC staff also noted that the NuScale analysis reflected a tank filled with 2 years of waste. The resulting SRST tank volume represented a tank that is roughly 42 percent full. In performing confirmatory calculations, the NRC staff assumed that the tank was filled to 80 percent, assuming some decay for the time it would take to fill the tank to roughly 80 percent. Based on its confirmatory calculation, the NRC staff determined that there was no dose impact in having a tank that is roughly 42 percent full versus the NRC staff's assumed 80-percent-full tank. This is based on the contributions of the short-lived radionuclides having decayed away in the additional 2 years it would take to fill the tank to 80 percent. Based on this confirmatory calculation and evaluation, the NRC staff has determined that the tank source terms described by the applicant appear to be reasonable. Because the zone designation, as specified by the applicant for the SRST, is greater than that determined by the NRC staff, the NRC staff finds this response acceptable. **The NRC staff is currently tracking RAI 9267, Questions 12.02-07 and 12.02-08, as confirmatory items**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9267, Questions 12.02-07 and 12.02-08 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.

FSAR Tier 2, Revision 0, Table 3C-1, "Environmental Qualification Zones—Reactor Building"; Table 3C-8; and Table 3C-6 describe the integrated dose in and around the NuScale module containment vessel (CNV) caused by normal operations and radiation exposure following an accident. FSAR Tier 2 Section 12.2.1.13, "Post-Accident Sources," states that the post-accident source remains within the CNV except for assumed leakage into the bioshield envelope, which is above the surface of the pool water and below the bioshield. It further states that four volumes are evaluated for the post-accident source term. FSAR Tier 2 Table 12.2-31, "Post-Accident Integrated Energy Deposition and Integrated Dose," lists the integrated post-accident source energy deposition versus time for both photons and electrons for these four volumes. FSAR Tier 2 Table 12.2-28, "Post-Accident Source Term Input Assumptions," lists some of the assumptions used in the analysis. FSAR Tier 2 Table 12.2-31 lists some time-integrated doses for periods following the accident. However, FSAR Tier 2, Revision 0, Section 12.2, does not contain a table that describes the airborne concentrations in the CNV or above the surface of the pool water and below the bioshield. Therefore, the NRC staff issued RAI 9261, Question 12.02-03 (ADAMS Accession No. ML17356A000), asking the applicant to provide additional information on the calculations used to determine the post-accident dose rates to account for sources of radiation other than just the gas cloud above the surface of the pool water and below the bioshield.

In its response to RAI 9261, Question 12.02-03 (ADAMS Accession No. ML18114A370), the applicant specified that it used an infinite cloud assumption for calculating the dose rates from gamma emissions in the bioshield area. This assumption results in the deposition in the volume of interest of all the gamma photon energy that is emitted. The applicant also considered shine from the containment atmosphere to the bioshield area. The applicant proposed updating FSAR Tier 2 Section 12.2.1.13 with this information.

In response to this question, the applicant also indicated that it planned to submit a revised accident source term NuScale TR-0915-17565, Revision 3, in late summer 2018. In addition, it changed the FSAR Tier 2 based on a revised source term. The NRC staff will evaluate this response in its entirety based on the review of the TR when it is submitted. **Therefore, the NRC staff is tracking RAI 9261, Question 12.02-03, as an open item**.

FSAR Tier 2, Revision 0, Table 12.2-29, "Post-Accident Core Inventory Release Fractions," lists the radionuclide groups and the associated release fractions. FSAR Tier 2 Table 12.2-30, "Post-Accident Containment Aerosol Removal Rates," describes how some radionuclides are removed from the containment atmosphere following an accident. FSAR Tier 2 Table 12.2-31 provides the integrated mega-electron-volt energy deposition and the integrated dose at specific time intervals during post-accident conditions. Because FSAR Tier 2, Revision 0, Section 12.2, does not list the isotopic inventory (i.e., isotope identification and concentration) during post-accident conditions, the NRC staff is unable to determine how the assumptions listed in Tables 12.2-28 and 12.2-29 have been applied. The post-accident isotopic concentrations in the CNV air volume liquid are used to determine the post-accident radiation levels in a variety of areas, including, but not limited to, areas above the RXB pool caused by shine from the air volume in the CNV, the area above the CNV but below the bioshield, and areas adjacent to the bioshield that are subject to radiation penetrating shielding or streaming through an opening. The post-accident isotopic concentrations in the RCS liquid are used to determine the post-accident radiation levels in a variety of areas, including, but not limited to, areas with pipes containing RCS liquids (e.g., the CVCS, PSS, and LRWS). Therefore, the NRC staff issued RAI 9268, Question 12.02-11 (ADAMS Accession No. ML18008A098), asking the applicant to provide additional information on the radionuclide concentrations inside the CNV and the reactor vessel (RV) following an accident. In its response to RAI 9268, Question 12.02-11 (ADAMS Accession No. ML18114A371), the applicant indicated that it planned to submit a revised accident source term NuScale TR-0915-17565, Revision 3, in late summer 2018. In addition, the applicant made changes to the FSAR Tier 2 based on a revised source term. The NRC staff will evaluate the applicant's response in its entirety based on the review of the TR when it is submitted. Therefore, **the NRC staff is currently tracking RAI 9268, Question 12.02-11, as an open item**.

FSAR Tier 2, Revision 1, Section 12.2.1.7, "Solid Radioactive Waste System," states that Table 12.2-18 lists the assumed values used to develop the SRWS source terms and the dimensions of the PSTs. FSAR Tier 2 Table 12.2-19 lists the radionuclide inventory of the

PSTs. FSAR Tier 2 Table 12.2-20 provides the PST gamma emission rate in photons per second. FSAR Tier 2 Section 12.2 does not appear to contain any other information about the amount of radioactive material that can be present in the PSTs and the sources of the material. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the volume of radioactive material in the PST occupies less than one-tenth of the height of the tank, as stated in the FSAR Tier 2. In addition, the model of the PST contained in the analytical package reviewed by the NRC staff appeared to be significantly different that the model described in FSAR Tier 2 Table 12.2-18. Therefore, the NRC staff issued RAI 9271, Question 12.02-21 (ADAMS Accession No. ML18008A286), asking the applicant to provide additional information on the methods, models, and assumptions used to determine the radionuclide content of the PSTs. In addition, the NRC staff issued RAI 9271, Question 12.02-22 (ADAMS Accession No. ML18008A286), asking the applicant to provide additional information on the parameters of the analytical model used to determine dose rates resulting from the radionuclide content of the PSTs.

In its response to RAI 9271, Questions 12.02-21 and 12.02-22, dated February 15, 2018 (ADAMS Accession No. ML18046A257), the applicant provided additional information on the waste streams used as inputs into the analysis for determining PST contents. In addition, the applicant updated the information in the FSAR Tier 2 tables to reflect the source geometry to be used in the NRC staff’s confirmatory analysis. The NRC staff also noted that the applicant’s analysis reflected a tank filled with waste resulting from 2 years of operation. The resulting waste volume in the PST represented a tank that was roughly 16 percent full. Confirmatory calculations performed by the NRC staff assumed that the tank was filled to 80 percent and assumed that the contents would decay during the time it would take to fill the tank to roughly 80 percent. Based on its confirmatory calculation, the NRC staff determined that there was a minimal dose impact in having a tank that is roughly 16 percent full versus the NRC staff’s assumed 80-percent-full tank. This is based on the contributions of the short-lived radionuclides having decayed away in the additional 8 to 10 years it would take to fill the tank to 80 percent. Based on this confirmatory calculation and evaluation, the NRC staff has determined that the tank source terms described by NuScale and the associated radiation zone designations for the rooms containing the PSTs are reasonable. Because the zone designation as specified by the applicant for the PSTs is greater than that determined by the NRC staff, the NRC staff finds this response acceptable. **The NRC staff is currently tracking RAI 9271, Questions 12.02-21 and 12.02-22, as confirmatory items**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9271, Questions 12.02-21 and 12.02-22 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.

FSAR Tier 2, Revision 0, Table 3C-6, states that the 60-year integrated gamma dose (rad) includes fission gammas, N-gammas, and coolant for the area outside of the CNV and under the bioshield. FSAR Tier 2 Section 12.2.1.2, “Reactor Coolant System,” states that FSAR Tier 2 Tables 12.2-3 and 12.2-4 provide the primary coolant gamma spectra. However, it contains no quantitative discussion of how these values are derived. FSAR Tier 2 Table 3C-6 provides the 60-year integrated N dose (rad) for the area outside of the CNV and under the bioshield. FSAR Tier 2 Sections 3.11 and 12.2 do not discuss the gamma dose rate from activation of the CNV steel, the stainless-steel lining of the bioshield cover, and the steel main steam and main feed water lines. Table 5-1, “Best Estimate of Fluence Expected to Be Experienced in Various NuScale Power Module Components and Locations,” of NuScale TR-0116-20781-P, Revision 0, describes the neutron fluence to the RV and CNV near the core but does not provide any neutron fluence information above the RV flange area. Based on the information available in the FSAR Tier 2, the NRC staff was not able to determine, with a reasonable degree of assurance, that the neutron and gamma dose rates provided by the applicant, for the areas under the bioshield were appropriate. The NRC staff needs to ascertain the gamma dose rates resulting from operation of the plant and to evaluate appropriate supporting information to assess the impact on a variety of review areas, including EQ, radiation streaming into adjacent areas, the amount of gamma radiation from neutron activation of materials, and operational radiation exposure for maintenance activities. Therefore, the NRC staff issued RAI 9291, Question 12.02-24 (ADAMS Accession No. ML18028A000), asking the applicant to provide additional information on the gamma-radiation field outside of the CNV above the pool water level and under the bioshield. In its response to RAI 9291, Question 12.02-24 (ADAMS Accession No. ML18058A720), the applicant stated that it determined doses in and around the power module using particle transport and performed shielding calculations for various source terms using Monte Carlo N-Particle Transport Code (MCNP), Version 6. Reactor coolant isotopes, including nitrogen (N) -16 and other water activation products were included. Isotopes like N-16 and other activation products are produced from neutron activation of water and other material suspended in the coolant water. Gamma photons are produced directly from fission, the decay of water activation products, the decay of fission products that have leaked from the fuel and into the coolant, and fission neutron interactions with structural materials. Changes to thermal hydraulic parameters affect the amount of short half-life radionuclides that are present in areas of the reactor vessel that can have a significant impact on the dose rates. Although RAI 9161, Question 11.01-1, did not request any information related to dose rates in this area, in its response to RAI 9161, Question 11.01-1 (ADAMS Accession No. ML18242A702) and the supplemental response to RAI 9161, Question 11.01-1 (ADAMS Accession No. ML18323A269), the applicant revised FSAR Tier 2 Tables 12.2-1, 12.2-3, and 12.2-4 to account for changes to the DBFFF and other changes. FSAR Tier 2 Table 12.2-4 provides the source term for coolant exiting the core and in the riser region, and FSAR Tier 2 Table 12.2-3 provides the source term for the remainder of the fluid in the RV, including the steam generator (SG) region and the pressurizer region. FSAR Tier 2 Table 12.2-1, “Core and Coolant Source Information,” provides assumptions and parameters for FSAR Tier 2 Tables 12.2-3 and 12.2-4. The applicant updated these tables based on changes in various assumptions, including changes in the DBFFF and an assumed increase in the RCS flowrate (discussed elsewhere), which increases N-16 concentrations in the cold leg and upper portions of the reactor coolant loop. As a result of these changes, the applicant explicitly included N-16 in the FSAR Tier 2 Table 12.2-3 and 12.2-4 source terms. In reviewing this updated information, the NRC staff performed a confirmatory analysis and estimated dose rates several orders of magnitude greater than what the applicant indicated in its response to RAI 9291, Question 12.02-24. In an October 23, 2018, public teleconference with the applicant, which was a continuation of an October 18, 2018, public meeting (see ADAMS Accession No. ML18318A298 for the meeting summary), the applicant indicated that the dose rate information in the response to RAI 9291, Question 12.02-24, and the associated FSAR Tier 2 changes had not been updated to address the revised FSAR Tier 2 Table 12.2-3 and 12.2-4 source terms. The applicant indicated that, although its response to RAI 9291, Question 12.02-24, indicated that the dose rate above the RV from the reactor coolant was 1.1 milli-rem per hour (mrem/h) (0.011 millisievert per hour (mSv/h), the dose rate was now approximately 1,600 mrem/h (16 mSv/h) using the revised source terms. As a result, the applicant agreed to update its response to RAI 9291, Question 12.02-24, to provide this updated information and the associated FSAR Tier 2 changes. In a November 28, 2018, supplemental response to RAI 9291, Question 12.02-24 (ADAMS Accession No. ML18332A397), the applicant provided information about the changes to the gamma dose rates, which also included changes to the neutron dose rates and other information. Following the receipt of the applicant's supplemental response, the NRC staff initiated an audit plan "Regulatory Audit Plan for the Audit of NuScale Documents Associated with EQ and Radiation Protection Neutron Source Term," (ADAMS Accession No. ML18348A986), to obtain additional information related to identifying, in the application, the type of radiation present; the quantity of radiation present; and the energy of that radiation, as it is used to determine the dose rate and total dose from the incident neutron and gamma radiation. RAI 9291 is specifically identified as one subjects of the ongoing audit. Therefore, **the NRC staff is tracking RAI 9291, Question 12.02-24 as an open item**.

In addition to revising FSAR Tier 2 Tables 12.2-1, 12.2-3, and 12.2-4 in its response to RAI 9161, Question 11.01-1, the applicant also revised FSAR Tier 2 Table 12.2-16, "Gaseous Radioactive Waste System Component Source Term Radionuclide Content," and FSAR Tier 2 Table 12.2-17, "Gaseous Radioactive Waste System Component Source Terms—Source Strengths." The revised tables provided a slightly lower source term than the original tables did. However, the applicant did not include any information in its response on why the source term would decrease when the DBFFF increased from 0.028 percent to 0.066 percent. Therefore, the NRC staff issued RAI 9621 Question 12.02-32 (ML17356A000), asking the applicant to provide information explaining the changes made. In addition, the NRC staff asked the applicant to describe how degasifying the reactor through the pressurizer in preparation for shutdown could impact the gaseous radioactive waste processing system source terms and to revise the source terms as appropriate. In a December 17, 2018, response to RAI 9621, Question 12.02-32 (ADAMS Accession No. ML18351A203), the applicant provided additional information related to the source term associated with degasification of the reactor coolant system for shutdown. However, the staff continues to have questions about the value of the assumed source term, and the methods, models, and assumptions used by the applicant to arrive at those source terms. This question is still under evaluation by the NRC staff. **Therefore the NRC staff is tracking RAI 9621, Question 12.02-32, as an open item**.

FSAR Tier 2, Revision 0, Section 11.4.2, "System Description," states that spent filters are removed from the filter housing and placed into an HIC. Once the HIC is full, it is dewatered, sealed, and stored for eventual offsite processing and disposal. FSAR Tier 2 Section 11.4.2.2, "Wet Solid Waste," indicates that this may include spent cartridge filters and filter membranes from the tubular ultrafiltration system and the reverse osmosis unit. FSAR Tier 2 Table 11.4-3, "Estimated Annual Volumes of Wet Solid Waste," states that some of these filters may be Waste Class B or C. U.S. Department of Energy (DOE) DOELLW-114F, "Greater-Than-Class C Low-Level Waste Characterization, Appendix F: Greater-Than-Class C Low-Level Radioactive Waste Light Water Reactor Projections," states that some individual PWR filters may have dose rates in the 50-rad/h (0.5-Gy/h) to 200-rad/h (2-Gy/h) range. The dose rates reported in DOELLW-114F were based on operating experience from commercial nuclear power plants. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)) and on information available in the FSAR Tier 2, the NRC staff was not able to determine the source term in the HIC filters. Therefore, the NRC staff issued RAI 9299, Question 12.02-25 (ADAMS Accession No. ML18028A001), asking the applicant to provide additional information on the radioactive material expected to be contained in the HIC filter media and the associated methods, models, and assumptions.

In its response to RAI 9299, Question 12.02-25, dated February 28, 2018 (ADAMS Accession No. ML18059B093), the applicant provided additional information on the radiological source terms used to estimate the contents of an HIC. The applicant's response referred to information in its responses to RAI 9267, Questions 12.02-07 and 12.02-08 (ADAMS Accession No. ML18046A257), in which the HIC source term and source dimensions were added to FSAR Tier 2 tables. In addition, the applicant's response stated that the source term estimate for the HIC was conservative because the applicant's analysis assumed that the contents of the HICs contain Class B and C waste resin from just the SRSTs that have been decaying for 2 years. The NRC staff found this response to be acceptable because the updated FSAR Tier 2 text allowed the NRC staff to perform a confirmatory analysis demonstrating that the applicant's approach was conservative. As a result, RAI 9299, Question 12.02-25, is closed.

In order to evaluate the effectiveness of post-accident radiation and shielding design, the NRC staff needs the post-accident source terms that may result in dose to plant operators from sources of radioactive material that may be present following an accident. The applicant has stated that NuScale TR-0915-17565, Revision 3, provides these source terms. The NRC staff has not yet received the revised TR describing the NuScale post-accident source term. The NRC staff will update the SER following its review of the TR.

12.2.4.2 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 and that 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, AOOs, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel and exposure to equipment important to safety. The description should also include models and parameters for the calculations.

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-by-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 applicant also assumed evaporation from the UHS pool to calculate airborne radioactivity concentrations in the UHS pool area airspace.

FSAR Tier 2, Revision 0, Table 12.2-32, "Input Parameters for Determining Facility Airborne Concentrations," describes assumptions for calculating airborne activity within the RXB. Transuranic (TRU) nuclides, such as americium, plutonium, and curium, are formed in irradiated uranium fuel by neutron activation and decay predominantly by alpha emission. These radionuclides are significant because of their presence in fluids in contact with reactor fuel and because alpha-emitting radionuclides have a significantly lower annual limit on intake (ALI) than beta-gamma emitting nuclides do (see Table 1, "List of Elements," in 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). FSAR Tier 2, Revision 0, Section 12.2.2, "Airborne Radioactive Material Sources," does not discuss the sources of airborne radioactivity within the RWB, and FSAR Tier 2 Table 12.2-32 does not list assumptions relevant to the determination of airborne activity concentrations in the RWB. In addition, based on information made available to the NRC staff during the Chapter 12 audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff could not identify how the applicant assessed the airborne activity within the RWB. The RWB has several components and active processes that contain significant quantities of radioactive material. Therefore, the NRC staff issued RAI 9255, Question 12.02-31 (ADAMS Accession No. ML18120A370), asking the applicant to provide additional information on the sources of airborne radioactive material within the RWB. In its response to RAI 9255, Question 12.02-31 (ADAMS Accession No. ML18129A415), the applicant did not provide any specific TRU activities but specified that the source terms in FSAR Section 12.2 are derived from the reactor core inventory and the resultant primary and secondary coolant source terms in FSAR Section 11.1 and from activated components. The applicant stated that the 10 CFR 52.47(a)(22) requirement to demonstrate how the design incorporates operating experience insights only applies to insights gained from generic letters and bulletins. The applicant indicated that there is no regulatory guidance for performing such a TRU airborne activity analysis that would include values for variables such as the TRU alpha-emitting radionuclide fuel escape rate coefficient, deposition rate on wetted surfaces, and airborne fraction when surfaces become dry. The response also stated that the radiation protection program, which will be developed as part of COL Item 12.5-1, will address alpha-emitting radionuclides. NuScale also indicated that it followed RG 8.25, "Air Sampling in the Workplace," as part of the criteria for the selection and placement of fixed continuous air monitors as specified in FSAR Section 12.3.4.3. The NRC staff disagrees with the applicant's basis outlined in its response to RAI 9255, Question 12.02-31, for not providing the requested information. However, based on the applicant's plant-specific design information and information about escape coefficients in other NRC guidance documents, the NRC staff was able to use information about the core isotopic inventory from NuScale's environmental report, "Applicants Environmental Report—Standard Design Certification (Rev. 1)—Section 1.0—Appendix B" (ADAMS Accession No. ML18086A070), to extrapolate potential airborne equilibrium values for major TRUs. This independent analysis found that the TRUs did not appear to be significant contributors to airborne radioactivity for the NuScale specific design; therefore, the NRC staff considers RAI 9255, Question 12.02-31, closed.

FSAR Tier 2, Revision 0, Section 12.2.1.8, "Reactor Pool Water," states that the radionuclide contribution resulting from neutron activation of the reactor pool water contents is not significant because of the reduced neutron flux in the reactor pool water. Based on the review of information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff determined that the applicant did not consider how the production of deuterium through the adsorption of a neutron by mononucleon hydrogen would increase the atomic abundance of deuterium in the UHS pool water over time. The increase in the atomic abundance of deuterium results in a change in the macroscopic cross-section of deuterium used to determine the amount of tritium produced by activation of water. The macroscopic cross-section of deuterium is based on the microscopic cross-section of deuterium (which does not change) and the relative atomic abundance of deuterium. Based on the NRC staff's review of information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff determined that the applicant used a fixed macroscopic cross-section of deuterium to calculate the production of tritium in the pool water. Because the methodology that the applicant used to calculate the tritium production rate in the UHS pool water does not account for the change in atomic abundance of deuterium in the UHS pool water over time, it underestimates the total production of tritium resulting from neutron activation of water. Because airborne tritium concentrations in the RXB are directly dependent on the UHS pool tritium concentration, the airborne tritium activity concentrations in the RXB may be underestimated. Therefore, in RAI 9259, Question 12.02-26 (ADAMS Accession No. ML18045A792), the NRC staff asked the applicant to provide information about how tritium production in the pool water varies over time as the concentration of deuterium increases in the pool water. In its response to RAI 9259, Question 12.02-26, dated February 14, 2018, (ADAMS Accession No. ML18045A792), the applicant stated that it had performed a study that evaluated the buildup of deuterium in the UHS pool. The study considered the tritium production from neutron activation and its removal through evaporation and concluded that the resultant increase in tritium production was negligible. Through an independent analysis performed using physical plant parameters available in the FSAR Tier 2, the NRC staff compared the amount of naturally occurring deuterium to the amount of deuterium produced through neutron activation of water and determined that the increase in tritium production was negligible. The NRC staff finds that the FSAR Tier 2 contains sufficient information about the concentration of tritium in the UHS pool water. Therefore, the NRC staff considers RAI 9259, Question 12.02-26, closed.

FSAR Tier 2, Revision 0, Table 12.2-32, describes assumptions for calculating airborne activity within the RXB. FSAR Tier 2, Revision 0, Section 12.2.2, does not discuss the sources of airborne radioactivity within the RWB, but it lists assumptions relevant to the determination of airborne activity concentrations in the RWB. TRU nuclides, such as americium, plutonium, and curium, are formed in irradiated uranium fuel by neutron activation and decay predominantly by alpha emission. These radionuclides are significant because of their presence in fluids in contact with reactor fuel, and alpha-emitting radionuclides have a significantly lower ALI than beta-gamma-emitting nuclides do (see 10 CFR Part 20, Appendix B, Table 1). FSAR Tier 2, Revision 0, Table 11.1-4, lists the radionuclide concentrations in the RCS. However, FSAR Tier 2 Table 11.1-4 does not list radiologically significant alpha-emitting radionuclides; therefore, FSAR Tier 2 Section 12.2 does not include the radiologically significant alpha-emitting radionuclides. FSAR Tier 2, Revision 0, Section 12.4.1.6, “Refueling Activities, including Dry Dock Outage Activities,” states that the major activities included in the dose assessment for refueling activities include disassembling the NPM and dry dock activities. While in the dry dock, components containing surfaces wetted by the RCS during operation will dry. Likewise, the surfaces of the dry dock pool wetted with pool water will dry. Although FSAR Tier 2 Section 12.3.3.3, “Reactor Building Heating Ventilation and Air Conditioning System,” states that the dry dock area is provided with exhaust flow to entrain airborne contamination that may result from NPM components being exposed to air during maintenance activities, FSAR Tier 2 Section 12.2 and FSAR Tier 2 Sections 12.3 and 12.4 do not describe the potential concentrations of radiologically significant alpha-emitting airborne radionuclides from dried surfaces. They do not discuss the airflow patterns, required airflow rates, and other design features provided to control airborne radioactive material during work in the dry dock. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the NRC staff could not identify how the applicant assessed the TRU airborne activity within the RXB. Therefore, the NRC staff issued RAI 9260, Question 12.02-27 and RAI 9260, Question 12.02 28, (ADAMS Accession No. ML18028A006 asking the applicant to provide additional information on the sources of TRU airborne radioactive material within the RXB and RAI 9255, Question 12.02-31 (ADAMS Accession No. ML ML18129A415), asking the applicant to provide additional information on the sources of TRU airborne radioactive material within the RWB.

In its response to RAI 9260, Question 12.02-27 and RAI 9260, Question 12.02-28 (ADAMS Accession No. ML18050A029), the applicant did not provide any specific TRU activities but specified that the source terms in FSAR Section 12.2 are derived from the reactor core inventory and the resultant primary and secondary coolant source terms provided in FSAR Section 11.1 and from activated components. The applicant stated that the 10 CFR 52.47(a)(22) requirement to demonstrate how the design incorporates operating experience insights only applies to insights gained from generic letters and bulletins. The applicant indicated that there is no regulatory guidance for performing such a TRU airborne activity analysis that would include values for variables such as the TRU alpha-emitting radionuclide fuel escape rate coefficient, deposition rate on wetted surfaces, and airborne fraction when surfaces become dry. The response also stated that the radiation protection program, which will be developed as part of COL Item 12.5-1, will address alpha-emitting radionuclides. The applicant also indicated that it followed RG 8.25 as part of the criteria for the selection and placement of fixed continuous air monitors as specified in FSAR Section 12.3.4.3. The NRC staff disagrees with the applicant's basis outlined in its response to RAI-9260, Question 12.02-28, for not providing the requested information. However, based on NuScale's plant-specific design information and on information about escape coefficients in other NRC guidance documents, the NRC staff was able to use information about the core isotopic inventory from NuScale's environmental report, titled "Applicants Environmental Report—Standard Design Certification (Rev. 1)—Section 1.0—Appendix B" (ADAMS Accession No. ML18086A070), to extrapolate potential airborne equilibrium values for major TRUs. This independent analysis found that the TRUs did not appear to be significant contributors to airborne radioactivity for NuScale's specific design; therefore, the NRC staff considers RAI 9260, Question 12.02-28, closed. Based on the NRC staff analysis of the applicant's response to RAI 9260 Question 12.02-27 and RAI 9260 Question 12.02-28, described above, the independent analysis found that the TRUs did not appear to be significant contributors to airborne radioactivity within the RWB; therefore, the NRC staff considers RAI 9255, Question 12.02-31, closed.

FSAR Tier 2, Revision 0, Section 12.2.3, "References," includes a reference to EPRI TR-3002000505, Volumes 1 and 2. TR-3002000505, Volume 2, 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. It further notes that, without operating reactor coolant pumps, the flow forces will be reduced. Some outcomes of reduced flow forces include increased deposition of suspended material, less solubilization of system deposits, and an increased rate of deposition in low flow rate areas. FSAR Tier 2, Revision 0, Section 12.2.1.3, states that, at the end of the fuel cycle, a crud burst is assumed, with the mixed-bed demineralizers being loaded with the entire radionuclide inventory increase resulting from the crud burst. The NuScale design has no reactor coolant pumps; therefore, during normal operation, the temperature gradients within the RCS drive the RCS system flow. As reactor power decreases, the temperature gradient decreases, which causes the RCS flow rate to decrease by about a factor of 10. Using the information provided in the application and information made available to the NRC staff as part of the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff was unable to determine how the application factored these aspects of the design into the estimated amounts of radioactive material projected to be initially present in the RCS following shutdown; the estimation of the effectiveness of the processes used to clean up the RCS; the amount of radioactive material that may be present inside of 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. Therefore, the NRC staff issued RAI 9263, Question 12.02-06 (ADAMS Accession No. ML17356A003), asking the applicant to provide additional information on the assumed RCS DFs and the resultant impact on radionuclide concentrations following reactor shutdown. In its response to RAI 9263, Question 12.02-06 (ADAMS Accession No. ML18022A396), the applicant stated that the duration and degree of the crud burst cleanup can vary widely based on the specific conditions at the time.

However, before CVCS disconnection and subsequent module disassembly, the CVCS crud burst cleanup of the primary coolant must continue until the projected dose rate at 1 meter above the reactor pool water is less than the criteria of 5 mrem/h (0.05 mSv/h) in EPRI TR-3002000505, Volumes 1 and 2, Revision 7. The applicant also proposed updating FSAR Tier 2 Section 12.2.1.8 to state that the post crud burst cleanup will continue until the projected dose rate at 1 meter above the UHS water is less than 5 mrem/h (0.05 mSv/h).

However, from reviewing the application, the NRC staff does not understand how 5 mrem/h (0.05 mSv/h) at 1 meter above the pool compares to the criteria of 2.5 mrem/h (0.025 mSv/h) in ANSI/ANS Std. 57.2-1983, "Requirements for Light-Water Reactor Spent Fuel Storage Facilities," (see DSRS Section 9.1.2 "New and Spent Fuel Storage," (ADAMS Accession No. ML15356A584)) to workers on the refueling bridge from pool water. In addition, as part of the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff determined that the applicant was calculating the dose to an operator on the refueling bridge based on a uniform concentration of water in the pool. The NRC staff notes that, when an NPM is disassembled, the concentration of radionuclides would be highest in the area near where the vessel was disassembled and would not result in an immediate uniform concentration across the entire pool. In a September 19, 2018, supplemental response to RAI 9263, Question 12.02-6 (ADAMS Accession No. ML18262A266), the applicant proposed an update to FSAR Tier 2 Section 12.2.1.8 to specify 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 2.5 mrem/h (0.025 mSv/h). This replaces the previous goal of 5 mrem/h (0.05 mSv/h) a meter above the UHS. The proposed criteria of less than 2.5 mrem/h (0.025 mSv/h) is consistent with the criteria in ANSI/ANS Std. 57.2-1983 that the NRC staff uses. Because a licensee operating a NuScale plant will have to operate the CVCS system during each outage until the projected dose rate to workers on the bridge is less than 2.5 mrem/h (0.025 mSv/h), consistent with ANSI/ANS Std. 57.2-1983, the NRC staff finds the design and proposed operation to be ALARA and, therefore, acceptable. **As a result, the NRC staff is tracking RAI 9263, Question 12.02-6, as a confirmatory item**, pending the incorporation of the proposed FSAR Tier 2 changes.

FSAR Tier 2 Revision 0, Section 12.2.2.1, "Reactor Building Atmosphere," states that airborne radioactivity may be present in the RXB atmosphere as a result of reactor pool evaporation or primary coolant leakage. The airborne concentration is modeled as a buildup to an equilibrium concentration based on the production and removal rate. The airborne concentration in the airspace above the reactor pool is determined by using the peak reactor pool water source term. FSAR Tier 2 Table 12.2-32 lists the input parameters and provides the pool evaporation rate. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff determined that the stated evaporation rate was based on assumed airflow rates over the pool surface and an assumed temperature of the UHS water. The NRC staff determined that the bulk average temperature limit in NuScale Technical Specification 3.5.3, "Ultimate Heat Sink," was significantly greater than the temperature assumed for determining the evaporation rate. As the pool temperature increases, the pool evaporation rate increases; therefore, the evaporation rate may be non-conservative for estimating airborne concentrations of tritium in the RXB. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff was not able to identify the bases of the assumed airflow rate above the UHS pool. The

NRC staff does not clearly understand what conditions (e.g., ventilation supply and exhaust flow rates) are assumed to meet the stated flow conditions. Therefore, the NRC staff issued RAI 9266, Question 12.02-12 (ADAMS Accession No. ML18008A099), asking the applicant to provide additional information on the assumed airflow rate over the UHS pool surface and other parameters related to the evaporation rate from the UHS pool and the subsequent estimates of concentrations of airborne radioactive material in the RXB. The applicant's response to RAI 9266, Question 12.02-12 (ADAMS Accession No. ML18058A901), provided some additional information in the FSAR Tier 2, but in the view of the NRC staff, additional information was needed. Following discussions between the NRC staff and the applicant, the applicant provided a supplementary response to RAI 9266, Question 12.02-12, on August 13, 2018 (ADAMS Accession No. ML18225A286), which updated the applicable portions of the FSAR Tier 2 to provide information on the assumed variables associated with pool evaporation rate and airborne equilibrium values of tritium and addressed the NRC staff's concerns about providing enough information in the FSAR Tier 2 to allow it to perform a confirmatory analysis. The applicant's response to RAI 9263, Question 12.02-06 dated September 19, 2018, (ADAMS Accession No. ML18058A901), provided the additional information requested by the NRC staff. **The NRC staff is currently tracking RAI 9266, Question 12.02-12, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

Because the methodology that the applicant used to calculate the evaporation rate from the UHS pool water appears to use non-bounding values, it may underestimate the total evaporation of tritium and other radionuclides from the reactor pool water. The applicant evaluated the evaporation rate of the reactor pool water at 100 degrees Fahrenheit (F). The applicant stated that it used this value because it is the design-basis temperature for the RXB heating, ventilation, and air conditioning (HVAC) system. The technical specification limit of the reactor pool water is 140 degrees F. Because airborne concentrations in the RXB are directly dependent on the UHS pool evaporation rate, the airborne activity concentrations in the RXB may be underestimated. Therefore, the NRC staff issued RAI 9266, Question 12.02-13 (ADAMS Accession No. ML18008A099), asking the applicant to provide additional information on the establishment of bounding assumptions for parameters related to the evaporation rate from the UHS pool and subsequent estimates of concentrations of airborne radioactive material in the RXB. After discussions with NuScale, the NRC staff was unable to obtain information needed to perform a confirmatory calculation to evaluate airborne tritium equilibrium values above 100 degrees F. The NRC staff is currently tracking RAI 9266, Question 12.02-13, as closed/unresolved. As a follow up to RAI 9266, Question 12.02-13, the NRC staff issued RAI 9613, Question 12.02-33 (ADAMS Accession No. ML18008A099), asking NuScale to either revise FSAR Tier 2 Table 12.2-36, "Input Parameters for Determining Facility Airborne Concentrations," to update the pool surface water temperature to reflect NuScale's Technical Specification 3.5.3 (i.e., the UHS bulk average temperature limit of 140 degrees F); establish a design-basis UHS temperature to limit evaporation and update the resultant change in the pool evaporation rate and provide a related update to FSAR Tier 2 Table 12.2-33, "Reactor Building Airborne Concentrations"; or provide a COL item in FSAR Tier 2 Section 12.5 that states that the COL applicant is responsible for developing the programmatic elements that address operating in unanalyzed conditions when the pool bulk temperature is above 100 degrees F. These program related elements would need to ensure that (1) adequate surveys and internal personnel monitoring for tritium and other radionuclides, as appropriate, are established and (2) the impact on the public dose is assessed. In a December 17, 2018, response to RAI 9613, Question 12.02-33 (ADAMS Accession No. ML18351A390), the applicant provided additional information related to the RXB airborne tritium source term. The NRC staff is continuing evaluation and modeling of the response. **The NRC staff is currently tracking RAI 9613, Question 12.02-33, as an open item**.

FSAR Tier 2, Revision 0, Table 12.2-32, states that the primary coolant source terms are derived from FSAR Tier 2, Revision 0, Table 11.1-4, which lists the RCS tritium (H3) concentration as 9.7000E-01 µCi/g. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff determined that the RCS tritium concentration listed in Table 11.1-4 was derived by assuming no recycling of RCS (i.e., all makeup water supplied to the RCS during the operating cycle was assumed to contain zero radioactivity, including tritium). FSAR Tier 2 Section 11.2, "Liquid Waste Management System," indicates multiple times that processed RCS liquids may be recycled for use in the RCS. The description of the CVCS in FSAR Tier 2 Section 9.3.4.2.1, "General Description," states that recycled, degassed reactor coolant from the LRWS can also be added back into the CVCS by a supply line upstream of the makeup pumps. The macroscopic cross-section of deuterium is based on the microscopic cross-section of deuterium (which does not change) and the relative atomic abundance of deuterium. With recycling, the relative atomic abundance of deuterium will increase over time from the activation of mononucleon hydrogen contained in the water. Based on the review of documents made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff determined that the applicant assumed a constant cross-section to produce tritium from the neutron activation of water in the reactor core. Therefore, the NRC staff issued RAI 9270, Question 12.02-19 (ADAMS Accession No. ML18008A224), asking the applicant to provide additional information on the buildup of deuterium in the RCS from recycling the RCS and the subsequent impact on tritium concentrations. In the applicant's response to RAI 9270, Question 12.02-19 (ADAMS Accession No. ML18155A622), the applicant indicated that it performed a study of the consequences of deuterium buildup in the RCS over the 60-year life of the plant from recycling primary coolant through the LRWS back to the RCS as makeup water. The study considered the production of deuterium in 12 NuScale modules with the dilution by the pool water from each refueling. This calculation indicated a maximum deuterium concentration in the RCS of 8.72E18 atoms per gram. This resulted in a production rate of tritium from deuterium activation in the core of 7.81E-3 micro-curies per second. This represented an increased tritium production of 9.1E-4 micro-curies resulting from deuterium buildup. The applicant concluded that the increased tritium product rate from deuterium buildup is too small to have any impact on the tritium values reported in FSAR Chapters 11 and 12. Through an independent analysis performed using physical plant parameters available in the FSAR Tier 2, the NRC staff compared the amount of naturally occurring deuterium to the amount of deuterium produced through neutron activation of water and determined that the increase in tritium production was negligible. Therefore, the NRC staff considers RAI 9270, Question 12.02-19, closed.

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 $1.3\times10^{+5}$ Becquerel per gram (3.5 µCi/g) or an alternate value for which the methods, models, and assumptions have been provided in the application. The methodology that the applicant used to calculate the tritium concentration in the RCS does not account for the buildup of tritium resulting from the recycling of previously used RCS fluid; therefore, RCS tritium concentration appears to be underestimated. The applicant has proposed an alternative and potentially non-conservative design-basis RCS tritium concentration value, which is used to determine airborne activity concentrations within the plant, without demonstrating that the health and safety of occupational workers is maintained and that the potential doses are ALARA for compliance with 10 CFR Part 20. Because airborne activity concentrations in equipment cubicles are more dependent on RCS activity concentrations and less on UHS pool tritium concentration, the airborne tritium activity concentrations in equipment cells may be underestimated by over a factor of 3. Because the NuScale FSAR Tier 2 submittal was not consistent with the acceptance criteria of DSRS Section 12.2 with respect to the RCS tritium concentration, the NRC staff issued RAI 9270, Question 12.02-20 (ADAMS Accession No. ML18008A224), asking the applicant to provide additional information on the buildup of tritium in the RCS as a result of recycling RCS fluid that contains tritium.

In its response to RAI 9270, Question 12.02-20 (ADAMS Accession No. ML18155A622), the applicant indicated that it considered two different primary coolant recycling modes to maximize radionuclide concentrations, including tritium. The applicant indicated that the mode that resulted in the highest tritium concentrations was the second recycling mode that models the recycling of primary coolant letdown as makeup back to the primary coolant system. In its response, the applicant described the recycling modes, explained how they are used and provided proposed FSAR Tier 2 updates. However, the proposed FSAR Tier 2 markups did not provide the peak RCS tritium concentration. In addition, in its response, the applicant updated tritium concentration information and proposed updates to FSAR Tier 2 Tables 11.1-4 through 11.4-7; Table 12.2-10, "Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup and Pool Surge Control System Component Source Terms—Radionuclide Content"; Table 12.2-32; and Table 12.2-33 based on changes to the realistic failed fuel fraction and DBFFF discussed in RAI 8759, Question 12.02-1, and RAI 9161, Question 11.01-1. These changes are also related to the response to the crud burst in RAI 9257, Question 12.02-14, and follow up RAI 9602. After a subsequent public meeting (ADAMS Accession No. ML18248A110) discussing this RAI, the applicant provided a supplemental RAI response on September 5, 2018 (ADAMS Accession No. ML18248A110). In its response, the applicant provided updated FSAR Tier 2 pages. Although FSAR Tier 2 Table 11.1-8 still contains the primary coolant average concentration of tritium, the applicant did provide a footnote to the table noting the maximum calculated peak primary coolant tritium activity of 3.5 μCi/g. Because the review of the applicant's calculations showed 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 response is acceptable with respect to tritium and the questions asked in this RAI. Based on an error in the airborne activity concentration calculations identified by the NRC staff during the DBFFF Phase II audit (see "Audit Summary for The Phase II Regulatory Audit of The Design Basis Failed Fuel Fraction for NuScale Power, LLC Design Certification Application," (ADAMS Accession No. ML18348A966)), the applicant agreed to provide a supplemental response to RAI 9270, Question 12.02-20. On November 19, 2018, (ADAMS Accession No. ML18323A288), the applicant provided a supplemental response that included changes to the airborne radioactivity concentrations. However, the NRC staff is continuing to work with the applicant to understand how the applicant arrived at the stated values. Therefore, **the NRC staff is tracking RAI 9270, Question 12.02-20, as open item**. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9270, Question 12.02-20 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.

FSAR Tier 2, Tier 2, Revision 0, Section 12.2.1.8, states that the neutron flux at the outside edge of the CNV was calculated to be approximately six orders of magnitude less than the average neutron flux in the core and continues to quickly decrease in the reactor pool's borated water. FSAR Tier 2 Section 12.2 does not provide any information on the flux and spectrum at the outside edge of the CNV. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the NRC staff was unable to identify the neutron flux and spectrum outside of the CNV near the reactor core. The NRC staff uses the stated neutron flux and energy to assess the impact on a variety of topics considered in the review, including the production of deuterium from neutron activation of hydrogen, the generation of tritium from activation of deuterium and neutron capture in boron, the generation of radioactive argon, and the activation of the containment structural materials. The NRC staff issued RAI 9283, Question 12.02-23 (ADAMS Accession No. ML18008A287), asking the applicant to provide additional information on the neutron radiation field outside of the CNV near the elevation of the reactor core. In its response to RAI 9283, Question 12.02-23 (ADAMS Accession No. ML18064A121), the applicant indicated that it performed calculations of water activation from neutrons escaping the core and entering the UHS water using identical energy groupings, as provided in FSAR Table 4.3-12, and the MCNP transport code and that the neutron flux at the outer edge of the CNV was many orders of magnitude less than the neutron flux in the core (approximately five orders of magnitude). The applicant also proposed to update FSAR Section 12.2.1.8 to correct an error and to state that the neutron flux on the outside of the CNV is many orders of magnitude less than the average neutron flux in the core. In a March 05, 2018, response to RAI 9283, Question 12.02-23 (ADAMS Accession No. ML18064A121), the applicant provided additional information related to the outside of the containment vessel at the reactor core elevation. Therefore**, the NRC staff is currently tracking RAI 9283, Question 12.02-23, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. However, the applicant did not provide the neutron fluence and spectra data that the NRC staff requested. Since the immediate exposure area is underwater, the NRC staff is evaluating the neutron fluence and spectral information related to areas of EQ and personnel protection under RAI 9282, which is discussed separately.

In order to evaluate the effectiveness of post-accident radiation and shielding design, the NRC staff needs the post-accident source terms that may result in dose to plant operators from airborne radioactive material that may be present following an accident. The applicant has stated that NuScale TR-0915-17565, Revision 3, provides these source terms. The NRC staff has not yet received the revised TR describing the NuScale post-accident source term. The NRC staff will update the SER following its review of the TR.

12.2.5 Combined License Information Items

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

Table 12.2-1 NuScale COL Information Items for Section 12.2

COL Item No.	Description	FSAR Tier 2 Section
12.2-1	A COL applicant that references the NuScale Power Plant design certification 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

12.2.6 Conclusion

The applicant has 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 has justified appropriate alternative methodologies. For post-accident shielding for vital area access, NuScale TR-0915-17565, Revision 3, provides the source terms. The NRC staff has not yet received the revised TR describing the NuScale post-accident source term. The NRC staff will update the SER following its review of the TR. The NRC staff reviewed and verified many of the sources in the NuScale design and determined that they are acceptable.

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 from 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 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 may be present, such as in the CVCS pump/valve room, the degasifier room, and the airspace above the reactor pool. However, these airborne source terms are still under review (see the discussion of RAI 9270).

The NRC staff particularly focused its review on the aspects of the NuScale application that were radiologically different in concept or implementation from currently licensed large light-water PWRs and deemphasized its review of sections of the application for which aspects of the NuScale design were less radiologically significant than the currently licensed fleet. Examples of increased focus included the application of a custom failed fuel fraction for the basis of the design of the shielding and ventilation systems, sources of radiation that are not contained by large masses of concrete shielding, the location of sensitive safety-related electrical equipment with respect to the types and sources of radiation, the implications of core power and flow regimes on RCS specific activity, and the use of shared systems for multiple modules. Examples of reduced NRC staff focus included the amount water shielding above irradiated fuel bundles and direct dose to MCR operators from post-accident radiation sources.

As described above, the NRC staff has reviewed the applicant's submittal against the requirements of 10 CFR Part 20 as it relates to: limits on doses to people in restricted areas, and the applicable requirements, including 10 CFR Part 19; sources of direct radiation exposure to members of the public, including 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 52.47(a)(5); 10 CFR 52.47(a)(22); and 10 CFR Part 50, Appendix A, GDC 4, GDC 19 and GDC 61, as they relate to the information on radiation sources provided by the applicant; 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 DC can be constructed and operated in conformity with the DC, the provisions of the AEA, and NRC regulations; and 10 CFR 20.1406(a) and 10 CFR 52.47(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.

With the exception of the open items and pending resolution of the confirmatory items discussed above, the NRC staff has determined that the NuScale design meets the applicable requirements discussed in Section 12.2. The NRC staff does not review programs during the design phase, therefore it is acceptable for COL applicants to address the operational considerations as described in the COL item applicable to this section. The NRC staff will review information submitted by the applicant to address the COL information item during the COL review.

12.3 Radiation Protection Design Features

This section covers both Section 12.3 and Section 12.4 of FSAR Tier 2 because NuScale DSRS Section 12.3–12.4 combines both sections.

12.3.1 Introduction

This section focuses on radiation protection design features, including the equipment used for ensuring that ORE 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; 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., SG tube plugging), refueling, and in-service inspection, provide a measure of the effectiveness of the proposed design features in reducing overall area dose rates.

12.3.2 Summary of Application

FSAR Tier 1: The FSAR Tier 1 information associated with this section includes FSAR Tier 1, Section 2.7, "Radiation Monitoring — Module Specific"; Section 2.8, "Equipment Qualification"; Section 3.3, Reactor Building Heating Ventilation and Air Conditioning System"; Section 3.9, "Radiation Monitoring—NuScale Power Modules 1–12"; Section 3.11, "Reactor Building"; Section 3.12, "Radioactive Waste Building"; Section 3.14, "Equipment Qualification - Shared Equipment", and consists of design features that demonstrate compliance with the occupational and public radiation safety requirements of 10 CFR Part 20, including those Tier 1 sections that address radiation shielding and zoning for radiological areas of the plant and radiation monitors, including the containment high radiation accident monitors, MCR ventilation accident radiation monitors, and fuel-handling area radiation monitors.

FSAR Tier 2: The applicant has described radiation protection design features in FSAR Tier 2 Sections 12.3 and 12.4, which is summarized, in part, as follows:

- Radiation protection design features include shielding, ventilation, radioactivity monitoring systems, and contamination control.

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

- Shielded compartments are provided for CVCS components located outside of the secondary shield in containment.

- A hot machine shop is provided so that maintenance can be performed on radioactive and contaminated equipment. The hot machine shop allows for maintenance and repair activities to be performed in lower radiation.

- 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.

ITAAC: FSAR Tier 1, Sections 2.7, 2.8, 3.3, 3.9, 3.11, 3.12, and 3.14provide the ITAAC associated with FSAR Tier 2, Sections 12.3 ITAAC related to the review of FSAR Tier 2 Section 12.3, are discussed in the following SER sections as indicated:

- Section 2.7, "Radiation Monitoring — Module Specific" – SER Section 14.3.8, "Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria";

- Section 2.8, "Equipment Qualification" – SER Section 14.3.6 – SER Section 14.3.6 "Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria";

- Section 3.3, Reactor Building Heating Ventilation and Air Conditioning System" – SER Section 14.3.8, "Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria";

- Section 3.9, "Radiation Monitoring—NuScale Power Modules 1–12" – SER Section 14.3.8, "Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria";

- Section 3.11, "Reactor Building" – SER Section 14.3.8, "Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria";

- Section 3.12, "Radioactive Waste Building"; Section 3.14– SER Section 14.3.8, "Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria";

- Section 3.14, "Equipment Qualification - Shared Equipment" – SER Section 14.3.6 "Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria";

Technical Specifications: FSAR Tier 2, Chapter 16, "Technical Specifications," Section 5.7, "High Radiation Area," addresses technical specifications for the control of high-radiation areas.

Technical Reports:

- NuScale TR-0116-20781-P, Revision 0

- NuScale TR-0915-17565, Revision 3

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 non-effluent sources associated with normal operations and AOOs.

- 10 CFR 20.1406 and 10 CFR 52.47(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 the minimizing leakage from systems outside of containment.

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

- GDC 4, “Environmental and Dynamic Effects Design Bases,” which requires 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 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) TEDE, 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 52.47(a)(5), 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 52.47(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(a)(25) and 10 CFR 52.47(a)(26), as they relate to the use of design interfaces for portions of the certified design that the NRC expects the COL applicant to implement.

- 10 CFR 52.47(b)(1), which requires a DC 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 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 ORE 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 Chapter 12 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 (NRR), to Regional Administrators, dated August 16, 1982, 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, and a memorandum from D.G. Eisenhut (NRR) 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 ORE 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 that FSAR Section 12 will provide.

- RG 1.143, as it relates to design features provided to minimize ORE 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, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," 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 ORE 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 Section 12.

- RG 8.10, as it relates to the commitment by management and vigilance by the radiation protection manager and NRC staff to maintain ORE 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 performing an assessment of collective occupational radiation doses as part of the ongoing design review process to ensure that such exposures will be ALARA.

- RG 8.25, 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 high-radiation areas and very high radiation areas (VHRAs).

- SRP Branch Technical Position (BTP) 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor

Plants," and SECY-94-198, "Review of Existing Guidance Concerning the Extended Storage of Low-Level Radioactive Waste," dated April 1, 1994, as they relate to design features provided to minimize ORE 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 Std. 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 (HPS) Std. 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 Std. 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 (ADAMS Accession No. ML062620355), and NUREG/CR-3587, "Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors," issued June 1986 (ADAMS Accession No. 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 that 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 Tier 2, Sections 12.3 and 12.4, and in other related sections of the FSAR Tier 2 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 follow the guidance of the RGs and applicable NRC staff regulatory positions or offered acceptable alternatives. In areas where the FSAR Tier 2 is consistent with the guidance in these RGs and NRC staff regulatory positions, the NRC staff can conclude that the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70 have been met. The sections below present the NRC staff's findings.

12.3.4.1 Radiation Protection Design Features

The reactor design incorporates features to help maintain ORE ALARA in accordance with the guidance in RG 8.8 and the requirements of 10 CFR 20.1101(b). These design features include facility design, shielding, ventilation, and area and airborne radiation monitors. These design features are founded in the ALARA design considerations described in FSAR Tier 2, Section 12.1, and discussed in SER Section 12.1.

12.3.4.1.1 Facility Design Features

Because the CNV is not accessible by personnel during operation, the sources of radiation inside the containment 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 from the reactor components.

Thermally treated Alloy 690 base metal is used for reactor coolant pressure boundary applications such as SG tubing material. Thermally treated Alloy 690 metal is used to reduce the possibility of intergranular stress-corrosion cracking, which could reduce equipment failure and, therefore, reduce worker dose resulting from maintenance activities.

FSAR Tier 2, Revision 0, Table 3C-6, states that the 60-year integrated N dose (rad) is for the area outside of the top of the pressurizer. NuScale TR-0116-20781-P, Revision 0, Table 5-1, describes the neutron fluence to the RV and CNV near the core but does not provide any neutron flux or spectrum information for the area above the pressurizer. The control rod drive mechanisms (CRDMs) are located in the area above the pressurizer and inside the CNV.

Based on information made available to the NRC staff as a result of the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), and RPAC participation in the CRDM audit (ADAMS Accession No. ML17158B428), the NRC staff became aware that the EQ program described in FSAR Tier 2 Section 3.11 did not include a number of B2 components (i.e., nonsafety-related and non-risk-significant components) that were located outside of the RCS pressure boundary but within the CNV. For example, information reviewed by the NRC staff during these audits specified the use of flexible metal hoses between the reactor's closed cooling water (RCCW) system and the CRDM magnet cooling coils. These hoses are classified as B2 items. When asked about this issue as part of the audit, the applicant stated that the hoses were rated for temperature conditions applicable to the RCCW system and not environmental conditions expected to be present outside of the RV but inside the CNV following the actuation of the reactor recirculation valves or the reactor vent valves. The NRC staff identified a similar situation for the RCCW thermal relief valves, which are located inside of the CNV and outside of the RV. Therefore, the NRC staff issued RAI 9245, Question 12.03-07 (ML18009B039), asking the applicant to provide additional information on how nonsafety-related SSCs located outside of the RV but inside the CNV are controlled to ensure that they do not adversely affect the operation of safety-related SSCs. In its response to RAI 9245, Question 12.03-07, dated March 8, 2018 (ADAMS Accession No. ML18067A788), the applicant stated that the components referred to in the original question are covered by the EQ program. However, the NRC staff required additional information to verify that all nonsafety-related equipment within the CNV but outside of the RV would not contribute to debris generation within the CNV. The applicant provided a supplemental response dated June 14, 2018 (ADAMS Accession No. ML18165A464) that included a revision to FSAR Tier 2 Section 3.11.6 to clarify that there would be no environmentally induced debris inside the CNV that would interfere with the proper functioning of the emergency core cooling system (ECCS). The NRC staff finds this response acceptable because this response clarifies that there will be no nonsafety-related equipment within the CNV that could interfere with the operation of the ECCS; therefore, it meets 10 CFR 50.46(b)(5) and 10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling," in providing assurance for long-term emergency cooling. In a June 14, 2018, supplemental response to RAI 9245, Question 12.03-7 (ADAMS Accession No. ML18165A463), the applicant provided additional information related to the qualification of equipment located inside of the containment vessel. **The NRC staff is currently tracking RAI 9245, Question 12.03-07, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

Based on information made available to the NRC staff as a result of the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), and the RPAC's participation in the CRDM audit, the NRC staff became aware of additional information about the length of the control rod drive shaft and the deceleration forces expected during the dropping of the control rods that could result in increased flexure at the control rod drive shaft to control rod assembly junction. This connection has the potential to create cobalt introduction rates into the RCS from wear at this junction. Therefore, the NRC staff issued RAI 9245, Question 12.03-08 (ML18009B039), asking the applicant to provide additional information on the amount of allowable flexure for the control rod drive shaft. In its response dated March 8, 2018 (ADAMS Accession No. ML18067A788), the applicant discussed the use of the control rod assembly cards in limiting the control rod flexure. In addition, the applicant provided a supplemental response dated June 14, 2018 (ADAMS Accession No. ML18165A464), which further clarifies the use of drive shaft supports in addition to the control rod assembly cards to limit the flexure of the CRDM shaft. The NRC staff finds this response acceptable because the applicant has described those features of the CRDM that would limit flexure of the shaft, and the limited flexure reduces the NRC staff's concern about cobalt introduction into the RCS. Therefore, the NRC staff considers RAI 9245, Question 12.03-08, closed.

FSAR Tier 2, Revision 0, Section 12.3.1, "Facility Design Features," describes facility design features that implement ALARA principles to minimize ORE. FSAR Tier 2 Section 12.3.1.1, "Equipment Design," provides specific design features for component types that aid in maintaining occupational exposures ALARA. However, the FSAR Tier 2 does not describe the design features of the dry dock provided to minimize ORE. FSAR Tier 2, Revision 0, Section 9.1.2.1, "Design Bases," states that smooth and nonporous surfaces prevent the buildup of radioactive material. FSAR Tier 2 Section 9.1.2.3.7, "Radiation, Shielding, and Maintaining Doses As Low As Reasonably Achievable," states that the surface finishes of the components for the fuel storage racks and spent fuel pool (SFP) liner are smooth to minimize accumulation of radioactive materials and to facilitate surface decontamination. Section 2.3.1.3.1.2 of EPRI TR-016780, "Advanced Light Water Reactor Utility Requirements Document" (URD), states, "The refueling pool wall liner shall be surface finished to reduce the adherence of contamination and increase the efficiency of refueling pool decontamination activities after draining. The liner plate shall have a No. 4 surface finish or better and the liner plate welds shall be ground smooth." The reason given in the URD for this specification is that past refueling experience in light-water reactors has shown that a smooth surface finish on the wall liners reduces the amount and depth of crevices that can accumulate contamination. NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," Volume 3, Parts 1 and 2, issued August 1994, documents the NRC staff's safety evaluation of the URD. However, the application does not describe the specification for the surface finish of those portions of the facility (i.e., the dry dock) that, when dry, may increase ORE as a result of direct radiation exposure from surface deposits of radioactive material or from airborne radioactive material caused by the suspension of radioactive material remaining on the pool wall surface following dry dock drain down. Therefore, the NRC staff issued RAI 9284, Question 12.03-29 (ML18028A003), asking the applicant to provide additional information on what degree of surface finish should be applied to the dry dock wall to demonstrate compliance with the requirements of 10 CFR 20.1101(b). In its response to RAI 9284, Question 12.03-29 (ADAMS Accession No. ML18047A750), the applicant specified that it will use both procedures and engineering controls to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical. The applicant indicated that FSAR Sections 9.2.5.3 and 9.1.3.3.8 state that the water level in the dry dock is adjusted to provide radiation shielding and that the PCUS is used to clean the water in the dry dock to reduce the radionuclide concentrations. The applicant also specified that, consistent with RG 8.8, FSAR Section 12.3.6.1.3 and Table 12.3-42 state that the surfaces of the UHS liner (which includes the dry dock liner) will be smooth to minimize contamination and facilitate decontamination and that FSAR Sections 9.4.2 and 12.3.3.3 state that the HVAC system provides elevated flow rates in this area to reduce airborne activity. However, the applicant did not specify a No. 4 finish for the dry dock area, as called for in the URD. Such a finish would allow decontamination of the dry dock to be performed more efficiently and would likely significantly reduce airborne contamination. The applicant did not address the issue in its supplemental response; therefore, **the NRC staff is currently tracking RAI 9284, Question 12.03-29, as an open item**.

Likewise, FSAR Tier 2 Chapters 3, 5, and 12 do not discuss the surface finish of the exterior CNV. Like the dry dock wall, when dry, ORE results from direct radiation exposure from surface deposits of radioactive material or from airborne radioactive material caused by the suspension of radioactive material remaining on the large, wetted surface area of the CNV wall following dry dock drain down. Therefore, the NRC staff issued RAI 9284, Question 12.03-30 (ML18028A003), asking the applicant to provide additional information on what degree of surface finish should be applied to those portions of the wetted CNV wall that may dry while in the dry dock area to demonstrate compliance with the requirements of 10 CFR 20.1101(b). In its response to RAI 9284, Question 12.03-30 (ADAMS Accession No. ML18047A750), the applicant referred to its response to RAI 9284, Question 12.03-29 (discussed above), indicating that, in lieu of providing surface finish specifications for the CNV, it will use procedures and other engineering controls to maintain doses in the area ALARA. Based on operating experience available to the NRC staff, a smooth finish of the CNV would allow decontamination to be performed more efficiently and would likely significantly reduce airborne contamination. The applicant did not address the degree of surface finish that should be applied to those portions of the wetted CNV wall that may dry while in the dry dock area in its supplemental response; therefore, **the NRC staff is currently tracking RAI 9284, Question 12.03-30, as an open item**.

FSAR Tier 2, Revision 0, Section 11.4.2.5.2, “Pumps,” states that two SRST transfer pumps are used to take suction from the decant portion of the resin storage tanks and provide water to sluice spent resins from the PCUS and CVCS demineralizers to an SRST. This provides the motive force to sluice resins while minimizing the generation of radioactive waste. These pumps can also be used to fluff the spent resins inside the SRST by recirculating decant water before transferring spent resins to an HIC. A similar arrangement exists for the PST.

FSAR Tier 2 Figure 11.4-2a, “Process Flow Diagram for Wet Solid Waste,” and FSAR Tier 2 Figure 11.4-2b, “Solid Radioactive Waste System Diagram,” show a line from the service air system that is separated from the suction of the resin transfer pumps by a single isolation valve.

The use of a single isolation valve increases the risk for air intrusion caused by valve leakage or misalignment. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the suction isolation valves for the pumps appear to be diaphragm disk valves. This information appears to be consistent with FSAR Tier 2 Section 11.4.2.5.3, “Piping and Valves,” which states that valves in slurry transfer lines are full-ported ball valves and that liquid process valves are diaphragm valves. Operating experience is available (e.g., EPRI TR-105852, “Valve Application, Maintenance, and Repair Guide,” Volume 1) to the NRC staff that indicates that leaks past the seats of diaphragm valves can occur as a result of poor stem travel adjustment, diaphragm age, and the oversetting of the stem travel. Typically, valves in the LRWS are not in a periodic performance testing (i.e., leakage testing) program, and the NRC staff has not seen any information in the application that indicates that they are in a performance testing program. Information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), indicated that these pumps are centrifugal pumps with an open impeller type. However, none of the information in the FSAR Tier 2 application or information made available to the NRC staff during the audit indicated what design features were provided to prevent pump damage. In addition to the items related to pump failures caused by air intrusion, as stated in FSAR Tier 2 Section 11.4.2.5.2, the pumps are used to take a suction through the backwash screens. Operating experience available to the NRC staff (e.g., NUREG/CR-4245, “In-Plant Source Term Measurements at Brunswick Steam Electric Station,” issued June 1985, and NUREG/CR-6365, “Steam Generator Tube Failures”) indicates that it is not uncommon for particulate matter smaller than the resin retention screen mesh size (i.e., corrosion and wear products and “resin fines”) to pass through these screens. After passing through the screens, this particulate matter can cause damage to sealing surfaces and accumulate in downstream components. The wear on sealing surfaces requires increased maintenance, whereas the accumulation of radioactive waste products causes increased dose rates and subsequent occupational radiation worker exposure. Working on plant components handling radioactive waste frequently involves high dose rates and high beta-gamma contamination levels and may involve high TRU contamination levels; therefore, the potential for high ORE is elevated. The physical arrangement of the service air line with the resin transfer pumps and the absence of design features to prevent pump air binding or pump damage does not appear to address operating experience (e.g., EPRI TR-1026498, “Report of the Expert Panel on the Effect of Gas Accumulation on Pumps”) and may result in increased ORE. Therefore, the NRC staff issued RAI 9285, Question 12.03-41 (ML18033A747), asking the applicant to provide additional information on the design features provided to prevent air-intrusion-related damage to these pumps. In addition, because the application does not contain appropriate supporting information on design features (e.g., resin screen mesh size, seal design parameters) provided to prevent pump/seal damage from corrosion products and resin fines and because it does not address operating experience that may result in increased ORE, the NRC staff issued RAI 9285, Question 12.03-42 (ML18033A747), asking the applicant to provide additional information on the design features provided to prevent damage to these pumps from material-penetrating resin retention elements. In its response to RAI 9285, Questions 12.03-41 and 12.03-42, dated March 9, 2018 (ADAMS Accession No. ML18068A633), the applicant detailed the possible pathways that air and water could take within the system and described the design features of pump flushing and line flushing that could be done to minimize ORE. The applicant also provided information to discuss two valves of separation when using air sparging to mix the resin. In addition, the applicant provided information about the pumps used for the SRST and the PST. The applicant detailed the use of pumps with no seals to eliminate the need to protect seals. The response to RAI 9285 Question 12.03-42, described design features of the pumps that addressed the NRC staff's concerns about pump reliability. Therefore, the NRC staff considers RAI 9285 Question 12.03-42, closed. In its supplemental response, dated June 4, 2018 (ADAMS Accession No. ML18155A540), the applicant clarified that the service air was used to fluff the resins through the sparging nozzles at the bottom of the SRSTs and PSTs and that air was not used to backwash the tank decant screens. The NRC staff finds these responses acceptable because the applicant has described those design features considered in the design of the SRST and PST to reduce ORE**. The NRC staff is currently tracking RAI 9285, Question 12.03-41, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

FSAR Tier 2, Revision 0, Section 12.3.1.1.13, "Material Selection," states that proper material selection is an important factor to balance component performance while reducing the amount of corrosion and activation products generated. The use of materials containing cobalt and nickel is minimized to reduce the quantity of activation products. Nickel-chromium-iron alloys, such as Inconel, have a high nickel content that can become Co-58 when activated. The production of Co-58 and Co-60 is reduced by using low-nickel and low-cobalt bearing materials to the extent practicable. FSAR Tier 2, Revision 0, Table 12.3-4, "Typical Cobalt Content of Materials," states that the maximum weight percent (w%) of cobalt in the austenitic stainless steel base materials is 0.15 w%. FSAR Tier 2, Revision 0, Section 5.2.3.2.2, "Compatibility of Construction Materials with Reactor Coolant," states that the use of cobalt-based alloys is minimized and that limits are established to minimize cobalt intrusion into the reactor coolant. Section 5.2.7, "Metallic Materials," of Volume 3, "ALWR Passive Plants," of the EPRI URD states that cobalt used for components fabricated with stainless steel or nickel base alloy with a large wetted surface area (e.g., major piping, clad) and with an operating temperature above 200 degrees F shall be restricted to a maximum content of 0.050 w%. In addition, the plant designer shall specify cobalt content targets (mean values) lower than 0.05 w% for components (1) that are made of stainless steel or a nickel-based alloy located in or near the core where neutron flux is high enough to cause activation of Co-59 to Co-60 in significant amounts, (2) that are expected to release significant quantities of corrosion products in the reactor coolant stream, and (3) that are expected to be a significant source of radiation exposure to plant maintenance personnel. Co-60 has a very long half-life and emits penetrating gamma rays, and it is the major contributor to buildup of radioactivity in the plant. NUREG-1242, Volume 3, Part 1, Section 5.2.7, "Metallic Materials," states that metallic materials in contact with reactor coolant, such as austenitic stainless steel or carbon and low-alloy steels, should be restricted in cobalt content to as low a level as practical for all components that are made of stainless steel or a nickel-based alloy and that have a large wetted surface area. However, the allowable cobalt content of bulk structural material exposed to a neutron flux in the application (e.g., the neutron reflectors described in FSAR Tier 2 Section 5.1.1) is a factor of 3 higher than industry and NRC-recognized criteria as discussed above. The large volume of material in conjunction with the higher allowable cobalt content will increase the cobalt available for irradiation in the NPM. Because cobalt is a major source of radiation exposure in operating nuclear power plants and during decommissioning, the use of a higher cobalt-containing material will increase radiation exposure contrary to the requirements of 10 CFR 20.1101(b) and fails to minimize the production of radioactive material in support of facilitating eventual decommissioning and minimizing, to the extent practicable, the generation of radioactive waste in accordance with 10 CFR 20.1406. Therefore, the NRC staff issued RAI 9286, Questions 12.03-11 and 12.03-12 (ML18012A749), asking the applicant to explain and justify the cobalt content of components in high neutron fluence areas and components in areas in contact with RCS coolant. In its response to RAI 9286, Questions 12.03-11 and 12.03-12 (ADAMS Accession No. ML18053A227), the applicant provided additional information on the cobalt specification for different materials. It indicated that, for large forgings, NuScale imposes a cobalt limitation of 0.05 w% maximum for large forgings, and for small components or lot sizes, the base metal is typically procured from warehouses that cannot always guarantee the availability of extra low cobalt materials. Therefore, for those materials, the overall maximum cobalt for the base metal is limited to 0.15 w%. The applicant also stated that the reactor pressure vessel base metal cobalt content is limited to 0.10 w% maximum. In Revision 1 to RAI 9286, Question 12.03-11, (ADAMS Accession No. ML18225A248), the applicant updated FSAR Tier 2 Table 12.3-4 to specify that the neutron reflector has a maximum cobalt content of 0.05 w%. Because NuScale is specifying a low cobalt content of 0.05 w% for large components and applies appropriate limits for other components in neutron flux and for surfaces in contact with the RCS, the NRC staff finds that the design appropriately limits cobalt, consistent with the requirements of 10 CFR 20.1101(b), and is, therefore, acceptable. Therefore, the NRC staff is tracking RAI 9286, Question 12.03-11, as a confirmatory item, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision, and RAI 12.03-12 closed.

FSAR Tier 2, Revision 0, Section 12.3.2, "Shielding," describes some of the design considerations for radiation shielding (e.g., stating that concrete is the material used for a significant portion of plant shielding). FSAR Tier 2 Section 12.3.2.2, "Design Considerations," states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. The material used for a significant portion of plant shielding is concrete. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. FSAR Tier 2 Section 12.3.2.4.3, "Reactor Building," states that cubicle walls are made of concrete that is supported by carbon steel plates; these walls are called "structural steel partition walls." FSAR Tier 2 Table 12.3-6 provides the nominal thickness of concrete for some of the walls in the RXB. FSAR Tier 2 Table 12.3-8, "Reactor Building Radiation Shield Doors," lists the shielded doors located in the RXB. FSAR Tier 2 Table 12.3-7, "Radioactive Waste Building Shield Wall Geometry," provides the nominal thickness of concrete for some of the walls in the RWB. FSAR Tier 2 Table 12.3-9, "Radioactive Waste Building Radiation Shield Doors," lists the shielded doors located in the RWB. In an NRC letter to NuScale, "Transmittal of Draft Standard Inspections, Tests, Analysis and Acceptance Criteria," dated April 8, 2016, and its enclosure that contains standardized ITAAC tables (ADAMS Accession Nos. ML16096A132 and ML16097A123), the NRC staff described the ITAAC that are applicable to its review of the NuScale application. These standard ITAAC include the following:

- R07, "As-Built Inspection and Reconciliation Analysis," verifies that the SSCs of the non-seismic Category I radioactive waste system are designed and constructed to the standards of RG 1.143 to withstand the design loads without loss of structural integrity. RG 1.143, Table 1, "Codes and Standards for the Design of SSC in Radwaste Facilities," describes the design codes and standards that the NRC expects licensees to meet to demonstrate that the health and safety of members of the public and workers at the facility will be protected for the operational conditions described within RG 1.143, Table 2, "Natural Phenomena and Internal/External Man-Induced Hazard Design Criteria for Safety Classification," and RG 1.143, Table 3, "Design Load Combinations."

- R09, “As-Built Inspection and Analysis Containment High Range Radiation Monitor—Location,” either checks that the radiation monitors were installed in the location described and that no obstructions to the view of the radiation monitors were added to the design or verifies that, if the design specified a percent of the containment atmosphere free volume view for each radiation monitor, the radiation monitors have been installed in a location and manner that provides for appropriate monitoring of the containment radiation levels following an accident.

FSAR Tier 1, Revision 0, Chapter 1, “Certified Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC),” Section 3.12, contains an ITAAC that ensures that the RWB will be designed as a Class RW-IIa radioactive waste structure in accordance with RG 1.143; however, it does not contain ITAAC that correspond to R07 with respect to verifying that the systems and components that contain radioactive waste were designed and constructed consistent with the guidance in RG 1.143, which enables licensees to demonstrate compliance with the provisions of 10 CFR Part 20 that are related to protecting the health and safety of members of the public and occupational radiation workers. Therefore, the NRC staff issued RAI 9303, Question 12.03-52 (ML18092B231), asking the applicant to provide additional information on the ITAAC that ensure that the radioactive waste SSCs have been designed and constructed in a manner that protects the health and safety of the public and occupational workers. In its response to RAI 9303, Question 12.03-52 (ADAMS Accession No. ML18149A643), the applicant specified that an ITAAC to verify that the NuScale radioactive waste systems are designed and constructed in accordance with RG 1.143 is not necessary for the following reasons:

- The NuScale radioactive waste systems do not have any safety-related or risk-significant functions.

- The NuScale radioactive waste systems do not support the safety-related or risk-significant functions of another system.

- The radioactive waste systems do not contain top-level design features, as described in FSAR Section 14.3.2.1.1, for shielding that protects the health and safety of workers.

- The health and safety of the public is protected by ITAAC that ensure high radiation will be contained within the RWB. The related ITAAC verify the following top-level design features:

  – High-radiation liquid in the LRWS is automatically isolated from the environment through containment of the liquid in the LRWS.

  – High-radiation gas in the gaseous radioactive waste system (GRWS) is automatically isolated from the environment through containment of the gas in the GRWS.

  – High-radiation gas in the RWB is contained and precluded from leakage to the outside environment by keeping the RWB pressure negative relative to the outside environment.

  – The as-built Class RW-IIa RWB maintains its structural integrity under the design-basis loads.

However, the ITAAC listed by the applicant do not address the design and installation of the system and components consistent with RG 1.143; therefore, the NRC staff is currently tracking RAI 9303, Question 12.03-52, as closed/unresolved and issued RAI 9608, Question 14.03.08-1, asking the applicant to provide additional justification for why an ITAAC on the design of the radioactive waste systems is not needed. **The NRC staff is tracking RAI 9608, Question 14.03.08-1, as an open item**.

FSAR Tier 1, Revision 0, Chapter 1, Section 3.9, does not contain ITAAC corresponding to R09 that verify that the containment high-range radiation monitors installed are consistent with the intent of 10 CFR 50.34(f)(2)(vii). Therefore, the NRC staff issued RAI 9303, Question 12.03-53 (ML18092B231), asking the applicant to provide additional information on the ITAAC for the containment high-radiation monitor. In its response to RAI 9303, Question 12.03-53 (ADAMS Accession No. ML18149A643), the applicant clarified that ITAAC 02.05.25 in FSAR Tier 1, Table 2.5-7, states that "the PAM Type B and Type C displays are indicated on the SDIS [safety display and information system] displays in the MCR," and that the displays discussed in the ITAAC are "the PAM Type B and Type C displays listed in Table 2.5-5 are retrieved and displayed on the SDIS displays in the MCR." Because the under-the-bioshield monitors are PAM Type B and Type C displays, ITAAC 02.05.25 will require that the inside-the-bioshield area radiation monitor is displayed on the SDIS in the MCR. As a result, the NRC staff concludes that the FSAR Tier 2 includes an appropriate ITAAC for the under-the-bioshield radiation monitors. As a result, the NRC staff finds the response to RAI 9303, Question 12.03-53, to be acceptable, and RAI 9303, Question 12.03-53, is closed.

The acceptance criteria of DSRS Section 12.3–12.4 state that, where the applicant's shielding design incorporates materials subject to degradation, such as through the effects of radiation (e.g., depletion of boron neutron absorbers), temperature extremes (e.g., degradation of polymer-based materials caused by high temperature), and density changes (e.g., sagging or settling of shielding material with age), the applicant should specify the methods in place to ensure that ORE remains ALARA and that the equipment exposures are maintained in accordance with the provisions of 10 CFR 50.49(e)(4) and 50.34(f)(2)(vii). The criteria further state that the application should identify the allowable constraints (e.g., minimum cooling airflow, maximum shielding material temperature, and maximum allowable neutron flux) and should describe how those parameters are measured and assessed over the design life of the facility. FSAR Tier 2 Section 12.3.2.2 states that, in addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. However, FSAR Tier 2, Revision 0, Section 12.3.2, does not identify any areas of the plant shielding (e.g., penetration shielding around hot pipes) that have limitations associated with the shielding material or for which specific design criteria (e.g., maximum temperature, radiation resistance) are required to ensure that the integrity of the shielding is maintained. FSAR Tier 1, Section 3.11, states that the RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding. It further states that FSAR Tier 1, Table 3.11-2, "Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria," contains the inspections, tests, and analyses for the RXB. FSAR Tier 1, Section 3.12, states that the RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding. It further states that that FSAR Tier 1, Table 3.12-2, "Radioactive Waste Building ITAAC," contains the inspections, tests, and analyses for the RWB. However, FSAR Tier 1, Section 3.11, and FSAR Tier 1, Section 3.12, do not describe ITAAC for verification of the design features (e.g., the minimum airflow rate around hot pipes to prevent degradation of concrete, minimum spacing between hot pipes and structural components provided to prevent degradation of materials) provided to ensure the continued integrity of the radiation shielding. Because routine tests or inspections of shielding material are not capable of assessing the degradation (e.g., dehydration of concrete caused by high temperature) of the shielding material that may adversely impact its ability to perform under design-basis conditions (i.e., accident or AOO source terms following DBEs such as earthquakes), the provision and function of design features for protection of shielding are important to ensure the continued integrity of the shielding. Therefore, the NRC staff issued RAI 9303, Question 12.03-54 (ML18092B231), asking the applicant to provide additional information on the adequacy of the ITAAC on design features to ensure the continued integrity of the radiation shielding. In its response to RAI 9303, Question 12.03-54 (ADAMS Accession No. ML18149A643), the applicant indicated that it has ITAAC for radiation shielding that ensures the health and safety of workers. The applicant indicated that it would not include protective features for radiation shielding in ITAAC. The NRC staff's primary concern about ensuring the integrity of radiation shielding was to ensure the integrity of the polyethylene bioshield shielding because radiation and other environmental conditions degrade polyethylene over time. However, the applicant is removing the polyethylene bioshield shielding from the NuScale design, as discussed in RAI 9294, Question 12.03-26 (ADAMS Accession No. ML18080A127). Because polyethylene will no longer be a part of the design, the NRC staff does not believe it is necessary to include an ITAAC as part of the design to ensure the integrity of the radiation shielding. As a result, the response is acceptable, and RAI 9303, Question 12.03-54, is closed.

Based on the above, except for the open and confirmatory items, the NRC staff concludes that the NuScale FSAR Tier 2 adequately addresses radiation protection design features.

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. 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). Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area in the RXB in FSAR Tier 2, Section 12.3, Figures 12.3-1a through 12.3-1i, and the RWB in Figures 12.3-2a and 12.3-2b and to ensure that ORE and doses to members of the public remain ALARA. FSAR Tier 2, Tables 12.3-6 and 12.3-7, provide concrete shielding thicknesses for rooms and cubicles containing significant radiation sources, which require shielding. These shielding thicknesses are based on the design-basis source terms provided in FSAR Tier 2, Chapter 11 and Section 12.2. The applicant used Oak Ridge National Laboratory's Standardized Computer Analyses for Licensing Evaluation (SCALE) (see "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design") and MCNP to perform source term and radiation shielding and zoning dose calculations. For most applications, concrete shielding is designed in accordance with ANSI/ANS Std. 6.4-2006, using material descriptions from Pacific Northwest National Laboratory (PNNL)-25870, "Compendium of Material Composition for Radiation Transport Modeling," Revision 1. Radiation shielding material other than concrete includes water to shield fuel and sources outside the CNV and the use of radiation shield doors as provided in FSAR Tier 2 Tables 12.3-8 and 12.3-9. These areas that rely on shielding materials other than concrete are discussed later.

FSAR Tier 2, Revision 0, Figure 12.3-2a, shows the Class A, B, and C HIC storage area (Room 030-034 in FSAR Tier 2 Figure 1.2-28) as Radiation Zone VII. FSAR Tier 2, Revision 0, Table 12.3-1, shows that areas designated as Radiation Zone VII have dose rates greater than or equal to 500 rad/h (5 Gy/h), with no upper limit specified. FSAR Tier 2 Table 12.3-7 lists the concrete thickness of interior walls of Room 030-034 and the concrete thickness of the ceiling of Room 030-034. FSAR Tier 2 Figure 12.3-2a shows the adjacent room (identified in FSAR Tier 2 Figure 1.2-28 as room number 030-004, which is a tank room) as Radiation Zone 1. Table 12.3-1 shows that areas designated as Radiation Zone I have dose rates greater than or equal to 0.05 mrem/h (0.0005 mSv/h) and less than or equal to 0.25 mrem/h (0.0025 mSv/h). The NRC staff's analysis of the stated dose rate in Room 030-034 and the attenuation provided by the concrete thickness between Rooms 030-034 and 030-004 is not consistent with the radiation zone assigned to Room 030-004 even without considering any source terms specific to Room 030-004. In addition, FSAR Tier 2, Figure 12.3-2b, identifies "Truck Bay" (Room 030-103 in FSAR Tier 2 Figure 1.2-30, "Radioactive Waste Building 100'-0" Elevation") as Radiation Zone II. Table 12.3-1 shows that areas designated as Radiation Zone II have dose rates greater than or equal to 0.25 mrem/h (0.0025 mSv/h) and less than or equal to 2.5 mrem/h (0.025 mSv/h). The NRC staff's analysis of the stated dose rate in Room 030-034 and the attenuation provided by the concrete thickness between Rooms 030-034 and 030-103 is not consistent with the radiation zone assigned to Room 030-103 even without considering any source terms specific to Room 030-103. Therefore, the NRC staff issued RAI 8859, Question 12.03-61 (ML18180A354), asking the applicant to provide additional information on the determination of the radiation zone designations for areas in the RXB and RWB. In its response to RAI 8859, Question 12.03-61 (ADAMS Accession No. ML18232A561), the applicant clarified that Room 030-004 was not adjacent to Room 030-034; however, Room 030-006 is adjacent to Room 030-034 with a 3-foot-thick wall between them. The applicant indicated that the HIC storage room source term is based on an assumed five HICs and one dryer drum arranged in two rows in a single layer centered in the room. The applicant indicated that it accounted for the elevation difference between Room 030-034 (elevation of 71 feet) and Room 030-103 (elevation of 100 feet) in calculating the dose in Room 030-103. In its evaluation of RAI 9269, the NRC staff performed an analysis that verified the radiation zoning of the HIC storage room and its adjacent areas. In its analysis, the NRC staff considered that the HIC storage area was filled with HICs and performed an analysis to determine the dose rates in the adjacent rooms and above the truck bay area. In its analysis, the NRC staff concluded that the radiation zoning specified by the applicant is acceptable; therefore, the applicant's response to RAI 8859, Question 12.03-61, is acceptable because the NRC staff received enough information to perform an analysis, and the subsequent analysis determined the zoning for nearby areas was appropriate. The NRC staff considers RAI 8859, Question 12.03-61, closed.

FSAR Tier 2, Revision 0, Section 12.3.2, describes some of the design considerations (e.g., stating that concrete is the material used for a significant portion of plant shielding). For most applications, concrete shielding is designed in accordance with ANSI/ANS Std. 6.4-2006. FSAR Tier 2 Section 12.3.2.3, "Calculation Methods," states that shielding credit and material selections for modeled cells are conservatively applied. PNNL-25870, "Compendium of Material Composition for Radiation Transport Modeling," Revision 1, provides the material compositions for air, concrete, water, and stainless steel. FSAR Tier 2 Section 12.3.2 does not contain any information about the assumption for concrete density other than the references to ANSI/ANS Std. 6.4-2006 and PNNL-25870. The NRC staff used the information from the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), to identify a number of differences between the shielding design information described in the RWB shielding calculation package and the relevant parameters specified in ANSI/ANS Std. 6.4-2006 and PNNL-25870. The NRC staff noted that the density specified for resin contained in some sources appeared to differ from that discussed in the FSAR Tier 2. The NRC staff used the information from the RPAC Chapter 12 Phase I audit to identify that the RXB and RWB shielding calculation packages referenced gamma photon strength values whose derivation was described in a different calculation package. Based on the NRC staff's review, the application of the photon binning method used by the applicant for some of the sources in the RXB and the RWB appeared to be inconsistent. Therefore, the NRC staff issued RAI 9279, Question 12.03-57(ML18134A244), asking the applicant to provide additional information on the derivation and use of parameters needed to ensure accurate shielding calculations. In the RAI 9279, the NRC staff identified the following four discrepancies in the shielding design information:

(1) The assumed concrete density for the RWB shield walls appear to be inconsistent with stated standards and non-conservative for radiation attenuation.

(2) The density specified for resin contained in some CVCS demineralizers appeared to be different than that discussed in the FSAR Tier 2.

(3) The density stated for the GAC filtration media appeared to be non-conservative for radiation attenuation in the GAC.

(4) The RXB and RWB shielding calculation packages referenced gamma photon strength values whose derivation was described in a different calculation package.

Based on the NRC staff's review of information available in FSAR Tier 2, and made available for the staff's review during an audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), the application of the photon source strength method used by the applicant for some sources in the RXB and the RWB appeared to be inconsistent. In its response dated August 21, 2018 (ADAMS Accession No. ML18156A544), the applicant directly addressed each of the discrepancies as follows:

(1) With respect to the concrete densities, NuScale provided the assumed density of the concrete and the related standards. As stated in FSAR Section 12.3.2.3, the RXB and RWB shielding calculations use a concrete density based on PNNL-25870, Revision 1 (i.e., 2.3 grams per milliliter (g/mL), material 99 for regular concrete), and SCALE, Version 6.1, manual regulatory concrete (developed for the NRC).

(2) With respect to the resin density, NuScale noted that shielding calculations for the RXB and RWB use a dry resin density of 760 grams per liter (g/L). This density was applied to ion exchanger vessels, SRSTs, resin transfer lines, and HICs. This density value is conservative because it is for a dry resin condition. Shielding calculation resin densities for a wet resin (water-saturated) condition is about 1,000 g/L; therefore, an assumed resin density of 760 g/L provides less self-shielding and, therefore, is conservative.

(3) With respect to the GAC density, NuScale stated that it calculated the GRWS decay bed vessel volumes using a 0.5-g/mL basis in accordance with a vendor's technical specification sheet for a range of 0.5- to 0.6-g/mL density of activated carbon for gas delay. The shielding calculation used the lower density as a design-basis self-shield density for GRWS decay beds and for the LWRS granulated, activated carbon beds that will be filled with liquid.

(4) With respect to the photon source strengths, NuScale stated that source strengths are generated using the industry standard code, SCALE. Tally multipliers are a shielding calculation variable that can be used to define the total source strength like a weight

card can be used to describe the total source strength. These values are developed within engineering calculations and, where necessary, reported in the FSAR. The NuScale quality assurance program and engineering procedures are implemented to ensure that engineering products are checked and validated.

The NRC staff found this response acceptable because NuScale justified and explained parameters used in the shielding analysis, which addressed the NRC staff's concerns about the potential for inconsistent application of shielding analysis techniques. The NRC staff considers RAI 9279, Question 12.03-57, closed.

FSAR Tier 1, Section 3.11, states that the RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding. It further states that FSAR Tier 1, Table 3.11-2, contains the inspections, tests, and analyses for the RXB. FSAR Tier 1, Table 3.11-1, Item 4, "Acceptance Criteria," states that the thickness of RXB radiation shielding barriers is greater than or equal to the required thickness specified in FSAR Tier 1, Table 3.11-1. FSAR Tier 1, Section 3.12, states that the RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding. In addition, the RWB includes radiation attenuating doors for normal operation and for post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed. FSAR Tier 1, Section 3.12, further states that FSAR Tier 1, Table 3.12-2, contains the inspections, tests, and analyses for the RWB. FSAR Tier 1, Table 3.12-2, Item 1, "Acceptance Criteria," states that the thickness of RWB radiation shielding barriers is greater than or equal to the required thickness specified in FSAR Tier 1, Table 3.12-1, "Radioactive Waste Building Shield Wall Geometry." FSAR Tier 2, Section 12.3.2.2, states that FSAR Tier 2, Table 12.3-6 and Table 12.3-7, show the nominal shielding thicknesses for rooms in the RXB and the RWB, respectively. FSAR Tier 2, Table 12.3-6, provides the nominal thickness of concrete for some of the walls in the RXB. FSAR Tier 2, Table 12.3-7, provides the nominal thickness of concrete for some of the walls in the RWB. As given in the Merriam-Webster dictionary, the definitions of "nominal" include being or relating to a designated or theoretical size that may vary from the actual (i.e., approximate) or existing, or being something in name or form only. Because the NRC staff did not understand what requirements the applicant was applying to the radiation shielding, it issued RAI 9279 (ML18134A244), Question 12.03-58, asking the applicant to provide additional information on the minimum specified shielding thicknesses and describe how the specified minimum shielding thicknesses compared to the shielding thicknesses used in the specific shielding analyses. The NRC staff is seeking to understand the difference between "minimal thickness" in FSAR Tier 1, Sections 3.11 and 3.12, and "nominal thickness" in FSAR Tier 2, Section 12.3.2.2. In addition, the NRC staff asked NuScale to describe and explain any differences between the thickness used in the shielding analysis packages for the RXB and shielding thicknesses described in FSAR Tier 1, Section 3.11. In its response dated June 5, 2018, the applicant clarified the thicknesses used and removed the use of the ambiguous descriptor "nominal." It also noted that shielding values provided in FSAR Tier 2, Tables 12.3-6 and 12.3-7, are the same values used in the shielding analysis for developing the radiation zone map; therefore, the NRC staff found this response and the proposed FSAR Tier 2 changes acceptable. **The NRC staff is currently tracking RAI 9279, Question 12.03-58, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

With regard to the radiation zone classification of the Class A, B, and C HIC storage area (shown in FSAR Tier 2, Revision 0, Figure 1.2-30), the NRC staff issued RAI 8860, Question 12.02-2 (ML17157B493), asking the applicant to identify whether drums from the drum dryer facility were allowed to be stored within the Class A, B, and C HIC storage area (SER Section

12.2 contains additional information). In its response to RAI-8860, Question 12.02-2, dated July 10, 2017 (ADAMS Accession No. ML17191B211), the applicant stated that drums from the drum dryer facility are not stored in the Class A, B, and C HIC storage area but rather in the mixed/chemical waste drum storage area (FSAR Tier 2 Figure 1.2-30, Room 030-007). The review of the response to RAI 8860, Question 12.02 2 (ADAMS Accession No. ML17191B211), is documented in SER Section 12.2, above. Based on the NRC staff review, as discussed SER Section 12.2, the NRC staff considers RAI 8860, Question 12.02 2 closed/unresolved. FSAR Tier 2, Revision 0, Section 11.2, notes that the drum dryer consists of a system designed to pump water into a 55-gallon drum (nominally 7.4 cubic feet (ft^3), which is heated and evacuated to rapidly evaporate the liquid in the drum until only solid material remains in the drum. The remaining concentrate contains all the nonvolatile radioactive material added to the drum, which, in turn, serves as the basis for establishing the dose rates near the drums. FSAR Tier 2 Table 12.2-13b lists the quantities of the isotopes expected to be present in a drum. FSAR Tier 2 Table 12.3-1 states that an area defined as Radiation Zone 3 has dose rates greater than or equal to 2.5 mrem/h (0.025 mSv/h) and less than or equal to 5 mrem/h (0.05 mSv/h). FSAR Tier 2, Revision 0, Figure 1.2-28, shows the drum storage area (Room 030-007) in the RWB. FSAR Tier 2 Figure 12.3-2a shows the mixed/chemical waste storage area, which corresponds to the drum storage area in FSAR Tier 2 Figure 1.2-28 (Room 030-007), as Radiation Zone III (i.e., dose rates greater than or equal to 2.5 mrem/h (0.025 mSv/h) and greater than or equal to 5 mrem/h (0.05 mSv/h). A NRC staff analysis indicated that the dose rate on a drum of dried liquid containing the amount of radioactive material listed in FSAR Tier 2 Table 12.2-13b may exceed the indicated radiation zone depicted on FSAR Tier 2 Figure 12.3-2a for the drum storage area by several orders of magnitude. Therefore, the NRC staff issued RAI 9280, Question 12.03-04 (ML18004A057), asking the applicant to provide additional information on the storage location for dried drums. In addition, the NRC staff issued RAI 9280, Question 12.03-05 (ML18004A057), asking the applicant to provide additional information on the methods, models, and assumptions that it used to establish the stated radiation zone in the drum storage room. The NRC staff also issued RAI 9280, Question 12.03-06 (ML18004A057), asking the applicant to provide additional information on the bounding values of radioactive material stored in the drum storage room, which is necessary to maintain the area within the designated radiation zone and to not impact the radiation zone designations of adjacent areas.

In its response to RAI 9280, Questions 12.03-04, 12.03-05, and 12.03-06, dated August 21, 2018 (ADAMS Accession No. ML18233A230), the applicant provided information about the drum dryer tank contents and storage location for drum dryer waste. It also gave the assumed source term for a drum dryer and the drum contents that it generates. In addition, the applicant stated that the waste generated from the drum dryers will be stored inside the Class A, B, and C HIC storage area. Based on the information in the applicant's response, the NRC staff was able to perform confirmatory calculations to verify the radiation zoning specified by the applicant, as described in the applicant's response to RAI 9269, to support further NRC staff evaluations. As a result, the NRC staff finds the applicant's response to RAI 9280, Questions 12.03-04, 12.03-05, and 12.03-06, acceptable because the applicant described enough information in its response to allow the NRC staff to develop a confirmatory calculation to verify the radiation zoning specified by the applicant in its design. Although the NRC staff does not agree with the applicant's rationale with respect to the presence of radioactive material in a mixed waste drum (i.e., the definition of mixed waste), the impact on the radiation zone designations and the associated radiological significance is minimal, and there is no regulatory compliance concern; therefore, the NRC staff finds the applicant's responses to RAI 9280, Questions 12.03-05 and 12.03-06, acceptable. However, the NRC staff found that the proposed changes to FSAR Tier 2 Section 12.2.1.7 in the response to RAI 9280, Question 12.03-4, indicated that FSAR Tier 2 Tables 12.2-15b and 12.2-16b provide the radionuclide inventory and source strengths for the drum dryer. However, in reviewing the tables in FSAR Tier 2 Section 12.2 and the applicant's response to RAI 9256, the NRC staff found that FSAR Tier 2 Tables 12.2-13b and 12.2-14b (not FSAR Tier 2 Tables 12.2-15b and 12.2-16b) provided the drum dyer source term. During a September 27, 2018, public teleconference, the applicant indicated that it would modify the table numbers in FSAR Tier 2 Revision 2 to match the information provided in its response to RAI 9280, Question 12.03-4. Therefore, **the NRC staff is currently tracking RAI 9280, Question 12.03-04, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision, and RAI 9280, Questions 12.03-05 and 12.03-06, are closed.

FSAR Tier 2, Revision 0, Section 12.2.1.4, "Reactor Pool Cooling, Spent Fuel Pool Cooling and Pool Cleanup Systems," states that the PCUS draws water from either the SFP cooling system or the reactor pool cooling system and removes impurities to reduce radiation exposures and to maintain water chemistry and clarity. The reactor pool cooling system and SFP cooling system heat exchangers are conservatively assumed to be filled with reactor pool water even though the shell side is normally filled with site cooling water. FSAR Tier 2, Revision 0, Figure 12.3-1f, "Reactor Building Radiation Zone Map—86' Elevation," shows the radiation zone for spent fuel pool heat exchanger (SFPHX) as Radiation Zone 0. FSAR Tier 2 Table 12.3-1 shows that areas designated as Radiation Zone 0 have dose rates greater than or equal to 0.05 mrem/h (0.0005 mSv/h). However, the results of the analysis performed by the NRC staff that included radiation contributions to the dose rate in the heat exchanger cell from isotopes, such as I-131, Cs-134, Cs-137, and Co-60, described in the application and information made available to the NRC staff as part of the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), appeared to differ by about a factor of 10 from those specified in the application. Because the NRC staff was unable to determine the bases for the designated radiation zone, it issued RAI 9281, Question 12.03-56 (ML18120A372), asking the applicant to provide additional information on the methods, models, and assumptions that it used to develop the assumed dose rate used as the bases for the radiation zone designation of this area. In its response to RAI 9281, Question 12.03-56, dated August 22, 2018 (ADAMS Accession No. ML18225A249), NuScale updated the radiation zoning for the SFPHX room. During normal operations (not during resin transfers), the SFPHX room was previously designated Radiation Zone 0. The RAI response updated the SFPHX room to Radiation Zone II. Because the applicant provided a radiation zone designation commensurate with NRC staff expectations of the radiological conditions during normal operation, the NRC staff finds this response acceptable. **The NRC staff is currently tracking RAI 9281, Question 12.03-56, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The NRC staff notes that the applicant included additional changes unrelated to the resolution of RAI 9281, Question 12.03-56 in the same RAI response document. The NRC staff is reviewing and tracking these additional, unrelated changes under the RAI numbers related to those questions.

FSAR Tier 2 Section 12.3.1.3.1 provides information on controls and design features for VHRA and specifies that VHRAs either are locked or have alarmed barriers. It also provides COL Item 12.3-2, which specifies that the COL applicant will develop administrative controls for access to VHRAs in accordance with the guidance of RG 8.38. FSAR Tier 2 Table 12.3-3 only identifies one VHRA in the NuScale FSAR Tier 2 (the Class A, B, and C HIC room in the RWB). The acceptance criteria in DSRS Section 12.3–12.4 state that 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 500 rads (5 grays) or more in 1 hour at 1 meter 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 CNVs of shutdown reactors). Physical controls for known VHRA areas are evaluated as part of the design review. The statement in the FSAR Tier 2 that it is optional for VHRAs to be locked appears potentially inconsistent with 10 CFR 20.1602 and RG 8.38. Therefore, the NRC staff issued RAI 9290, Question 12.03-21 (ML18026A726), asking the applicant to provide additional information on the requirements for locking VHRA areas. FSAR Tier 2, Revision 0, Section 12.3–12.4, does not describe the design features or the requirement of the design that prevent personnel from being locked in a high-radiation area in accordance with 10 CFR 20.1601(d). Therefore, the NRC staff issued RAI 9290, Question 12.03-22 (ML18026A726), asking the applicant to provide additional information on the design features provided to prevent an individual from being inadvertently locked in a high-radiation area. In its response to RAI 9290, Questions 12.03-21 and 12.03-22, dated February 16, 2018 (ADAMS Accession No. ML18047A189), the applicant stated that the only VHRA is the Class A, B, and C HIC room on the 71-foot elevation of the RWB. The applicant also stated that the door to enter this room is locked to prevent unauthorized access. In addition, the applicant provided FSAR Tier 2 text to clarify the control access to VHRA. The NRC staff finds the response to RAI 9290, Questions 12.03-21 and 12.03-22, acceptable because the applicant clarified the text in FSAR Tier 2 Section 12.3.1.3.1 to be consistent with RG 8.38 and because it meets the requirements of 10 CFR 20.1602 by specifying controls upon entry while not impeding the exit from this VHRA. The NRC staff considers RAI 9290, Questions 12.03-21 and 12.03-22, closed.

FSAR Tier 2, Revision 0, Section 12.3.2, describes some of the design considerations for radiation shielding (e.g., stating that concrete is the material used for a significant portion of plant shielding). FSAR Tier 2 Section 12.3.2.2 states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. FSAR Tier 2 Section 12.3.2.4.3 states that cubicle walls are made of concrete supported by carbon steel plates called "structural steel partition walls." FSAR Tier 2 Table 12.3-6 provides the nominal thickness of concrete for some of the walls in the RXB. FSAR Tier 2 Table 12.3-8 lists the shielded doors located in the RXB. FSAR Tier 2 Table 12.3-7 provides the nominal thickness of concrete for some of the walls in the RWB. FSAR Tier 2 Table 12.3-9 lists the shielded doors located in the RWB. The NRC staff used information from the RPAC Chapter 12 Phase 1 audit to identify some shielding design calculations that referenced the use of additional steel (i.e., in addition to the structural steel partition walls already noted) shielding to limit dose rates in adjacent areas. FSAR Tier 2, Revision 0, Section 12.3.2, does not identify the specific areas where additional shielding is required. FSAR Tier 2 Table 12.3-6 and FSAR Tier 2 Table 12.3-7 provide the nominal thickness of concrete for some of the walls in the RXB and RWB. However, neither table identifies the location or the minimum thickness of any additional steel shielding material. Therefore, the NRC staff issued RAI 9294, Question 12.02-23 (ML18026A727), asking the applicant to describe the locations where additional steel shielding material is credited and the associated methods, models, and assumptions used to establish the identified amount of shielding material at those locations. In its response to RAI 9294, Question 12.03-23 (ADAMS Accession No. ML18080A127), the applicant specified that the design of various walls within the NuScale facility includes the use of steel plate lined concrete. Specifically, the 20-inch concrete and steel partition walls within the RXB consist of two ½-inch steel plates with 19 inches of concrete in between them. The 20-inch concrete and steel composite slabs for floors and ceilings also consist of two ½-inch steel plates with 19 inches of concrete in between them. The applicant proposed adding footnotes to FSAR Table 12.3-6 describing the designed thickness of the steel and concrete for the partition walls and composite slabs.

In addition, the applicant specified that an additional 1-inch-thick steel plate is modeled to cover the LRWS ion exchange and charcoal bed cubicle, and an additional 2-inch-thick plate of steel is modeled to cover the drum dryer skid cubicles. In addition, the applicant stated that the shielding analysis of the solid radioactive waste system models a 4.5-inch-thick lead shield around the HIC in the HIC fill station room. The applicant stated that it would revise FSAR Tier 2 Section 12.3.2.4.4, "Radioactive Waste Building," and Table 12.3-7 to reflect these additional shields. However, in the FSAR Tier 2 markups, it was not clear to the NRC staff whether this shielding will be provided in the design or whether it will just be modeled in the shielding calculations. In Revision 1 of its response to RAI 9294, Question 12.02-23 (ADAMS Accession No. ML18228A861), the applicant updated FSAR Tier 2 Table 12.3-7 to clarify that the shielding specified above (or an equivalent amount of attenuation to the shielding specified above) will be provided on the LRWS, drum dryer skid cubicles, and HICs. The NRC staff finds the response acceptable because the steel shielding being credited in the design will be specified in the FSAR Tier 2. **As a result, the applicant's response to RAI 9294, Question 12.02-23, is being tracked as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

FSAR Tier 2 Section 12.3.2.2 states that radiation shield doors are designed to have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed. FSAR Tier 2, Revision 0, Section 12.3.2, does not identify the specific areas where additional shielding is required. FSAR Tier 2 Tables 12.3-8 and 12.3-9 list the radiation shield doors located in the RXB and RWB. However, neither table identifies the location or the minimum thickness of any additional steel shielding material that is used in the rooms that the doors are shielding; therefore, it is not clear to the NRC staff which "shielding thickness" applies to the radiation shielding doors (i.e., just the concrete shielding or the concrete shielding plus any other shielding enhancements). Therefore, the NRC staff issued RAI 9294, Question 12.03-24 (ML18026A727), asking the applicant to describe the locations where additional steel shielding material is credited and describe how that information is used to determine the required thickness of radiation shield doors. In its response to RAI 9294, Question 12.03-24 (ADAMS Accession No. ML18080A127), the applicant reiterated that the design thicknesses for the shield doors will provide an equivalent (or greater) radiation attenuation as the surrounding wall in which the shield door is located and that the shield walls are provided in FSAR Tables 12.3-6 and 12.3-7. As discussed above in its response to RAI 9294, Question 12.03-23, the applicant added footnotes to FSAR Tables 12.3-6 and 12.3-7 describing the use of steel shielding in the design. Because the radiation shield doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed and because the walls with steel shielding and the thicknesses of the steel and concrete shielding are being used on those walls are now specified in the FSAR, the amount of radiation attenuation required by each door is now clear. As a result, the response is acceptable, and RAI 9294, Question 12.03-24, is closed.

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 rads/h) should be shielded and clearly marked with a sign stating that potentially lethal radiation fields are possible. If removable shielding is used to reduce dose rates to less than 1 Gy/h (100 rads/h), it must also be explicitly marked as noted above. FSAR Tier 2, Revision 0, Sections 12.3.2.4.3 and 12.3.2.4.4, identify a number of areas (e.g., resin demineralizers, filters, SRSTs) that may contain quantities of radioactive material resulting in radiation dose rates exceeding 100 rads/h (1 Gy/h). However, FSAR Tier 2, Revision 0, Section 12.3.2, does not identify the specific areas where removable shielding is used. FSAR Tier 2 Tables 12.3-6, 12.3-8, and 12.3-7 provide the nominal thickness of concrete for some of the walls in the RXB and RWB. However, neither table identifies the location of removable shielding material. FSAR Tier 2, Revision 0, Section 12.3.2, does not specify that those portions of the facility capable of having radiation levels greater than 1 Gy/h (100 rads/h) where removable shielding is used are clearly marked with a sign that states that potentially lethal radiation fields are possible. Therefore, the NRC staff issued RAI 9294, Question 12.03-25 (ML18026A727), asking the applicant to describe the locations where removable shielding material is credited for the radiation shielding design and for those areas with potential dose rates exceeding 100 rads/h (1 Gy/h) and describe how the areas are marked. In its response to RAI 9294, Question 12.03-25 (ADAMS Accession No. ML18080A127), the applicant specified that removable shielding in the RXB in the NuScale design includes the bioshield over each power module and floor shield plugs for the CVCS particulate filters and resin traps. In the RWB, removable shielding includes shield plugs for the HIC storage room and HIC filling room. In the response to RAI 9294 Question 12.03-25, the applicant proposed that FSAR Section 12.3.2.4.3 would describe the shielding plugs at an elevation of 35 feet, 8 inches in the RXB for accessing the CVCS filters and resin traps for maintenance purposes. The applicant specified that appropriate postings and signage will be affixed, as determined by the radiation protection program (COL item 12.5-1). Therefore, the NRC staff finds this portion of the applicant's response to RAI 9294, Question 12.03-25, to be acceptable.

However, although the FSAR Tier 2 (with the proposed changes provided in the RAI response) specifies that the design includes removable shield plugs for the CVCS particulate filters and resin traps, the HIC filling area, and the HIC storage area, it does not provide any information on the radiation attenuation capabilities of the removable shielding. In Revision 1 of the applicant's response to RAI 9294, Question 12.03-25 (ADAMS Accession No. ML18228A861), the applicant proposed to update FSAR Tier 2 Section 12.3 to specify that the shield floor plugs provide an equivalent radiation attenuation as the shield floor that contains the plug. The NRC staff finds that it is appropriate for the shield plugs to have the same equivalent attenuation capability as that of the floor, and they will limit worker exposure in surrounding low dose rate areas, including the truck bay, which is located at the elevation above the HIC filling and storage rooms. As a result, the NRC staff finds the applicant's response to be acceptable. **Therefore, RAI 9294, Question-12.03-25, is being tracked as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

The acceptance criteria in DSRS Section 12.3–12.4 states that an assessment of design features provided to protect shielding material subject to degradation, such as through the effects of radiation (e.g., depletion of boron neutron absorbers), temperature extremes (e.g., degradation of polymer-based materials caused by high temperature), density changes (e.g., sagging or settling of shielding material with age) 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 shielding material. FSAR Tier 2, Section 12.3.1.2.3, "Penetrations," states that, if penetrations through shield walls 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, penetrations are either shielded or elevated above floor level. FSAR Tier 2 Section 12.3.2.2 states that, in addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. However, FSAR Tier 2, Revision 0, Section 12.3.2, does not identify any areas of the plant shielding (e.g., penetration shielding around hot pipes) that have limitations associated with the shielding material or for which specific design criteria (e.g., maximum temperature, radiation resistance) are required to ensure that the integrity of the shielding is maintained. Therefore, the NRC staff issued RAI 9294, Question 12.03-26

(ML18026A727), asking the applicant to describe (1) the locations in the RXB and RWB where the integrity of the radiation shielding material could be challenged by the local environmental conditions, (2) the locations in the RXB and RWB where shielding material other than steel or concrete is used, (3) any required controls to ensure the continued integrity of the shielding material, and (4) the corresponding COL item, if the COL applicant is expected to have a program for that purpose.

In its response to RAI 9294, Question 12.03-26 (ADAMS Accession No. ML18080A127), the applicant indicated that it had modified the bioshield design to remove borated polyethylene from the design and that it had already modified FSAR Tier 2 Revision 1 and Chapter 3 to reflect the removal of the polyethylene. The NRC staff noted that the applicant included proposed modifications to FSAR Tier 2 Table 12.3-6 that showed the removal of the polyethylene from that table in its response to RAI 9298, Question 12.03-17. The NRC staff evaluates these changes in its review of the applicant's response to RAI 9298, Question 12.03-17. In its response to RAI 9294, Question 12.03-26, the applicant specified that the radiation shielding design details and materials related to items such as shield wall penetration shielding have not been finalized. It also specified that the COL applicant will be responsible for the details of the design, testing, and inspection of potentially degradable shielding materials. However, the FSAR Tier 2 did not provide any COL information item or other information on shielding design details and potential degradation caused by environmental conditions that the COL applicant would need to provide. The NRC staff also noted that this issue is related to RAI 9282 on neutron fluence and spectrum because the neutron spectrum will affect the dose rates at the penetrations. Additionally, the issue is related to RAI 9295, Question 12.03-55, which requests additional information on the shielding for the penetrations for the RXB pool wall. Based on the unresolved issues discussed above, **RAI 9294, Question 12.03-26, remains an open item**.

FSAR Tier 1, Revision 0, Section 3.11, and FSAR Tier 1, Section 3.12, contain the ITAAC related to the radiation shielding. FSAR Tier 1, Section 3.11, states that the RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding. It further states that FSAR Tier 1, Table 3.11-2, contains the ITAAC for the RXB. FSAR Tier 1, Table 3.11-1, Item 4, states that the thickness of RXB radiation shielding barriers is greater than or equal to the required thickness specified in FSAR Tier 1, Table 3.11-1. FSAR Tier 1, Section 3.11, further states that the RXB includes radiation-attenuating doors for normal operation and post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed. FSAR Tier 1, Section 3.12, states that the RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding. In addition, the RWB includes radiation-attenuating doors for normal operation and for post-accident radiation shielding. These doors also have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed. FSAR Tier 1, Section 3.12, further states that FSAR Tier 1, Table 3.12-2, contains the ITAAC for the RWB. FSAR Tier 1, Table 3.12-2, Item 1, states that the thickness of RWB radiation shielding barriers is greater than or equal to the required thickness specified in FSAR Tier 1, Table 3.12-1.

FSAR Tier 2, Section 12.3.2.2, states that FSAR Tier 2, Tables 12.3-6 and Table 12.3-7, show the nominal shielding thicknesses for rooms in the RXB and the RWB, respectively. FSAR Tier 2, Table 12.3-6, provides the nominal thickness of concrete for some of the walls in the RXB. FSAR Tier 2, Table 12.3-7, provides the nominal thickness of concrete for some of the walls in the RWB. Because the NRC staff was unable to determine whether the ITAAC for radiation shielding encompassed the different types and associated thicknesses of material needed to meet the underlying requirement for shielding, the NRC staff issued RAI 9294, Question 12.03-27 (ML18026A727), asking the applicant to provide additional information on the ITAAC for radiation shielding.

As discussed above, in its response to RAI 9294, Question 12.03-23 (ADAMS Accession No. ML18080A127), the applicant specified that 20-inch concrete and steel partition walls within the RXB consist of two ½-inch steel plates with 19 inches of concrete in between them. In addition, the 20-inch concrete and steel composite slabs for the floors and ceilings consist of two ½-inch steel plates with 19 inches of concrete in between them. The applicant proposed adding footnotes to FSAR Table 12.3-6 describing the designed thickness of the steel and concrete for the partition walls and composite slabs. The applicant also specified that an additional 1-inch-thick steel plate is modeled to cover the LRWS ion exchange and charcoal bed cubicle and that an additional 2-inch-thick plate of steel is modeled to cover the drum dryer skid cubicles. In addition, the shielding analysis of the solid radioactive waste system models a 4.5-inch-thick lead shield around the HIC in the HIC fill station room. The applicant will revise FSAR Section 12.3.2.4.4 and Table 12.3-7 to reflect these additional shields.

Consistent with the applicant's response to RAI 9294, Question 12.03-23, in the response to RAI 9294, Question 12.03-27 (ADAMS Accession No. ML18080A127), the applicant proposed to update FSAR Tier 1, Tables 3.11-1 and 3.12-1, to provide the same information on steel shielding in FSAR Tier 1. However, in the FSAR Tier 2 markups, it was not clear to the NRC staff whether this shielding will be provided in the design or whether it will just be modeled in the shielding calculations. In Revision 1 of its response to RAI 9294, Question 12.03-27 (ADAMS Accession No. ML18228A861), the applicant updated FSAR Tier 2 Table 12.3-7 to clarify that the shielding specified above (or an equivalent amount of attenuation to the shielding specified above) will be provided on the LRWS, drum dryer skid cubicles, and HICs. The NRC staff finds the applicant's response to RAI 9294, Question 12.03-27, acceptable because the steel shielding that is being credited in the design will be specified in the FSAR Tier 2. **As a result, the applicant's response to RAI 9294, Question 12.03-27, is being tracked as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

FSAR Tier 2, Revision 0, Figure 12.3-1g, "Reactor Building Radiation Zone Map—100' Elevation," shows that the area between column lines RX4 and RX6, column lines RXA and RXB (north of the reactor pool), and column lines RXD and RXE (south of the reactor pool), depicted on FSAR Tier 2 Figure 1.2-216, "Reactor Building 100'-0" Elevation," as steam galleries, (Rooms 010-411 and 010-418, respectively) is labeled as Radiation Zone 0. FSAR Tier 2, Revision 0, Table 12.3-1, "Normal Operation Radiation Zone Designations," shows that areas designated as Radiation Zone 0 have dose rates less than or equal to 0.05 mrem/h (0.0005 mSv/h). FSAR Tier 2, Revision 0, Figure 3.6-16, "Postulated High-Energy Main Steam System Pipe Routing beyond the NuScale Power Module (COL Applicant Scope)," and Figure 3.6-17, "Postulated High-Energy Feedwater System Pipe Routing beyond the NuScale Power Module (COL Applicant Scope)," show the main steam and main feed water lines passing through the RXB pool wall (RXD or RXB). NuScale FSAR Tier 2, Revision 0, Table 3C-6, provides the 60-year integrated N dose (rad) for the area outside of the CNV and under the bioshield. FSAR Tier 2 Figure 12.3-1g depicts the areas under the bioshield as Radiation Zone VI (dose rates greater than or equal to 1 rad/h (0.01 Gy/h) and less than or equal to 500 rad/h (5 Gy/h) from FSAR Tier 2 Table 12.3-1). FSAR Tier 2 Section 12.3.2 does not discuss any specific shielding features between the NPM bay under the bioshield and the steam galleries to prevent neutron streaming into the steam galleries (Rooms 010-411 and 010-418). FSAR Tier 2 Section 12.3.2.2 does not discuss any considerations for the types of shielding material or the environmental controls provided to ensure shielding integrity and for shielding material that may be installed around the high-temperature main steam and main feed water lines. FSAR Tier 2 Table 12.3-6 does not discuss the neutron fluence from under the bioshield area as a shielding consideration for the steam galleries. Therefore, the NRC staff is unable to substantiate the applicant’s designation of the steam galleries as Radiation Zone 0. Therefore, the NRC staff issued RAI 9295, Question 12.03-55 (ML18120A369), asking the applicant to provide additional information on the methods, models, and assumptions that it used to establish steam gallery radiation zones depicted in the FSAR Tier 2. In a supplementary response to RAI 9295, Question 12.03-55 (ADAMS Accession No. ML18235A648), the applicant provided COL Item 12.3-8 for the COL applicant to describe the radiation shielding design measures used to compensate for the main steam and the main feed water piping penetrations through the RXB pool wall between the NuScale NPM bays and the RXB steam galleries near the 100-foot elevation. Currently, the applicant has not submitted detailed drawings to the NRC staff that would identify other major penetrations, such as ventilation, that would need a similar COL item. In addition, the applicant did not provide a design interface in accordance with 10 CFR 52.47(a)(25) and (26). As a result, **the NRC staff is currently tracking RAI 9295, Question 12.03-55, as an open item**.

FSAR Tier 2, Revision 0, Figure 12.3-1g, shows the area above the reactor pool area (Room 010-022 on FSAR Tier 2 Figure 1.2-216) as Radiation Zone II. FSAR Tier 2, Revision 0, Table 12.3-1, shows that areas designated as Radiation Zone II have dose rates greater than or equal to 0.25 mrem/h 0.0025 mSv/h) and less than or equal to 2.5 mrem/h (0.025 mSv/h). FSAR Tier 2, Revision 0, Table 12.2-10, provides the radionuclides concentration in the reactor pool water. The NRC staff’s independent calculations of the dose rate from the UHS pool water using the radionuclide concentrations listed in the column “Reactor Pool Water (µCi/gram)” in FSAR Tier 2 Table 12.2-10 appeared to be inconsistent with the radiation zone assigned to Room 010-022 in FSAR Tier 2 Figure 12.3-1g. Therefore, the NRC staff issued RAI 9296, Question 12.03-60 (ML19015A274), asking the applicant to provide additional information on the methods, models, and assumptions that it used to establish the UHS radiation zones depicted in the FSAR Tier 2. In its response to RAI 9296, Question 12.03-60, dated June 11, 2018 (ADAMS Accession No. ML18162A353), the applicant provided information that supported the current radiation zone for Room 010-022 in FSAR Tier 2 Figure 12.3-1g. The applicant noted in its response that the radiation zoning was consistent with areas accessible to personnel. The NRC staff’s confirmatory calculations support this assertion in that dose rates at the edges of the pool agree with the current radiation zone. In addition, although the NRC staff’s confirmatory calculations note that dose rates that are 1 meter above the pool exceed the limits for the radiation zone, this area is effectively inaccessible during routine operations. The area of concern above the pool would be on the refueling bridge, which is approximately 10 feet above that pool and is lined with ¼-foot steel. The NRC staff’s confirmatory calculations support the applicant’s response that the dose rates on the refueling bridge are within the bounds of the current FSAR Tier 2 radiation zoning; therefore, RAI 9296, Question 12.03-60, is closed.

FSAR Tier 2, Revision 0, Figure 12.3-1g, shows the area above the reactor pool area (Room 010-022 on FSAR Tier 2 Figure 1.2-216) as Radiation Zone II. FSAR Tier 2, Revision 0, Table 12.3-1, shows that areas designated as Radiation Zone II have dose rates great than or equal to 0.25 mrem/h (0.0025 mSv/h) and less than or equal to 2.5 mrem/h (0.025 mSv/h). FSAR Tier 2, Revision 0, Section 12.3.2.4.1, “NuScale Power Module,” states that FSAR Tier 2, Section 3.7.3, describes the bioshield design. FSAR Tier 2, Revision 0, Table 3.7.3-12, “Bioshield Face Plate Self-Weight,” and FSAR Tier 2, Revision 0, Figure 3.7.3-2, “Conceptual Bioshield Vertical Face Plate,” depict the bioshield face plate as a hollow space covered by two ¼-inch steel plates. In the NRC staff’s view, the ½-inch steel plating described in Table 3.7.3-12 represents a minimal amount of neutron, or gamma, shielding material. FSAR Tier 2, Revision 0, Table 3C-6, provides the 60-year integrated N dose (rad) for the area outside of the CNV and under the bioshield. FSAR Tier 2 Figure 12.3-1g depicts the areas under the bioshield as Radiation Zone VI (dose rates greater than or equal to 1 rad/h (0.01 Gy/h) and less than 500 rad/h (5 Gy/h) from FSAR Tier 2 Table 12.3-1). Because there is a minimal amount of radiation-attenuating material between Radiation Zone VI located under the bioshield area and the area above the reactor pool area (Room 010-022), the NRC staff was unable to verify the applicant's basis for its designation of this area (Room 010-022) as Radiation Zone II. Therefore, the NRC staff issued RAI 9297, Question 12.03-59 (ML18134A246), asking the applicant to provide additional information on the methods, models, and assumptions that it used to establish the radiation zones within the RXB. In its response to RAI 9297, Question 12.03-59 (ADAMS Accession No. ML18233A472), the applicant stated that, because there were no accessible areas above the pool water and because no personnel were expected to be adjacent to an operating NPM at the pool level, there was no need to provide a radiation zone designation. Because the radiation zone designations are used to establish radiation protection design features for the protection of personnel and equipment and because there is no equipment or personnel in this area, the NRC staff found the applicant's response to RAI 9297, Question 12.03-59, acceptable, therefore, the NRC staff considers RAI 9297, Question 12.03-59, to be closed.

FSAR Tier 2, Revision 0, Section 12.3.2.2, states that, in addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. FSAR Tier 2 Table 12.3-6 identifies the only use of polyethylene, and it provides the nominal thickness of concrete for some of the walls in the RXB. FSAR Tier 2 Table 12.3-8 lists the shielded doors in the RXB. FSAR Tier 2 Table 12.3-9 lists the shielded doors in the RWB. FSAR Tier 2 Section 12.3 does not contain any information on the assumption for concrete density other than the references to ANSI/ANS Std. 6.4-2006 and PNNL-25870. The NRC staff used information from the RPAC Chapter 12 Phase I audit (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), to review some of the shielding calculation information for the RXB and RWB. The NRC staff noticed that the polyethylene shielding specified for the bioshield cover is high-density polyethylene and includes 5-percent natural boron. FSAR Tier 2, Sections 12.3.2, 12.3.2.3, and 12.3.2.4.3, do not specify that the polyethylene is supposed to contain boron; they also do not specify the values of key assumptions, such as the minimum polyethylene density or the minimum w% of boron in the polyethylene documented. Therefore, the NRC staff issued RAI 9298, Question 12.03-17 (ML18026A724), asking the applicant to provide additional information on the shielding material used in the design, including the methods, models, and assumptions that is used to establish the stated radiation zones within the RXB. In its response dated August 22, 2018 (ADAMS Accession No. ML18234A445), the applicant indicated that high-density polyethylene will not be used in the reactor pool area floor and that it has removed the 2 inches of high-density polyethylene from the design and replaced it with an additional 2 inches of concrete. The applicant also added 0.5 inch of steel. However, because the applicant did not provide any information on the changes in dose rates above the bioshield area resulting from the deletion of borated polyethylene, **the NRC staff is currently tracking RAI 9298, Question 12.03-17, as an open item**.

FSAR Tier 1, Section 3.11, states that the RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding. It further states that FSAR Tier 1, Table 3.11-2, contains the ITAAC for the RXB. FSAR Tier 1, Table 3.11-1, Item 4, states that the thickness of RXB radiation shielding barriers is greater than or equal to the required thickness specified in

FSAR Tier 1, Table 3.11-1. However, FSAR Tier 1, Table 3.11-1, does not specify the boron content of the polyethylene or describe the minimum polyethylene density. Therefore, the NRC staff issued RAI 9298, Question 12.03-18 (ML18026A724), asking the applicant to provide additional information on the borated polyethylene shielding material used in the design, including the methods, models, and assumptions that it used to establish the stated radiation zones within the RXB. In its response dated August 22, 2018 (ADAMS Accession No. ML18234A445), the applicant indicated that high-density polyethylene will not be used in the reactor pool area floor and that it has removed the 2 inches of high-density polyethylene from the design and replaced it with an additional 2 inches of concrete. The applicant also added 0.5 inch of steel. Because borated polyethylene is no longer at this location, this ITAAC is no longer necessary. Therefore, the NRC staff considers RAI 9298, Question 12.03-18 closed.

The acceptance criteria in DSRS Section 12.3–12.4 note that, where the applicant's shielding design incorporates material subject to degradation such as through the effects of radiation (e.g., depletion of boron neutron absorbers) or temperature extremes (e.g., degradation of polymer-based materials caused by high temperature), methods are in place to ensure the integrity of the shielding. DSRS Section 12.3–12.4 also describes how the application identifies the constraints (e.g., minimum cooling airflow, maximum shielding material temperature, and maximum allowable neutron flux) and how those parameters are measured and assessed over the design life of the facility. Therefore, the NRC staff issued RAI 9298, Question 12.03-19 (ML18026A724), asking the applicant to provide additional information on the design features and required constraints needed to ensure the integrity of the shielding over the design life of the plant. The NRC Staff also asked the applicant to identify, the control mechanisms proposed to ensure that the environmental conditions remained within the constraints needed to ensure the integrity of the radiation shielding over the design life of the plant. The NRC asked the applicant to include in their response a description of these parameters and the associated controls that should be implemented by the COL Licensee (e.g., COL item). In its response dated August 22, 2018 (ADAMS Accession No. ML18234A445), the applicant indicated that high-density polyethylene will not be used in the reactor pool area floor and that it has removed the 2 inches of high-density polyethylene from the design and replaced it with an additional 2 inches of concrete. The applicant also added 0.5 inch of steel. Based on the NRC staff review, because borated polyethylene, which is subject to environmentally induced and radiation induced degradation, is no longer at this location, the requested ITAAC is no longer necessary, and the additional controls needed to ensure degradation of radiation shielding material are no longer required. Therefore, the NRC staff considers RAI 9298, Question 12.03-19 closed.

FSAR Tier 2, Revision 0, Section 12.3.2.2, states that, in addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. However, FSAR Tier 2 Table 12.3-6 identifies only polyethylene as a shielding material other than concrete. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I (see "Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design," (ADAMS Accession No. ML18124A182)), audit, it appears that the shielding design also credits a number of steel/iron plates that are provided for uses other than structural support. Therefore, the NRC staff issued RAI 9298, Question 12.03-20 (ML18026A724), asking the applicant to provide additional information on the use of various shielding materials, other than concrete, credited in the shielding analyses and on the management of plant operation to maintain the integrity of the radiation shielding over the design life of the plant. Because FSAR Tier 2 Section 12.3 Revision 1, does not identify specific locations where radiation shielding material that is subject to environmentally induced or radiation induced degradation, is in use, the NRC staff finds this response acceptable. Therefore, the NRC staff considers RAI 9298, Question 12.03-20 closed.

FSAR Tier 2, Revision 0, Section 12.2.1.1, “Reactor Core,” states that FSAR Tier 2 Table 12.2-1 provides the fission neutron, n-gamma, and fission gamma source strength and neutron energy spectrum information. However, NuScale TR-0116-20781-P, Revision 0, does not provide any information about photon/gamma strength or spectrum information. FSAR Tier 2 Table 12.2-1 gives values for the fission neutron source strength and the fission neutron n-gamma. The NRC staff is not clear on the intended difference between the two listed parameters and how each value is expected to be used. The neutron flux and energy spectrum listed in FSAR Tier 2 Table 12.2-1 are not well defined, nor does the FSAR Tier 2 explain the derivation of the stated values. Based on information made available to the NRC staff during the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the NRC staff was not able to characterize the neutron radiation fields in the areas mentioned above. The NRC staff needs to know the neutron flux and energy spectrum and to have the appropriate supporting information to evaluate and confirm activation products and resulting dose rates in the FSAR. Therefore, the NRC staff issued RAI 9300, Question 12.03-28 (ML18028A002), asking the applicant to provide additional information on the neutron flux and energy spectra in FSAR Tier 2 Section 12.3 used for radiation shielding and activation. In its response to RAI 9300, Question 12.03-28 (ADAMS Accession No. ML18066A112), the applicant provided proposed changes to FSAR Tier 2, Section 12.2.1.1 and Table 12.2-1. Because the applicant has provided information that is used to demonstrate its compliance with the requirement of 10 CFR 52.47(a)(5) to identify the kinds and quantities of radiation present in the plant, **the NRC staff is currently tracking RAI 9300, Question 12.03-28, as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision. The response to this RAI did not address the NRC staff’s questions about the neutron spectrum and fluence at points of interest above the core. However, the NRC staff is evaluating those aspects under RAI 9282, which is addressed separately.

FSAR Tier 2, Revision 0, Table 12.2-12, lists the drum dryer as one of the components that contains radioactive material. This FSAR Tier 2 table states that the source is modeled as a 20-foot-by-7-foot-by-6-foot (840-ft^3) solid body with a parallelogram for each face. FSAR Tier 2, Revision 0, Section 11.2, notes that the drum dryer consists of a system designed to pump water into a 55-gallon drum (nominally 7.4 ft^3), which is heated and evacuated to rapidly evaporate the liquid in the drum until only solid material remains in the drum. The remaining concentrate contains all the nonvolatile radioactive material added to the drum, which, in turn, serves as the basis for establishing the dose rates near the drums. FSAR Tier 2 Table 12.2-13b lists the quantities of the isotopes expected to be present in a drum. Based on information from the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the NRC staff’s analysis of the radioactive material content of a dried drum was consistent with the information provided in FSAR Tier 2 Table 12.2-13b. FSAR Tier 2 Table 12.3-1 states that an area defined as Radiation Zone III has dose rates greater than or equal to 2.5 mrem/h (0.025 mSv/h) and less than or equal to 5 mrem/h (0.05 mSv/h). FSAR Tier 2, Revision 0, Figure 1.2-30, shows drum dryer room A (Room 030-106) and drum dryer room B (Room 030-107) in the RWB. FSAR Tier 2, Figure 12.3-2b, shows the area corresponding to drum dryer room A on Figure 1.2-30 as Radiation Zone III (i.e., dose rates greater than or equal to 2.5 mrem/h (0.025 mSv/h) and less than or equal to 5 mrem/h (0.05 mSv/h). FSAR Tier 2 Figure 12.3-2b does not provide a radiation zone designation for the area corresponding to drum dryer room B on Figure 1.2-30. The source of radiation within the drum dryer room is the concentrated material contained within the drum. Because 10 CFR 20.1003 states that a high-radiation area is defined at 30 centimeters (0.984 foot) from the radiation source, the NRC staff uses a distance of 1 foot from the source (the drum) for assessing radiation zone designations in the application. The volume of a drum (nominally 7.4 ft^3) is significantly less than the volume of the drum dryer room (840 ft^3). Therefore, the NRC staff compared the estimated dose rates based on dimensions (i.e., of the drum dryer room) listed in FSAR Tier 2 Table 12.2-13b to the estimated dose rates 1 foot from a drum. The NRC staff's analysis indicated that the dose rate on a drum of dried liquid containing the amount of radioactive material listed in FSAR Tier 2 Table 12.2-13b may exceed the indicated radiation zone depicted on FSAR Tier 2, Figure 12.3-2b, for drum dryer room A by several orders of magnitude. Therefore, the NRC staff issued RAI 9302, Question 12.03-14 (ML18026A630), asking the applicant to provide additional information on the methods, models, and assumptions that it used to establish the depicted radiation zone designations. Because the dimensions of the source in the drum dryer room (i.e., a 55-gallon drum) are not consistent with the size of the source specified in FSAR Tier 2, Table 12.2-12, a potentially non-conservative estimate of the maximum dose rate in the zone could result. Therefore, the NRC staff also issued RAI 9302, Question 12.03-15, asking the applicant to provide additional information on the methods, models, and assumptions that it used to establish the depicted radiation zone designations.

In its response to RAI 9302, Questions 12.03-14, and 12.03-15, dated August 27, 2018 (ADAMS Accession No. ML18239A417), the applicant provided an updated zone map for the RWB (in response to RAI 9302, Question 12.03-14) and described the source configuration to use for determining radiation zoning areas (in response to RAI 9302, Question 12.03-15). Combined with the information in the response to RAI 9256 (ADAMS Accession No. ML18225A249), which contains the drum dryer source term, the NRC staff was able to verify the zone reported by the applicant in this updated response. As a result, the NRC staff finds the response acceptable because the applicant provided information to allow the NRC staff to verify the radiation zoning reported by the applicant in its submittal. In addition, the applicant provided FSAR Tier 2 markups to update radiation zoning information and provided the source geometry in FSAR Tier 2, Table 12.2-12, needed to perform a confirmatory analysis. However, although the drum dryer source term information and zoning for drum dryer room A are acceptable, the radiation zone maps provided did not clearly show the designation for drum dryer room B. In addition, the radiation zoning designations for a few other areas in the revised RWB radiation zone figures are unclear. Therefore, the NRC staff is unable to reach a conclusion on the acceptability of the applicant's response to RAI 9302, Question 12.03-14, until the applicant provides updated radiation zone drawings. **As a result, RAI 9302, Question 12.03-14, remains open, and RAI 9302, Question 12.03-15, is being tracked as a confirmatory item**, pending the incorporation of the proposed changes in a future FSAR Tier 2 revision.

Based on the above, except for the open and confirmatory items, the NRC staff concludes that the NuScale FSAR Tier 2 adequately addresses radiation 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 anticipated to contain higher levels of radioactivity. FSAR Tier 2, Table 12.2-26, provides the ventilation air change rates for the pool airspace, the CVCS pump/valve rooms, and degasifier rooms. SER Section 12.2 discusses RAIs related to determining airborne activity concentrations.

The applicant did not provide information on an initial leakage test consistent with 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737, Task Action Plan Item III.D.1.1, in FSAR Tier 2, Revision 0, Table 14.2-36, “Gaseous Radioactive Waste System Test # 36”; Table 14.2-38, “Chemical and Volume Control System Test # 38”; or Table 14.2-53, “Process Sampling System Test # 53.” Therefore, the NRC staff issued RAI 9275, Question 12.03-13 (ML18018A776), asking the applicant to provide additional information on how the initial test program (ITP) complies with the requirements in 10 CFR 50.34(f)(2)(xxvi). The applicant responded to RAI 9275, Question 12.03-13 on March 16, 2018 (ADAMS Accession No. ML18075A285). The disposition of this question is related to the applicant’s use of a proposed alternate post-accident source term. Therefore, **the NRC staff is tracking RAI 9275, Question 12.03-13, as an open item**.

FSAR Tier 2 Section 12.3.1.1.7, “Ventilation,” states that the duct air velocity is kept at sufficiently high velocities to keep particulates suspended. FSAR Tier 2 Section 12.3.3.2, “Design Features to Minimize Personnel Exposure from Heating Ventilation and Air Conditioning Equipment,” states that ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts and that the duct air velocity is kept at sufficiently high velocities to keep particulates suspended. FSAR Tier 2 Section 12.3.3, “Ventilation,” states that the plant HVAC systems are designed to provide a controlled environment for personnel and equipment during normal operation. 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. FSAR Tier 2 Section 12.3.3.2 states that the building ventilation systems are designed to maintain an airflow inside the building from areas of low airborne potential to areas of higher airborne potential. FSAR Tier 2, Section 9.4.2.2.1, “Component Description,” states that ducting interior and exterior surfaces have relatively smooth finishes to reduce the localized collection of radioactive contamination.

FSAR Tier 2 Chapter 9 and FSAR Tier 2 Chapter 12 do not provide any additional information on the physical parameters of the HVAC systems in the RXB or RWB that support these statements (e.g., the referenced industry standards on bend radii, surface finish criteria/grade, or local flow rates versus duct sizes needed to maintain airborne radioactive materials suspended in the ducts’ air, local airflow rates needed to sweep air from areas of lower contamination to higher contamination). In addition, they do not discuss the design features (e.g., flow balancing dampers) provided to support establishing the required flowrates. Therefore, the NRC staff issued RAI 9277, Question 12.03-48 (ML18092A994), asking the applicant to provide additional information on the design features related to maintaining the stated performance criteria. In its response to RAI 9277, Question 12.03-48, dated May 23, 2018 (ADAMS Accession No. ML18143B390), the applicant stated that HVAC system details are not finalized; therefore, the FSAR Tier 2 does not describe the system to the level of detail requested by this RAI. However, the primary design consideration of keeping the RXB and RWB negative relative to the outside environment was addressed in FSAR Tier 1 and was covered by ITAAC described in FSAR Tier 1 Revision 1 Table 3.3-1: “Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria.” The applicant has identified that the COL applicant will complete the design and has identified the relevant industry guidance documents that the COL applicant will use. Because the COL applicant is expected to finalize the design, the NRC staff finds that the level of detail provided at this stage provides reasonable assurance that adequate ventilation will be provided to ensure radiological protection for the worker. Therefore, the NRC staff considers RAI 9277, Question 12.03-48 closed.

FSAR Tier 2, Revision 0, Table 14.2-96, "Reactor Building Ventilation System (RBVS) Capability (Test #96)," and FSAR Tier 2 Table 14.2-20, "Reactor Building HVAC System Test # 20," do not include any test parameters or criteria related to the radiation protection design functions for the RXB HVAC system, as described above. Therefore, the NRC staff issued RAI 9277, Question 12.03-49 (ML18092A994), asking the applicant to provide additional information on the elements of the ITP provided to verify the functionality of the radiation protection design features of the HVAC system in the RXB. In its response to RAI 9277, Question 12.03-49, dated May 23, 2018 (ADAMS Accession No. ML18143B390), the applicant stated that HVAC system details are not finalized; therefore, the FSAR Tier 2 does not describe the system to the level of detail requested by this RAI. However, the primary design consideration of keeping the RXB and RWB negative relative to the outside environment was addressed in FSAR Tier 1 and was covered by ITAAC described in FSAR Tier 1 Revision 1 Table 3.3-1: "Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria." The applicant has identified that the COL applicant will complete the design and has identified the relevant industry guidance documents that the COL applicant will use. The applicant stated that "a system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program (Section 14.2). Airflow rates are measured and balanced in accordance with the guidelines of Sheet Metal and Air Conditioning Contractors National Association (SMACNA) HVAC Systems Testing, Adjusting and Balancing." Because the COL applicant is expected to complete the design, the NRC staff finds that the level of detail provided at this stage provides reasonable assurance that adequate ventilation will be provided to ensure radiological protection for the worker. Therefore, the NRC staff considers RAI 9277, Question 12.03-49 closed.

FSAR Tier 2, Revision 0, Table 14.2, does not contain tests of the radiation protection design functions, described above, of the HVAC system in the RWB similar to those provided for the RXB in FSAR Tier 2 Tables 14.2-96 and Table 14.2-20. FSAR Tier 2, Revision 0, Table 14.2, does not contain a test of the RWB ventilation system that includes any test parameters or criteria related to ensuring the functionality of the radiation protection design features of the RWB ventilation system. There do not appear to be any tests that verify that functional requirements are met, such as testing that the radioactive waste building heating, ventilation, and air conditioning system is maintaining the design environment for SSCs in the RWB (e.g., airflow from areas of low contamination to high contamination), checking that sufficient flow rates exist in the ducts to prevent the settling of radioactive material in the ducts, testing that negative pressure with respect to adjacent areas inside and outside of the facility is maintained, and verifying that the ventilation of areas that may contain explosive/flammable gases is adequate. Therefore, the NRC staff issued RAI 9277, Question 12.03-50 (ML18092A994), asking the applicant to provide additional information on the elements of the ITP provided to verify the functionality of the radiation protection design features of the HVAC system in the RWB. In its response to RAI 9277, Question 12.03-50, dated May 23, 2018 (ADAMS Accession No. ML18143B390), the applicant stated that HVAC system details are not finalized; therefore, the FSAR Tier 2 does not describe the system to the level of detail requested by this RAI. However, the primary design consideration of keeping the RXB and RWB negative relative to the outside environment was addressed in FSAR Tier 1 and was covered by ITAAC described in FSAR Tier 1 Revision 1 Table 3.3-1: "Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria." . The applicant has identified that the COL applicant will complete the design and has identified the relevant industry guidance documents that the COL applicant will use. The applicant stated the following:

> *The Radioactive Waste Building HVAC system (RWBVS) supports personnel access and equipment functions by maintaining a suitable operating environment in the Radioactive Waste Building (RWB), including the RWB control and monitoring room. The RWBVS also supports the control of radioactive contamination by maintaining airflow from areas of lesser potential contamination to areas of greater potential contamination, maintaining the RWB at a negative pressure with respect to the outside atmosphere, and collecting potentially contaminated discharges vented from equipment in the RWB.*

The applicant further stated that "a system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program (Section 14.2). Airflow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC Systems Testing, Adjusting and Balancing." Because the COL applicant is expected to finalize the design, the NRC staff finds that the level of detailed provided at this stage provides reasonable assurance that adequate ventilation will be provided to ensure radiological protection for the worker. Therefore, the NRC staff considers RAI 9277, Question 12.03-50 closed.

FSAR Tier 2, Revision 0, Section 9.4.2.2.1, states that the four air handling units with variable speed supply air fans provide the cooling and heating of the ventilation air serving the RXB. FSAR Tier 2 Section 9.4.2.2.2.2, "Off-Normal Operation," states that, on a high radiation alarm in the SFP area, the isolation damper of the RXB general exhaust from the dry dock area and the SFP area is closed and supply fans reduce the capacity to accommodate the reduction in exhaust. This change in supply airflow ensures that air in other areas of the RXB continues to flow from areas of low contamination to areas of potentially higher contamination and that exhaust from the RWB and annex building continues to flow into the RXB ventilation system exhaust. However, FSAR Tier 2, Revision 0, Tables 14.2-96 and 14.2-20, do not include any test parameters or criteria related to the radiation protection design functions described (e.g., variable speed controller operation for normal HVAC operation, variable speed controllers following a high radiation signal, and operation of the SFP and dry dock ventilation dampers). Therefore, the NRC staff issued RAI 9277, Question 12.03-51 (ML18092A994), asking the applicant to provide additional information on the elements of the ITP included to verify the functionality of the radiation protection design features of the HVAC system in the refueling and dry dock areas of the RXB. In its response to RAI 9277, Question 12.03-51, dated May 23, 2018 (ADAMS Accession No. ML18143B390), the applicant stated that HVAC system details are not finalized; therefore, the FSAR Tier 2 does not describe the system to the level of detail requested by this RAI. However, the primary design consideration of keeping the RXB and RWB negative relative to the outside environment was addressed in FSAR Tier 1 and was covered by ITAAC described in FSAR Tier 1 Revision 1 Table 3.3-1: "Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria." The applicant has identified that the COL applicant will complete the design and has identified the relevant industry guidance documents that the COL applicant will use. The applicant stated that the plant control system controls the RXB ventilation system as shown in FSAR Figure 7.0-20, "Plant Control System Internal Functions and External Interfaces." Therefore, the plant control system controls the speed of the RXB ventilation system variable speed fans in both normal and off-normal operations. The applicant also identified several existing tests in FSAR Tier 2 Section 14.2 that are provided to test aspects of the HVAC system to ensure the operability of some radiation protection design features. Because the COL applicant is expected to finalize the design, the NRC staff finds that the level of detailed provided at this stage provides reasonable assurance that adequate ventilation will be provided to ensure radiological protection for the worker. Therefore, the NRC staff considers RAI 9277, Question 12.03-51 closed.

Based on the above, the NRC staff concludes that the NuScale FSAR Tier 2 adequately addresses ventilation controls, pending resolution of the airborne activity concentration issues identified in Section 12.2.

12.3.4.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

FSAR Tier 2, Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation," discusses fixed airborne and area radiation monitors. Information related to the monitors appears in FSAR Tier 2, Tables 12.3-11 and 12.3-12. FSAR Tier 2, Table 12.3-10, provides information on radiation monitors credited to monitor post-accident radiation levels in accordance with the guidance in RG 1.97. 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. Because the NuScale NPMs include a very small containment compared to large light-water reactors, the inclusion of radiation monitors inside containment is not feasible. Therefore, the under-the-bioshield monitors meet the intent of 10 CFR 50.34(f)(2)(xvii) to monitor radiological conditions during an accident. These monitors meet the range and placement criteria specified in NUREG-0737, Task Action Plan Item II.F.1.3. These monitors are environmentally qualified and post-accident monitoring Type B and Type C variables in accordance with the guidance in RG 1.97, Revision 3.

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 Tier 2, Section 12.5, "Operational Radiation Protection Program," requires the COL applicant to describe. 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 RG 1.97, RG 8.2, and RG 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 by the FSAR Tier 2 and DSRS Section 12.3–12.4, provides examples of appropriate locations for radiation monitors in PWRs. The NuScale design includes radiation monitors in areas consistent with that provided in ANSI/ANS HPSSC-6.8.1-1981.

FSAR Tier 2, Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," states that the design and controls for operation of the fuel handling equipment and fuel storage racks prevent an inadvertent criticality using geometrically safe configurations and plant programs and procedures for criticality control. The FSAR Tier 2 Section 9.1.1 states that the fuel storage racks have an effective multiplication factor (K_{eff}) that meets 10 CFR 50.68. FSAR Tier 2 Section 9.1.1 does not mention how the requirement of 10 CFR 50.68(b)(6) is met. FSAR Tier 2, Section 9.1.2.3.6, "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 Section 12.3. These instruments allow operators to initiate appropriate safety actions. FSAR

Tier 2 Section 9.1.2.3.6 does not indicate whether the radiation monitor mentioned is used to satisfy the requirement of 10 CFR 50.68(b)(6). FSAR Tier 2 Section 12.3.4 states that the area radiation monitors located in the reactor pool area and the SFP area provide the same functions as the general plant location monitors and that they monitor the fuel storage and handling areas. In addition, a local area radiation monitor is mounted on the refueling bridge with a local and MCR alarm function that monitors refueling activities. FSAR Tier 2 Section 12.3.4.1, “Design Bases,” does not mention 10 CFR 50.68(b)(6). FSAR Tier 2 Section 12.3.4.2, “Fixed Area Radiation Monitoring Instrumentation,” does not include the 10 CFR 50.68(b)(6) requirement as one of the criteria for placement of a radiation monitor. Although FSAR Tier 2 Section 9.1.1 indicates that the fuel storage racks meet several of the requirements of 10 CFR 50.68, the FSAR Tier 2 does not clearly state which regulation (i.e., 10 CFR 50.68 or 10 CFR 70.24) the design meets for the requirements discussed above. In addition, FSAR Tier 2, Sections 9.1.1, 9.1.2, and 12.3, do not identify the radiation monitors provided to satisfy the requirements of 10 CFR 70.24(a)(1) or 10 CFR 50.68(b)(6) and do not state how the requirements for radiation monitors are being met. Therefore, the NRC staff issued RAI 9288, Question 12.03-09 (ML18010B131), asking the applicant to clearly identify (1) which regulation (10 CFR 70.24(a)(1) or 10 CFR 50.68(b)(6) the applicant was following (i.e., which specific radiation monitors were credited for satisfying the radiation monitoring requirements) and (2) the basis for how the type and location of the radiation monitors specified satisfied the applicable regulatory requirements. In its response to RAI-9288, Question 12.03-9, dated March 12, 2018 (ADAMS Accession No. ML18071A385), the applicant stated that NuScale was complying with the requirements of 10 CFR 50.68(b) in lieu of 10 CFR 70.24. The applicant proposed changes to FSAR Tier 2, Section 9.1.1.1, “Design Basis,” stating that the radiation monitors described in FSAR Tier 2, Section 12.3.4, were provided for compliance with 10 CFR 50.68(b)(6). The applicant also proposed a change to FSAR Tier 2, Section 12.3.4.1, stating that area radiation monitoring provides radiation monitoring in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions consistent with the requirements of 10 CFR 50.68(b)(6). FSAR Tier 2, Revision 1, Table 12.3-12, “Fixed Area Radiation Monitors,” shows that there are 10 gamma-sensitive reactor pool radiation monitors on the 126-foot elevation of the RXB. FSAR Tier 2, Revision 1, Section 12.3.4.2, states that the fixed area radiation monitor placement conforms to the criteria for selection and placement of the area radiation monitoring instrumentation in ANSI/ANS HPSSC-6.8.1-1981. ANSI/ANS HPSSC-6.8.1-1981, Section 4.2.3, requires detectors to be located such that inadvertent shielding by structural materials is minimized. Based on the information contained within the FSAR Tier 2 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 applicant’s response is consistent with the requirements of 10 CFR 70.24 and 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. Therefore, RAI 9288, Question 12.03-9, is being tracked **as a confirmatory item**, and the NRC staff will confirm that the applicant includes this information in a future revision of the FSAR Tier 2.

FSAR Tier 2, Section 12.3.4.1, states that the radiological monitoring equipment is designed to provide monitoring of 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 Tier 2 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 Tier 2 Section 7.2 discusses the ERDS. FSAR Tier 2 Section 12.3.4.3, “Airborne Radioactivity Monitoring Instrumentation,” states that fixed continuous airborne monitor 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 and that FSAR Tier 2 Section 7.2 discusses the ERDS connection. Although FSAR Tier 2 Section 12.3.4 states that area and airborne radiation monitors can provide information to the ERDS and although FSAR Tier 2 Section 7.2 states that the ERDS system will provide the interconnection to the NRC emergency data system, these FSAR Tier 2 sections do not identify which specific instruments will be used to satisfy the requirements of 10 CFR Part 50, Appendix E, Section VI.2(a)(i). Therefore, the NRC staff issued RAI 9289, Question 12.03-10 (ML18012A748), asking the applicant to clearly identify which radiation monitors will be used to satisfy the requirements of 10 CFR Part 50, Appendix E, Section VI.2(a)(i). In its response to RAI 9289, Question 12.03-10, dated February 15, 2018 (ADAMS Accession No. ML18046A106), the applicant stated that 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). In addition, the applicant provided information on COL Item 13.3-3, which directs a COL applicant to follow NUREG-1394, "Emergency Response Data System (ERDS) Implementation," Revision 1, issued June 1991 (ADAMS Accession No. ML0807900380). NUREG-1394 provides guidance for how to comply with the ERDS requirements. The specific instruments will be identified as part of COL Item 13.3-3. The NRC staff finds this response acceptable because the applicant has described a COL item that will have the COL applicant describe those radiation monitors that will be used to comply with 10 CFR Part 50, Appendix E, Section VI.2(a)(i). The NRC staff considers RAI 9289, Question 12.03-10 closed.

Based on the above, except for the confirmatory item, the NRC staff concludes that the NuScale FSAR Tier 2 adequately addresses area and airborne radiation monitoring requirements.

12.3.4.1.5 Minimization of Contamination

Under 10 CFR 20.1406, the NRC requires each licensee to describe how it intends to minimize, to the extent practicable, contamination of the facility and the environment and the generation of radioactive waste. The regulation also requires applicants to describe how it will facilitate decommissioning. The guidance in DSRS Section 12.3–12.4 states that design features described by the applicant should facilitate eventual decommissioning and minimize, to the extent practicable, contamination of the facility and the environment and the generation of radioactive waste. RG 4.21 contains a basis acceptable to the NRC staff for complying with the requirements of 10 CFR 20.1406. Wherever the applicant adheres to this guidance, the NRC staff can have reasonable assurance of its compliance with 10 CFR 20.1406.

FSAR Tier 2, 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 Tier 2 Revision 1, Section 12.3.6.1 "Facility Design Objectives for 10 CFR 20.1406," describes 4 design objectives and 2 operational program objectives used by the applicant during the design phase, and specified for use by COL Licensees utilizing the approved design. As described by the applicant, 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*

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 Tier 2, Table 12.3-13, "NuScale Power Plant Systems with NRC Regulatory Guide 4.21 Evaluation," lists systems that were evaluated using the guidance in RG 4.21. FSAR Tier 2, Tables 12.3-14 through 12.3-44, 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. In addition, the FSAR Tier 2 includes COL 12.3-7, which specifies that the COL applicant must develop a plant wide RG 4.21 program to address the operational and programmatic considerations and site radiological environmental monitoring aspects of the minimization of contamination program 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.

FSAR Tier 2, Revision 0, Section 9.2.3.1, "Demineralized Water System" (DWS), states that the DWS does not contain radioactive materials but does interface with radioactive systems. FSAR Tier 2 Section 9.2.3.1 states that, where the DWS interfaces with radioactive waste processing systems or radioactive liquid containing systems, the DWS includes backflow preventers in each of the lines that distribute demineralized water to these systems. FSAR Tier 2 Section 9.2.3.1 further states that FSAR Tier 2 Section 12.3 presents compliance with the requirements of 10 CFR 20.1406. FSAR Tier 2 Table 9.2.3-2, "Demineralized Water System Monitoring Parameters," lists DWS chemical parameters that require analysis. However, FSAR Tier 2 Section 12.3–12.4 does not mention the backflow preventers as a mechanism used to prevent contamination of the DWS. In addition, contrary to the guidance in Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment," dated May 6, 1980, FSAR Tier 2 Table 9.2.3-2 does not contain a requirement to check the DWS for radioactive material contamination. EPA 570989007-1989, "Cross-Connection Control Manual," issued July 1989, describes six basic types of devices that can be used to correct cross-connections: (1) air gaps, (2) barometric loops, (3) atmospheric- and pressure-type vacuum breakers), (4) double check valves with intermediate atmospheric vents, (5) double check valve assemblies, and (6) reduced pressure principle devices. FSAR Tier 2 Sections 12.3 and 12.4 and Table 12.3-15, "Regulatory Guide 4.21 Design Features for Balance-of-Plant Drain System," do not describe any of these devices. In addition to installation requirements, backflow prevention devices also have specific initial testing requirements and periodic testing requirements. However, FSAR Tier 2, Revision 0, Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," does not contain any ITP requirements for the backflow preventers described in FSAR Tier 2 Section 9.2.3.1; FSAR Tier 2 Section 9.2.3 does not contain any COL items related to providing the testing for backflow preventers; and FSAR Tier 2, Revision 0, Table 1.8-2, does not describe a COL item for testing backflow preventers. Therefore, the NRC staff issued RAI 8787, Question 12.03-02, dated June 2, 2017 (ML17153A376), asking the applicant to provide additional information on the description of the backflow preventers, the initial testing requirements for the backflow preventers, and sample program elements needed to ensure cross-contamination of the DWS system does not occur or is quickly detected. In its response dated August 1, 2017 (ADAMS Accession No. ML17213B400), the applicant stated that the backflow prevention guidance provided by the document cited in EPA 570989007-1989 only applies to potable water systems connected to non-potable water systems, and because the DWS is not connected to any potable water systems, regulatory requirements do not exist for testing these valves in the ITP. The applicant stated that the COL licensee will select the type of backflow prevention device to be used in the DWS. The applicant provided proposed changes to FSAR Tier 2 Table 12.3-13 and FSAR Tier 2 Table 9.3.2-4, "Local Sample Points," to add the DWS system; FSAR Tier 2 Figure 9.2.3-1, "Demineralized Water System Diagram," to add a backflow preventer; and the supporting description in FSAR Tier 2 Section 9.2.3, "Demineralized Water System." The applicant's response also proposed modifying FSAR Tier 2, Table 9.3.2-4, to include radionuclides as the sample analysis. In this same RAI response, the applicant included radionuclides and tritium as changes to the sample analysis for the utility water system (UWS). Although the NRC staff believes that because the concentration of tritium in the RCS and the mobility of tritium through water-containing systems would warrant inclusion of tritium as one of the specific isotopes to monitor for showing compliance with 10 CFR Part 20, Subpart F, "Surveys and Monitoring," based on the low radiological consequences of exposure to tritium at the low concentration values expected, the NRC staff has accepted the applicant's response. The NRC staff has verified the changes in FSAR Tier 2, Revision 1; therefore, RAI 8787, Question 12.03-02 is closed.

FSAR Tier 2 Section 9.2.9 states that the UWS distributes clarified water to the RXB and RWB. However, FSAR Tier 2, Revision 0, Chapters 9, 11, and 12, do not describe the intended use of UWS water in the RXB and RWB. FSAR Tier 2, Revision 0, Table 12.3-43, "Regulatory Guide 4.21 Design Features for Utility Water System," asserts that the only portion of the UWS that may become contaminated is the discharge basin. The UWS is composed of two separate sections: (1) the supply side, which provides uncontaminated water for use by plant operators, and (2) the disposal side, which is normally expected to contain radioactivity. Contrary to the guidance in Bulletin 80-10 to perform sampling to detect inadvertent contamination of normally nonradioactive systems, Table 9.3.2-4 does not contain a requirement to check the supply portion of the UWS for radioactive material contamination. Therefore, the NRC staff issued RAI 8787, Question 12.03-03 (ML17153A376), asking the applicant to provide additional information on the sampling parameters and sampling frequency for the UWS with respect to checking for radioactive material contamination. In its response dated August 1, 2017 (ADAMS Accession No. ML17213B400), the applicant provided proposed changes to the FSAR Tier 2 to show that water from the UWS will be used in the RXB and RWB for plant maintenance, including general wash downs. The applicant revised FSAR Section 9.2.9 to include this information. The applicant revised FSAR Tier 2 Table 9.3.2-4 to indicate that liquid grab samples will be taken from the supply portion of the UWS to test for the presence of radionuclides, including tritium. Because the applicant proposed changes to the FSAR Tier 2 that would identify that the UWS extended into the RXB and RWB and would specify periodic sampling of the UWS for radioactive material, the NRC staff finds that the applicant's response is consistent with the requirements in 10 CFR 20.1406 and the guidance in RG 4.21. The NRC staff has verified the changes in FSAR Tier 2, Revision 1; therefore, RAI 8787, Question 12.03-03, is closed.

Information in FSAR Tier 2, Revision 0, and in the applicant's response to RAI 8963, Question 03.08.05-23, dated October 17, 2017 (ADAMS Accession No. ML17290B267), indicates that the pool leakage detection system (PLDS) may not cover portions of the pool liner. For example, the response to RAI 8963 Question 03.08.05-23, did not address areas of the UHS pool liner, such as the vertical liner plates. It is not clear to the NRC staff how the applicant intends to meet the regulatory requirements for providing sufficient information in an early stage of the design that demonstrates the capability of the PLDS to detect low leakage rates from structures containing pool water. The "Liquid Radioactive Release Lessons Learned Task Force Final Report," dated September 1, 2006 (ADAMS Accession No. ML062650312), states that radioisotopes, including fission products, have been released from SFPs. The report further notes that the potential exists for unplanned and unmonitored releases of radioactive liquids to migrate offsite undetected, including those portions of SFPs that are not visible to operators. Leakage (and the resultant contamination) that enters the ground below the plant may be undetected. Radioactive contamination in ground water on site may migrate undetected off site. The task force identified leakage from SFPs as one of the main components resulting in ground water contamination. In its response to RAI 8963, Question 03.08.05-23, the applicant stated that FSAR Tier 2, Section 9.1.3.2.5, describes the PLDS. That section states that the PLDS consists of floor leakage channels, perimeter leakage channels, channel drainage lines, leak collection headers, leakage rate measuring lines, and valves. The floor leakage channels are embedded in the concrete beneath the field-welded seams of the pool floor liner plates in the UHS pools and the dry dock. A perimeter channel is embedded in concrete at the wall and floor liner joint area. Based on the NRC staff's interpretation of this response and on information in the FSAR Tier 2, it appears that the NuScale design only monitors for leakage from welds located on the base mat and at the juncture of the walls and the base mat. However, the guidance in SRP Section 9.1.2, "New and Spent Fuel Storage," does not distinguish between those sections of the pool liner located on the wall and those sections of the liner located on the base mat. The acceptance criteria of DSRS Section 12.3–12.4 state that the acceptability of the design features described in the application will be based on the guidance in RG 4.21 and DSRS Section 12.3–12.4, Appendix 12.3–12.4-A, Attachment A, "Evaluation and Scoping Information for Structures, Systems, and Components 10 CFR 20.1406—Design Review." DSRS Section 12.3–12.4, Appendix 12.3–12.4-A, specifically identifies SSCs, such as SFPs, separated from the environment by a single barrier or with below-grade concrete-to-concrete joints (e.g., the UHS pool in the NuScale RXB). DSRS Section 12.3–12.4, Appendix 12.3-12.4-A, Attachment B, "Examples of Structures, Systems, and Components for 20.1406 Review," specifically identifies the SFP as an area for the NRC staff to review. Therefore, the NRC staff issued RAI 9292, Question 12.03-43 (ML19015A278), asking the applicant to provide additional information on the design features provided to detect leakage from other areas of the UHS pool (e.g., the vertical walls).

FSAR Tier 2, Revision 0, Table 9.3.2-4, does not list the PLDS as one of the process sampling points. Furthermore, the applicant's response to RAI 8963, Question 03.08.05-23, states that channels collect leakage from the pool liner plates and direct it to a sump or to collection header piping that leads to a sump that is part of the radioactive waste drain system (RWDS). The RWDS sumps are in the RXB gallery areas at the top of the concrete at an elevation of 24 feet, 0 inch. However, FSAR Tier 2 Figure 1.2-10, "Reactor Building 24'-0" Elevation," and FSAR Tier 2 Figure 12.3-1a do not show sumps on this elevation of the RXB. Therefore, the NRC staff issued RAI 9292, Question 12.03-44 (ML19015A278), asking the applicant to provide additional information on the location and design of the PLDS components.

The "Liquid Radioactive Release Lessons Learned Task Force Final Report," dated September 1, 2006 (ADAMS Accession No. ML062650312), describes an event in which the liner leakage detection system became clogged with boric acid precipitate. FSAR Tier 2 Section 9.1.3.2.5, "Pool Leakage Detection System," and the applicant's response to RAI 8963, Question 03.08.05-23, state that the PLDS shall be designed to support periodic testing and inspection of PLDS components to identify leakage from the RXB components pool liner welds. However, with the PLDS located in radioactive waste system sumps, it is not clear to the NRC staff what design features are provided to facilitate the inspection and cleaning of the PLDS during the operation of one or more NPMs or the during the refueling of an NPM. Therefore, the NRC staff issued RAI 9292, Question 12.03-45 (ML19015A278), asking the applicant to provide additional information on the design features provided to minimize radiation exposure in accordance with 10 CFR 20.1101(b) and 10 CFR 20.1701(a) during maintenance and testing activities of the PLDS.

FSAR Tier 2 Section 9.1.3.2.5 states that the sumps in the RWDS are monitored for level and that the RWDS supports the leakage detection function of the PLDS by providing local and control room indication and associated alarms when the leakage rate from the PLDS reaches a predetermined level. FSAR Tier 2 Section 9.3.3.2.3, "System Operation," states that the PLDS works in cooperation with the RWDS equipment drain subsystem. The PLDS drains are not individually monitored; however, because all other drains into the equipment drain system are manually initiated, unplanned changes in sump volume can be attributed to the PLDS. The RG 4.21 guidance on the regulatory requirements of 10 CFR 20.1406 states that structures and components, such as an SFP and associated piping, should have the capability to detect and quantify small leakage rates (e.g., several gallons per week) from each zone that may be masked by other sources of radioactive liquid entering the sumps. Therefore, the NRC staff issued RAI 9292, Question 12.03-46 (ML19015A278), asking the applicant to provide additional information on the design features that support the ability to detect low flow rate leaks from the PLDS into the RWDS.

Because the PLDS appears to be connected to sumps in the RXB that are part of the RWDS, the NRC staff is unable to determine how to prevent any ingress of water from the sump into the PLDS that may contaminate the PLDS with borated water that may dry and clog the PLDS or be mistaken for pool leakage. Therefore, the NRC staff issued RAI 9292, Question 12.03-47 (ML19015A278), asking the applicant to provide additional information on the design features that prevent backflow of water from the RWDS into the PLDS.

The applicant provided responses to RAI 9292, Questions 12.03-43, 12.03-44, 12.03-45, 12.03-46, and 12.03-47, dated February 8, 2018 (ADAMS Accession No. ML18099A365). Based on the responses, the NRC staff determined that the applicant would need to provide additional information for each question. From June 14, 2018, through June 28, 2018, the NRC staff conducted an audit of the PLDS for the UHS to resolve RAI 9292, Questions 12.03-43, 12.03-44, 12.03-45, 12.03-46, and 12.03-47 (see ADAMS Accession No. ML18158A164).

Based on the applicant's response to RAI 9292, Question 12.03-43 (ADAMS Accession No. ML18099A365) and on information provided by the applicant during the audit, the NRC staff determined that there were no leak chase channel systems in the walls of the UHS. The NRC staff is concerned about the potential for leakage through the UHS vertical liner plates to follow cracks or gaps in the concrete for those areas of the vertical liner not covered by leak collection chases.

As discussed during the audit exit meeting, the applicant will submit a supplemental response to RAI 9292, to provide comprehensive documentation of its strategy to identify and minimize the flow of borated water into the UHS reinforced concrete to ensure that the structural integrity of the UHS walls will be maintained throughout of the life of the license.

With regard to RAI 9292, Question 12.03-44, the audit provided information that allowed the NRC staff to understand NuScale's methods for taking samples of the PLDS. The applicant discussed the ability to take direct samples of the RXB sumps and discussed taking samples directly at the PLDS channels. The NuScale design allows for the isolation of channels and direct sampling from the channels; however, the NRC staff informed the applicant that the FSAR Tier 2 does not provide this information. Design drawings reviewed during the audit depicted the radioactive waste collection sumps. In its application and responses to the NRC staff's questions, the applicant has stated that the plant operator will have the ability to control all sources of water flowing into the radioactive waste tanks. However, during the audit, the NRC staff noted that one of the PLDS drain lines goes to tank 00-RWD-TNK-0004. This tank has several inputs to the tank from which the PLDS sample is to be drawn that apparently cannot be isolated.

As agreed upon during the audit exit meeting, the applicant will submit a supplemental response to RAI 9292, Question 12.03-44, to discuss the methods for taking a representative sample of the PLDS. In addition, the NRC staff asked the applicant to update FSAR Tier 2 Table 9.3.2-4 to include the local sampling points for the PLDS.

With regard to RAI 9292, Question 12.03-45, (ML19015A278) the audit provided information that details the methods for cleaning and inspecting the east-west channels of the floor leakage detection system. The FSAR Tier 2 and RAI responses that the NRC staff received do not discuss these methods. In addition, the NRC staff seeks to understand the methods for inspecting and cleaning the channels beneath the NPMs. During the review of the audit, the NRC staff observed design drawings that specified carbon steel as the material for the PLDS channels. In discussions held during the audit meetings, the applicant specified that stainless steel would be used instead.

As agreed upon during the audit exit meeting, the applicant will submit a supplemental response to RAI 9292, Question 12.03-45, to specify the cleaning and inspection methods for the east-west pool channels and the channels located under and around the NPMs. In addition, the NRC staff asked the applicant to update the design drawings and to provide an updated RAI response to specify the materials that are being used for the PLDS channels.

With regard to RAI 9292, Question 12.03-46, (ML19015A278) the audit provided information that allowed the NRC staff to ask questions about the level detection capability of PLDS sumps. NuScale stated that the level detection instrumentation will have the capability of detecting 2 gallons per week (i.e., 0.76 milliliter per minute) of unexpected leakage from each zone. The NRC staff observes that, for the 4-foot-by-4-foot equipment drain sump volume, the addition of 2 gallons per week would result in a level change of about 0.2 inch per week. The applicant stated that the ITP described in FSAR Table 14.2-6 will test the MCR alarm when the RWDS fill rate exceeds the PLDS leakage rate setpoint. Although the local leakage collection isolation and sampling points would permit quantification of leakage at a flow rate of several gallons per week, it is not clear to the NRC staff what amount of leakage the RWDS level detection and control system will be able to detect and alarm.

As agreed upon during the audit exit meeting, the applicant will submit a supplemental response to RAI 9292, Question 12.03-46, to clarify the amount of leakage each sump of the PLDS will be able to detect and annunciate an alarm.

With regard to RAI 9292, Question 12.03-47, (ML19015A278) the audit provided information that allowed the NRC staff to identify that the applicant had additional information on NuScale's ability to prevent backflow into the UHS through the use of check valves, as described in the applicant's original RAI response. However, as discussed during the audit, the NRC staff seeks to understand whether the described methods to prevent backflow apply to those sumps and PLDS channels at a lower elevation.

As agreed upon during the audit exit meeting, the applicant will submit a supplemental response to RAI 9292, Question 12.03-47, that discusses the leakage collection methods for the 19-foot elevation leakage collection sumps and for sampling from the corresponding 19-foot elevation channels. In addition, the NRC staff seeks clarification on whether the same methods of preventing contamination for the higher elevation sumps apply to the 19-foot elevation sumps. If such methods do not apply, the NRC staff requests that the applicant update the RAI response to discuss the methods used to ensure that the PLDS is not contaminated from backflow. The applicant submitted a supplemental response to RAI 9292 on November 15, 2018 (ADAMS Accession No. ML18319A292). The supplemental response remains under review by the NRC staff. **Based on the above the NRC staff is currently tracking RAI 9292, Questions 12.03-43, 12.03-44, 12.03-45, 12.03-46, and 12.03-47, as open items**.

Except for the open items, the NRC staff concludes that the NuScale FSAR Tier 2 adequately addresses minimization of contamination.

12.3.5 Combined License Information Items

Table 12.3-1 lists COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2, related to radiation protection design features.

Table 12.3-1 NuScale COL Information Items for Section 12.3

COL Item No.	Description	FSAR Tier 2 Section
12.3-1	A COL applicant that references the NuScale Power Plant design certification will develop the administrative controls regarding access to high-radiation areas per the guidance of Regulatory Guide 8.38.	12.3
12.3-2	A COL applicant that references the NuScale Power Plant design certification will develop the administrative controls regarding access to VHRAs per the guidance of RG 8.38.	12.3
12.3-3	A COL applicant that references the NuScale Power Plant design certification 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
12.3-4	A COL applicant that references the NuScale Power Plant design certification 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	A COL applicant that references the NuScale Power Plant design certification will describe design criteria for locating additional area radiation monitors.	12.3.4.2

COL Item No.	Description	FSAR Tier 2 Section
12.3-6	A COL applicant that references the NuScale Power Plant design certification 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
12.3-7	A COL applicant that references the NuScale Power Plant design certification will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary to demonstrate compliance with 10 CFR 20.1406, and the guidance of RG 4.21.	12.3.6

Note: Objectives 5 and 6 are applicant defined terms, and are described in the preceding SER discussion regarding features for compliance with 10 CFR 20.1406 and 10 CFR 52.47(a)(6).

12.3.6 Conclusion

As described above, the NRC staff has reviewed the applicant's submittal against the requirements of 10 CFR Part 20 as it relates to: limits on doses to people in restricted areas, and the applicable requirements, including 10 CFR Part 19; sources of direct radiation exposure to members of the public, including the generally applicable environmental radiation standards in 40 CFR Part 190; 10 CFR 20.1406 and 10 CFR 52.47(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 52.47(a)(5); 10 CFR 52.47(a)(22); and 10 CFR Part 50, Appendix A, GDC 4, GDC 19 and GDC 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 GDC 30 as they relate to RCS pressure boundary radiation monitoring; 10 CFR 52.47(a)(25) and 10 CFR 52.47(a)(26), as they relate to the use of design interfaces for portions of the certified design that the NRC expects the COL applicant to implement; 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 DC can be constructed and operated in conformity with the DC, the provisions of the AEA, and NRC regulations; and 10 CFR 20.1406(a) and 10 CFR 52.47(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.

With the exception of the open items and pending resolution of the confirmatory items discussed above, the NRC staff has determined that the NuScale design meets the applicable requirements discussed in Section 12.3.

12.4 Dose Assessment

12.4.1 Introduction

This section of the application describes the basis for the dose assessment process and provides detailed information on the expected occupancy of plant radiation areas for each radiation zone and the estimated annual person-sievert (person-rem) doses associated with major functions, such as operation, radioactive waste handling, normal maintenance, special maintenance (e.g., SG tube plugging), refueling, and in-service inspection, in accordance with the provisions of RG 8.19 (the DC FSAR or COL FSAR). This section of the application also describes any additional dose-reducing measures taken because of the dose assessment process for specific functions or activities. The NRC staff based its review on the descriptions provided in the application.

12.4.2 Summary of Application

FSAR Tier 1: There is no Tier 1 information for this area of review.

FSAR Tier 2: The applicant discussed the occupational radiation dose assessment in FSAR Tier 2 Section 12.4, "Dose Assessment," and dose estimates for normal operation, including refueling, and provided information related to post-accident actions and dose estimates for post-accident sampling and analysis.

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

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

Technical Reports:

1. NuScale TR-0915-17565, Revision 3

12.4.3 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 using 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, and 10 CFR 20.1702, 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, as they relate to the facility design features that affect the radiation exposure to a member of the public from non-effluent sources associated with normal operations and AOOs.

- 10 CFR 20.1601, 10 CFR 20.1602, 10 CFR 20.1901, 10 CFR 20.1902, 10 CFR 20.1903, and 10 CFR 20.1904, as they relate to the identification of potential sources of radiation exposure and the controls of access to and 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 unauthorized removal from the place of storage.

- 10 CFR 50.34(f)(2)(xvii), using NuScale-specific source terms that require the applicant to provide instrumentation to monitor containment radiation intensity (high level).

- 10 CFR 52.47(a)(5) as it relates to identifying the kinds and quantities of radioactive materials expected to be produced in the operation of the facility and the means for controlling and limiting radioactive effluents and radiation exposures within the limits in 10 CFR Part 20.

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

- GDC 4, "Environmental and Dynamic Effects Design Bases," which requires 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 design as necessary to permit adequate access and to protect safety equipment from the radiation environment.

- 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) TEDE, as defined in 10 CFR 50.2, to the whole body or the equivalent to any part of the whole body for the duration of the accident.

- GDC 63, as it relates to detecting excessive radiation levels in the facility.

- 10 CFR 52.47(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(a)(25) and 10 CFR 52.47(a)(26), as they relate to the use of design interfaces for portions of the certified design that the NRC expects the COL applicant to implement

DSRS Section 12.3–12.4 lists the acceptance criteria 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, as it relates to minimizing ORE 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 1.52, as it relates to radiation protection considerations for atmosphere cleanup systems that are engineered safety features that are operable under postulated DBA conditions and are designated as "primary systems."

- RG 1.69, as it relates to the requirements and recommended practices acceptable for the 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, DSRS Chapter 7, and a memorandum from D.G. Eisenhut (NRR) 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 1.97, the SRP, DSRS Section 11.6, and a memorandum from D.G. Eisenhut (NRR) to Regional Administrators dated August 16, 1982, as they relate to a method acceptable to the NRC staff for complying with the NRC regulations that require the licensee to provide and calibrate radiation monitoring instrumentation and as they relate to monitoring variables and systems that are important to safety during and following an accident.

- RG 1.140, 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 ORE 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 that FSAR Section 12 will provide.

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

- RG 8.2, 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 ORE 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 that FSAR Section 12 will provide.

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

- RG 8.15, 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, as it relates to a method acceptable to the NRC staff for performing an assessment of collective occupational radiation doses as part of the ongoing design review process to ensure that such exposures will be ALARA.

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

- RG 8.38, as it relates to the physical controls for personnel access to high-radiation areas and VHRAs.

- SRP BTP 11-3 and SECY-94-198, as they relate to design features provided to minimize ORE 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, 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/HPS Std. N13.1-2011, 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 Std. 6.4-2006, as it relates to requirements and recommended practices for the construction of concrete radiation shielding structures.

12.4.4 Technical Evaluation

This section provides information on the dose assessment for both normal operation and post-accident actions. The NRC staff reviewed FSAR Tier 2, Section 12.4, for completeness against the criteria in DSRS Section 12.3–12.4. The NRC staff ensured that the applicant had either committed to follow 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 Tier 2 adheres to these RGs and NRC staff positions, the NRC staff can conclude that the relevant requirements of 10 CFR Part 20 have been met. The applicant specified that it plans to submit a revised NuScale TR-0915-17565, Revision 3, which will include the revised accident source term methodology. The changes in accident source term methodology are likely to impact accident source terms and dose rates and may impact the evaluation of this area. Therefore, the NRC staff will conduct a full review following the submittal of NuScale TR-0915-17565, Revision 3. In addition, the NRC staff selectively compared 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 should be 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 Tier 2, Sections 12.3 and 12.4, and other related sections of the FSAR Tier 2 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 follow the guidance in the RGs and applicable NRC staff positions or had offered acceptable alternatives. In areas where the FSAR Tier 2 is consistent with the guidance in these RGs and NRC staff positions, the NRC staff can conclude that the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 70 have been met. The sections below present the NRC staff's findings.

FSAR Tier 2 Section 12.4.1.8, "Post-Accident Actions," discusses post-accident sampling and analysis for both primary liquid sampling and containment gas sampling. It indicates that (1) access may be required to the CVCS gallery, counting room, and hot lab in the RXB, (2) the annex building counting room may be required to sample and analyze liquid samples, and (3) access to the utilities area and steam gallery on the 100-foot elevation of the RXB may be required for containment gaseous grab sampling. The discussion on performance of these activities indicates that doses will be under the 5-rem occupational dose limit, including ingress and egress, but that it may be necessary to install temporary shielding and temporary ventilation equipment. The information in the FSAR Tier 2 did not include post-accident radiation zone maps or provide information related to the entry and egress paths. Therefore, the NRC staff issued RAI 8775, Question 12.03-01 (ML17116A002), dated April 25, 2017, asking the applicant to provide information on the radiation zones, all of the missions needed to support obtaining the required samples, the expected travel paths, and the durations. In its response to RAI-8775, Question 12.03-01, dated June 26, 2017 (ADAMS Accession No. ML17177A699), the applicant provided a new proposed FSAR Tier 2 Table 12.4-8, "Post-Accident Sampling Operator Dose," which includes a timeline and estimated dose for steps of the sampling process. In its response, the applicant provided proposed FSAR Tier 2 Figure 12.3-4a; Figure 12.3-4b; and Figure 12.3-4c, "Reactor Building Post-Accident Radiation Zone Map—100' Elevation," depicting RXB post-accident radiation zone maps for the 50-, 75- and 100-foot elevations. The applicant also proposed changes to FSAR Tier 2 Section 12.4.1.8. FSAR Tier 2, Section 9.3.2.2.3, "System Operation," states that the sample line purge fluid may be collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWSs. Proposed FSAR Tier 2 Table 12.4-8 states that a purge collection tank may be used to collect water from the sample line. The table also states that ¼ inch of temporary lead equivalent shielding material may be staged to reduce dose rates during sampling. Based on the information provided by the applicant, the NRC staff considers RAI 8775, Question 12.03-01, as closed/unresolved and issued RAI 9278, Questions 12.03-31, 12.03-32, 12.03-33, 12.03-34, 12.03-35, 12.03-36, 12.03-37, 12.03-38, 12.03-39, and 12.03-40, as discussed below.

Based on information contained in FSAR Tier 2 Revision 0 Section 12, and information made available to the staff during the Phase I audit, (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), the NRC staff identified concerns with the methods, models, and assumptions used by the applicant to demonstrate compliance with 10 CFR 50.34(f)(2)(vii) and the limits 10 CFR Part 20, for samples taken following accidents. In RAI 9278, Question 12.03 31; RAI 9278, Question 12.03 32; RAI 9278, Question 12.03 33; RAI 9278, Question 12.03 34; RAI 9278, Question 12.03 35; RAI 9278, Question 12.03 36; and RAI 9278, Question 12.03 37 (ADAMS Accession No. ML18028A004), the staff asked the applicant to provide additional information about the source terms used, and the resultant dose consequences of acquiring samples following an accident. The applicant provided responses to the questions in RAI 9278 (ADAMS Accession No. ML18136A869). However, in an April 24, 2018, response to RAI 9261 (ADAMS Accession No. ML18114A370); an April 24, 2018, response to RAI 9268 (ADAMS Accession No. ML18114A371); and a May 18, 2018, response to RAI 8837 (ADAMS Accession No. ML18138A383), the applicant specified that it plans to submit a revised NuScale TR 0915 17565, Revision 3, which will include the revised accident source term methodology. The changes in accident source term methodology are likely to affect accident source terms and dose rates and may affect the response to and evaluation of this question. Therefore, the NRC staff will conduct a full review of this response following the submittal of NuScale TR-0915-17565, Revision 3. **As a result, the NRC staff is currently tracking RAI 9278, Question 12.03 31; RAI 9278 Question 12.03-32; RAI 9278 Question 12.03-33; RAI 9278 Question 12.03-34; RAI 9278 Question 12.03-35; RAI 9278 Question 12.03-36; and RAI 9278 Question 12.03-37 as open items**.

FSAR Tier 2, Revision 0, Section 9.3.2.2.3, and proposed FSAR Tier 2 Table 12.4-8 indicate the need for special tools, such as reach rods for remote operation of local sample valves, mechanical disconnects for sample tubing connections and the associated remote disconnect tools, and shielded carts for sample transport. The acceptance criteria in DSRS Section 12.3–12.4 specifically mention provisions for portable shielding and remote handling tools. Some special tools, like those for handling tubing connectors, are not routinely available, and the use of routinely available remote handling tools, such as tongs or pliers, may significantly extend the amount of time in the area. In addition, although transport carts for radioactive filters and other large components are expected to be available for use, they are not suited for the task of transporting samples. However, a COL item does not exist to direct the COL applicant to provide these types of items. Therefore, the NRC staff issued RAI 9278, Question 12.03-38 (ML18028A004), asking the applicant to describe the equipment and special tools expected to be used to maintain dose to the operators to less than 5 rem while collecting RCS samples following an accident. The NRC staff also asked the applicant to indicate whether the COL applicant will be responsible for providing this material (i.e., a COL item). Based on the applicant’s response to RAI 9278, Question 12.03-38, dated May 10, 2018 (ADAMS Accession No. ML18136A871), the NRC staff agreed with the applicant that a COL Item was not necessary. Therefore, the NRC staff considers RAI 9278, Question 12.03-38 closed.

FSAR Tier 2, Revision 0, Section 9.3.2.2.3, and proposed FSAR Tier 2 Section 12.4.1.8 indicate that the post-accident primary coolant sample is collected through the normal CVCS sample line flow path to the primary system sample at a panel located in the CVCS gallery; however, they appear to indicate that the sample does not flow through the normal PSS sample panel. FSAR Tier 2 Section 9.3.2.2.2, “Component Description,” discusses first- and second-stage cooling for samples streams over approximately 100 degrees F but also notes that second-stage cooling is provided when the sample stream is directed to an analysis panel. Proposed FSAR Tier 2 Table 12.4-8 does not include expected dose from post-accident fluid that is contained in sample coolers. Therefore, the NRC staff issued RAI 9278, Question 12.03-39 (ML18028A004), asking the applicant to provide the methods, models, and assumptions used to model the source term from the sample cooler heat exchangers that the analytical models use to determine the dose for obtaining RCS samples following an accident. On May 16, 2018, the applicant provided responses to RAI 9278 (ADAMS Accession No. ML18136A869). However, in an April 24, 2018, response to RAI 9261 (ADAMS Accession No. ML18114A370); an April 24, 2018, response to RAI 9268 (ADAMS Accession No. ML18114A371); and a May 18, 2018, response to RAI 8837 (ADAMS Accession No. ML18138A383), the applicant specified that it plans to submit a revised TR-0915-17565, Revision 3, which will include the revised accident source term methodology. The changes to the accident source term methodology are likely to affect accident source terms and dose rates and may affect the response to and evaluation of this question. Therefore, the NRC staff will conduct a full review of this response following the submittal of the revised TR-0915-17565, Revision 3. As a result, **the NRC staff is currently tracking RAI 9278, Question 12.03-39, as an open item**.

FSAR Tier 2, Revision 0, Section 9.3.2.2.3, and proposed FSAR Tier 2 Section 12.4.1.8 state that the sample line purge volume may be directed to the LWRS. FSAR Tier 2 Figure 9.3.3-1, “Radioactive Waste Drain System Diagram,” shows that the drains in the RXB go to sumps 41 A/B–46 A/B. FSAR Tier 2 Section 9.3.2.2.3 notes that additional operator actions, such as raising containment pressure, may be required to allow sampling. In addition, FSAR Tier 2 Section 9.3.2.2.3 notes that sample fluids greater than 100 degrees F may require cooling. FSAR Tier 2 Section 9.3.2.2.3 and the response to RAI 8775, Question 12.03-1 (ADAMS Accession No. ML17177A699), do not address any actions needed to ensure proper operation of these systems during post-accident conditions. As noted above, NUREG-0737 lists other areas that should be considered in determining the vital areas and stipulates that, if these areas are not considered vital areas, justification should be provided for not including them. Therefore, the NRC staff issued RAI 9278, Question 12.03-40 (ML18028A004), asking the applicant to describe the radiological impact on affected areas of the plant from collecting RCS samples following an accident. On May 16, 2018, the applicant provided responses to RAI 9278 (ADAMS Accession No. ML18136A869). However, in an April 24, 2018, response to RAI 9261 (ADAMS Accession No. ML18114A370); an April 24, 2018, response to RAI 9268 (ADAMS Accession No. ML18114A371); and a May 18, 2018, response to RAI 8837 (ADAMS Accession No. ML18138A383), the applicant specified that it plans to submit a revised TR-0915-17565, Revision 3, which will include the revised accident source term methodology. The changes to the accident source term methodology are likely to affect accident source terms and dose rates and may affect the response to and evaluation of this question. Therefore, the NRC staff will conduct a full review of this response following the submittal of the revised TR-0915-17565, Revision 3. As a result, **the NRC staff is currently tracking RAI 9278, Question 12.03-40, as an open item**.

FSAR Tier 2, Section 12.4, states that the dose assessment presented in FSAR Tier 2 Section 12.4 includes the estimated radiation exposures to plant personnel who perform work activities involving normal operations, maintenance and inspections, refueling activities, and waste handling using the methodology in RG 8.19 to demonstrate that the facility design is consistent with 10 CFR Part 20. In addition, 10 CFR 52.47(a)(22) requires applicants to demonstrate how the plant design incorporates operating experience insights. To estimate the OREs for the NuScale facility, various work activities and work durations are compiled along with the expected significant (greater than 0.1 mrem/h (0.001 mSv/h) radiation fields that would be encountered. FSAR Tier 2 Section 12.4.1 states that FSAR Tier 2 Table 12.4-5, “Occupational Dose Estimates from Special Maintenance,” and FSAR Tier 2 Table 12.4-7, “Occupational Dose Estimates from Refueling Activities,” list the ORE dose estimates for special maintenance and refueling activities. The NRC staff used information from the RPAC Chapter 12 Phase I audit (see “Audit Report - Regulatory Audit of Radiation Protection and Environmental Qualification for NuScale Power Design,” (ADAMS Accession No. ML18124A182)), in its review of the bases for the estimated doses for the work activities described in these tables. The NRC staff observed that the dose rates used for performing the dose estimates did not appear to be consistent with 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). For example, the NRC staff noted that the dose rates assumed for work near the SGs did not appear to be within an order of magnitude of the median value depicted in EPRI TR-1015119. Although the NRC staff cannot quantitatively assess the change in dose rates associated with the smaller NuScale design, it did qualitatively consider the relative size of the plant on the assumed dose rates. RG 8.19 states that plant experience, which is available from industry groups like EPRI, provides useful information for performing this assessment. 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. 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 man-rem 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. The ORE dose estimates provided in FSAR Tier 2, Section 12.4, do not appear to have been based on relevant plant data, such as the cited industry experience; therefore, the component dose rates used to perform the ALARA analysis appear to be biased low. The potential result is that the method for identifying possible changes to the plant design to minimize ORE, as required by 10 CFR 20.1101(b), may improperly determine that the dose savings do not justify the proposed design change. Therefore, the NRC staff issued RAI 9293, Question 12.03-16 (ML18026A633), asking the applicant to provide additional information on and justification of the dose rates assumed for the performance of the ALARA dose assessment.

In its response to RAI 9293, Question 12.03-16 (ADAMS Accession No. ML18080A113), the applicant specified that both procedures and engineering controls will be used to maintain exposures to radiation as low as practical. The applicant also stated that it believes that 10 CFR 52.47(a)(22) only applies to NRC generic letters and bulletins issued after the most recent revision of the applicable standard review plan and 6 months before the docket date of the application and that none of the cited references, including EPRI TR-1015119, meet the definition of operating experience. However, 10 CFR 52.47(a)(22) is not limited only to NRC generic letters and bulletins; therefore, the NRC staff does not agree with the applicant’s assessment.

Although the applicant indicated that it believes that EPRI TR-1015119 does not apply, it indicated that it conducted a study that demonstrates that the estimated dose rates for an operator performing SG maintenance activities are appropriate for the NuScale design. The study was based on the approximate cold-leg channel head center average dose rate from EPRI TR-1015119, Figure 5-7. The applicant also specified that SG maintenance activities during a refueling outage are performed on the secondary side with personnel on the outside of the NPM. Because radiation emanating from the SG inside the NPM must travel through the reactor pressure vessel steel wall or through the reactor pressure vessel and CNV steel walls, the radiation exposure is significantly reduced. Therefore, the applicant concluded that it appropriately accounted for the radiocobalt buildup in the SGs and that the results of the OREs in FSAR Section 12.4 are appropriate.

The NRC staff used information in the applicant's response to RAI 9293, Question 12.03-16, and other operational experience regarding dose rates expected at similar locations in the NuScale design, to perform confirmatory calculations using the MicroShield modeling code consistent with source terms, materials and resultant dose rates provided by the applicant. Although the NRC staff disagrees with the applicant's rationale, the RAI response and the confirmatory calculations provide the NRC staff with reasonable assurance that the FSAR Tier 2 adequately accounts for outage ORE and that the outage ORE meets the applicable regulatory requirements. Based on this, the NRC staff found the applicant's response to be acceptable, and RAI 9293, Question 12.03-16, is considered closed.

12.4.5 Combined License Information Items

Table 12.4-1 lists COL information item numbers and descriptions related to dose assessment from FSAR Tier 2, Table 1.8-2.

Table 12.4-1 NuScale COL Information Items for Section 12.4

COL Item No.	Description	FSAR Tier 2 Section
12.4-1	A COL applicant that references the NuScale Power Plant design certification will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.	12.4.1

12.4.6 Conclusion

As described above, the NRC staff has reviewed the applicant's submittal against the requirements of 10 CFR Part 20 as it relates to: limits on doses to people in restricted areas, and the applicable requirements, including 10 CFR Part 19; sources of direct radiation exposure to members of the public, including the generally applicable environmental radiation standards in 40 CFR Part 190; 10 CFR 50.49(e)(4); 10 CFR 52.47(a)(5); 10 CFR 52.47(a)(22); and 10 CFR Part 50, Appendix A, GDC 4, GDC 19 and GDC 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 52.47(a)(25) and 10 CFR 52.47(a)(26), as they relate to the use of design interfaces for portions of the certified design that the NRC expects the COL applicant to implement.

With the exception of the open items and pending resolution of the confirmatory items discussed above, the NRC staff has determined that the NuScale design meets the applicable requirements discussed in Section 12.4.

The applicant specified that it plans to submit a revised TR-0915-17565, Revision 3, which will include the revised accident source term methodology. The changes to the accident source term methodology are likely to affect accident source terms and dose rates and may affect the response to and evaluation of this question. Therefore, the NRC staff will conduct a full review of this response following the submittal of TR-0915-17565, Revision 3.

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 and the qualifications of the personnel responsible for conducting various aspects of the radiation protection program and 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 ORE will be ALARA, must be in place. These procedures and methods include those used in normal operation, refueling, in-service 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 Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: 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: FSAR Tier 2, Chapter 16, Section 5.7, addresses technical specifications for the control of high-radiation areas.

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

12.5.3 Regulatory Basis

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

- 10 CFR 19.12, “Instruction to Workers,” as it relates to keeping workers informed about the storage, transfer, or use of radioactive materials or radiation and instructing them about the risk associated with ORE, necessary precautions, procedures to reduce exposures, and the purpose and function of the protective devices used.

- 10 CFR 19.13, as it relates to requirements for informing workers of the results of their individual monitoring.

- 10 CFR 20.1101, as it relates to (1) development, documentation, and implementation of a radiation protection program, (2) the use of procedures and controls to achieve doses to workers and the public that are ALARA, as defined in 10 CFR 20.1003, and (3) the review and audit of the content and implementation of the radiation protection program.

- 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, and 10 CFR 20.1204, as they relate to demonstrating compliance with internal and external dose limits.

- 10 CFR 20.1206 and 10 CFR 20.2105, “Records of Planned Special Exposures,” as they relate to the authorization, control, and documentation of planned special exposures to adult workers.

- 10 CFR 20.1207, as it relates to control of occupational radiation doses received by minors.

- 10 CFR 20.1208, “Dose Equivalent to an Embryo/Fetus,” as it relates to control of radiation doses received by the embryo or fetus of a declared pregnant worker.

- 10 CFR 20.1301 and 10 CFR 20.1302, as they relate to controlling radiation doses to individual members of the public and the maximum dose rate in unrestricted areas.

- 10 CFR 20.1406, as it relates to the facility design and procedures for operation of the plant for minimizing contamination of the facility site.

- 10 CFR 20.1501, “General,” as it relates to the performance of surveys to comply with the regulations in 10 CFR Part 20.

- 10 CFR 20.1502, “Conditions Requiring Individual Monitoring of External and Internal Occupational Dose,” as it relates to requirements for monitoring individual occupational exposure to radiation.

- 10 CFR 20.1501(c), as it relates to the calibration of radiation protection instruments used for quantitative radiation measurements.

- 10 CFR 20.1601, 10 CFR 20.1602, and 10 CFR 20.1901, as they relate to the identification of potential sources of radiation exposure and the controls of access to and work within areas of the facility with a high potential for radiation exposure.

- 10 CFR 20.1701 and 10 CFR 20.1702, as they relate to controlling the concentrations and limiting the intake of radioactive materials in the air.

- 10 CFR 20.1703, “Use of Individual Respiratory Protection Equipment,” as it relates to the use of respiratory protective equipment to limit the intake of radioactive material.

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

- 10 CFR 20.1802, “Control of Material Not in Storage,” as it relates to controlling licensed material that is not in storage.

- 10 CFR 20.1902; 10 CFR 20.1903; 10 CFR 20.1904; and 10 CFR 20.1905, “Exemptions to Labeling Requirements,” as they relate to the posting of, and control of access to, radiation areas, high-radiation areas, VHRAs, airborne radioactivity areas, and other indicators necessary to identify and quantify the presence of radioactive materials in an area.

- 10 CFR 20.1906, “Procedures for Receiving and Opening Packages,” as it relates to appropriate handling of packages that contain certain quantities of radioactive materials

- 10 CFR 20.2001, “General Requirements,” and 10 CFR 20.2006, “Transfer for Disposal and Manifests,” as they relate to the transfer of radioactive materials and the disposal of low-level radioactive waste.

- 10 CFR 20.2101, “General Provisions”; 10 CFR 20.2102, “Records of Radiation Protection Programs”; 10 CFR 20.2103, “Records of Surveys”; 10 CFR 20.2104, “Determination of Prior Occupational Dose”; 10 CFR 20.2105; 10 CFR 20.2106, “Records of Individual Monitoring Results”; 10 CFR 20.2107, “Records of Dose to Individual Members of the Public”; and 10 CFR 20.2110, “Form of Records,” as they relate to maintaining records of: individuals who are provided with personnel monitoring equipment, individuals who are exposed to radiation, and the radiation protection program, including surveys.

- 10 CFR 20.2201, “Reports of Theft or Loss of Licensed Material,” as it relates to reports to the NRC required from licensees immediately after they become aware of any loss or theft of certain quantities of licensed material.

- 10 CFR 20.2202, “Notification of Incidents”; 10 CFR 20.2203, “Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Material Exceeding the Constraints or Limits”; 10 CFR 20.2204, “Reports of Planned Special Exposures”; and 10 CFR 20.2205, “Reports to Individuals of Exceeding Dose Limits,” as they relate to requirements for reports to the NRC concerning individual exposures that exceed regulatory limits, incidents requiring notification, levels of radiation or concentrations of radioactive materials in excess of certain values, and planned special exposures.

- 10 CFR 20.2206, “Reports of Individual Monitoring,” and 10 CFR 19.13, “Notifications and Reports to Individuals,” as they relate to requirements for licensees to report the results of individual monitoring and inform workers of the results of their individual monitoring.

- 10 CFR 50.34(f)(2)(viii) and 10 CFR 50.34(f)(2)(xxvii),as they relate to monitoring in-plant radiation and airborne radioactivity for routine and accident conditions.

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

- 10 CFR 50.65(a), as it relates to providing reasonable assurance that SSCs important to safety, including those that are relied on to mitigate accidents or transients or that are used in plant emergency operating procedures (i.e., the radiation monitors or radiation protection features described in DSRS Section 12.3–12.4), are capable of fulfilling their intended functions

- 10 CFR 50.120, “Training and Qualification of Nuclear Power Plant Personnel,” as it relates to the provisions and requirements for training radiation protection technicians

- 10 CFR Part 50, Appendix A, GDC 64, “Monitoring Radioactivity Releases,” as it relates to monitoring the reactor containment atmosphere and spaces that contain components for the recirculation of fluids resulting from a loss-of-coolant accident.

- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” and Subpart H, “Quality Assurance,” of 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” as they relate to quality assurance programs.

- 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material”; 10 CFR Part 40, “Domestic Licensing of Source Material”; and 10 CFR Part 70, as they relate to the requirements for the receipt, storage, and use of byproduct, source, and special nuclear material.

- 10 CFR Part 37, “Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material,” as it relates to the physical protection requirements for tracking the aggregation of radioactive material.

- 10 CFR 71.5, “Transportation of Licensed Material,” and 10 CFR Part 71, Subpart G, “Operating Controls and Procedures,” as they relate to the control of licensed radioactive material during packaging and transportation.

- 10 CFR Part 20, Subpart K, “Waste Disposal,” as it relates to the transfer of low-level radioactive materials and waste.

- 29 CFR 1910.134, “Respiratory Protection,” as it relates to the program for the use of mixed hazard respiratory protection in the radiologically controlled areas of the plant.

DSRS Section 12.5 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 1.8, as it relates to compliance with NRC regulations on the qualification of nuclear power plant personnel.

- RG 1.33, as it relates to compliance with NRC quality assurance regulatory requirements during nuclear power plant operations.

- RG 1.97 and DSRS Section 7.2 (for safety-related equipment), as they relate to compliance with NRC regulations to provide instrumentation to monitor plant variables and systems during and following an accident.

- RG 1.97, the SRP, and DSRS Section 11.6, as they relate to compliance with NRC regulations that require the licensee to provide instrumentation for monitoring plant variables and systems that are important to safety during and following an accident.

- RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” as it relates to providing reasonable assurance that safety-related SSCs and nonsafety-related SSCs that are relied on to mitigate accidents or transients or are used in plant emergency operating procedures are capable of performing their intended functions.

- RG 8.2, as it relates to general information on radiation monitoring programs for administrative practices.

- RG 8.4, “Personnel Monitoring Device—Direct-Reading Pocket Dosimeters,” as it relates to standards for direct- and indirect-reading pocket dosimeters used for personnel dose or dose rate measurements.

- RG 8.7, “Instructions for Recording and Reporting Occupational Radiation Exposure Data,” as it relates to the specification of records necessary to describe the ORE of individuals and to the conditions under which the exposure may occur.

- RG 8.8, as it relates to meeting the requirements of 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 by providing radiation protection information pertaining to actions taken during the design, construction, operation, and decommissioning of a facility to ensure that ORE remains ALARA.

- RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program,” as it relates to appropriate concepts, models, equations, and assumptions to be used in determining the extent of an individual’s intake of radioactive materials and the resulting committed internal dose.

- RG 8.10, as it relates to meeting the requirements of 10 CFR 20.1101(b) and the definition of ALARA in 10 CFR 20.1003 concerning commitment by the applicant’s management and vigilance by the radiation protection manager and the radiation protection NRC staff to maintain ORE ALARA.

- RG 8.13, “Instruction Concerning Prenatal Radiation Exposure,” as it relates to the description of the instruction that must be provided for biological risks to embryos or fetuses resulting from prenatal ORE.

- RG 8.15, as it relates to elements of acceptable respiratory protection programs.

- RG 8.27, as it relates to a radiation protection training and retraining program consistent with the ALARA objective and acceptable to the NRC staff for meeting the training requirement of 10 CFR Part 19.12.

- RG 8.28, “Audible-Alarm Dosimeters,” as it relates to the appropriate use of audible-alarm dosimeters and the conditions under which they should not be relied on to perform their intended function.

- RG 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure,” as it relates to providing appropriate instruction to workers, consistent with the requirements of 10 CFR 19.12, on the risks to individuals who might be exposed to ORE.

- RG 8.34, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses,” as it relates to criteria acceptable to the NRC staff that licensees may use to determine when monitoring is required and to methods acceptable to the NRC staff for calculating occupational doses when the intake is known.

- RG 8.35, “Planned Special Exposures,” as it relates to guidance on the conditions and prerequisites for permitting planned special exposures in accordance with 10 CFR Part 20 and the associated specific monitoring and reporting requirements.

- RG 8.36, “Radiation Dose to the Embryo/Fetus,” as it relates to determining the total radiation dose to the embryo or fetus as the sum of the deep-dose equivalent to, and the dose to the embryo or fetus from, the intakes of the declared pregnant worker.

- RG 8.38, as it relates to guidance on acceptable methods to control access to high-radiation areas and VHRAs in nuclear power plants that follows the requirements in 10 CFR Part 20.

- NUREG/CR-0041, “Manual of Respiratory Protection against Airborne Radioactive Material,” issued January 2001, as it relates to providing technical information to licensees on the appropriate application of respiratory protective devices for protection against airborne radioactive materials, including the selection and maintenance of equipment and personnel training.

- NUREG-0731, “Guidelines for Utility Management Structure and Technical Resources,” issued September 1980, as it relates to appropriate NRC staffing levels and technical expertise considered essential within a utility to properly support nuclear power plant operation.

- NUREG-0938, “Information for Establishing Bioassay Measurements and Evaluations of Tritium Exposure,” issued June 1983, as it relates to monitoring individuals for exposure to tritium.

- NUREG-1736, as it relates to the requirements for a radiation protection program (including a program review and audit) and compliance with 10 CFR Part 20.

- NUREG-2155, “Implementation Guidance for 10 CFR Part 37, ‘Physical Protection of Category 1 and Category 2 Quantities of Radioactive Material,’” Revision 1, issued January 2015, as it relates to the requirements for assessing the quantities of radioactive material (e.g., inventory of radioactive material for the purposes of identifying and controlling aggregated quantities).

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

- ANSI/ANS Std. 3.1-1993, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants,” reaffirmed 1999, as it relates to criteria for the selection, qualifications, responsibilities, and training of personnel in operating and support organizations as appropriate for the safe and efficient operation of nuclear power plants.

- ANSI/HPS Std. N13.6-1999, “Practice for Occupational Radiation Exposure Records Systems,” as it relates to guidance to the employer for the systematic generation and retention of records related to ORE.

- ANSI/HPS Std. N13.11-2009, “Personnel Dosimetry Performance—Criteria for Testing,” as it relates to the performance criteria for personal radiation dosimeters that require processing.

- ANSI/HPS Std. N13.14-1994, “Internal Dosimetry Programs for Tritium Exposure—Minimum Requirements,” as it relates to personnel monitoring.

- ANSI/HPS Std. N13.30-2011, “Performance Criteria for Radiobioassay,” as it relates to detection and dosimetry of internally deposited radionuclides.

- ANSI/HPS Std. N13.42-1997, “Internal Dosimetry for Mixed Fission Activation Products,” as it relates to monitoring radiation dose from internally deposited radionuclides.

- ANSI/Institute of Electrical and Electronics Engineers (IEEE) Std. 309-1999, “Test Procedure for Geiger-Mueller Counters,” reaffirmed 2006, as it relates to guidance on the specification of test conditions, such as the associated electronic circuitry, environment, and counting rate, to ensure the appropriate evaluation of operating characteristics.

- ANSI Std. N42.20-2003, “Performance Criteria for Active Personnel Radiation Monitors,” as it relates to the accuracy and overall performance of personnel radiation monitors.

- ANSI Std. N42.17A-2003, “Performance Specifications for Health Physics Instrumentation—Portable Instrumentation for Use in Normal Environmental Conditions”; ANSI Std. N42.17B-R2005, “Performance Specifications for Health Physics Instrumentation—Occupational Airborne Radioactivity Monitoring Instrumentation”; ANSI Std. N42.17C-R2005, “Performance Specifications for Health Physics Instrumentation—Portable Instrumentation for Use in Extreme Environmental Conditions”; and IEEE Std. N42.30-2002, “American National Standard for Performance Specification for Tritium Monitors,” as they relate to the accuracy and overall performance of portable survey instruments.

- ANSI Std. N323A-1997, “American National Standard Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments”; ANSI Std. N323B-2003, “Radiation Protection Instrumentation Test and Calibration, Portable Survey Instrumentation for Near Background Operation”; ANSI Std. N323C-2009, “Radiation Protection Instrumentation Test and Calibration—Air Monitoring Instruments”; and ANSI Std. N323D-2002, “Installed Radiation Protection Instrumentation,” as they relate to the calibration and maintenance of portable radiation survey instruments.

- Memorandum from L.W. Camper to D.B. Matthews and E.E. Collins, dated October 10, 2006 (ADAMS Accession No. ML062620355), and NUREG/CR-3587 (ADAMS Accession No. ML081360413), as they relate to operating programs that facilitate decommissioning

- IEEE Std. N42.12-1994, “American National Standard Calibration and Usage of Thallium Activated Sodium Iodide Detector Systems for Assay of Radionuclides”; IEEE Std. N42.14-1999, “Calibration and Use of Germanium Spectrometers for the Measurement of Gamma-Ray Emission Rates of Radionuclides”; and IEEE Std. N42.25-1997, “American National Standard Calibration and Usage of Alpha-Beta

Proportional Counters," as they relate to the calibration and maintenance of radiation protection laboratory instruments

12.5.4 Technical Evaluation

NuScale FSAR Tier 2, Section 12.5, states that the COL applicant must provide the radiation protection program. FSAR Tier 2 Section 12.1 states that the COL applicant must provide the ALARA program. FSAR Tier 2 Section 12.3 states that the COL applicant must provide programs to minimize contamination of the facility. The review of programs is beyond the scope of review conducted for a FSAR Tier 2.

12.5.5 Combined License Information Items

Table 12.5-1 lists COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2, that are related to the operational radiation protection programs that the COL applicant will describe to ensure that OREs and public radiation exposures are ALARA.

Table 12.5-1 NuScale COL Information Items for Section 12.5

COL Item No.	Description	FSAR Tier 2 Section
12.5-1	A COL applicant that references the NuScale Power Plant design certification 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

12.5.6 Conclusion

The NRC staff does not review programs during the design phase, therefore it is acceptable for COL applicants to address the operational considerations as described in the COL item applicable to this section. The NRC staff will determine compliance with the requirements of 10 CFR Part 20 in these areas during the COL review.