12 RADIATION PROTECTION

This chapter of the final safety evaluation report (FSER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 12, "Radiation Protection," of the NuScale Power, LLC (hereinafter referred to as the applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 5 of the DCA, dated July 29, 2020 (Agencywide Document Access and Management System (ADAMS), Accession No. ML20225A071). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the DCA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this safety evaluation to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the DCA and not converted.

Information contained in DCA Part 2, Tier 2, Chapter 12, addresses radiation protection policy considerations, design considerations, and operational considerations applied during the design process. Where appropriate, combined license (COL) action items and certification requirements and restrictions were described. Radiation sources, including the quantities of contained solid and liquid radioactive material and airborne radioactive material, and the associated bases are described. This chapter of the application describes the radiation protection design features provided to protect members of the public, the workers, and the environment, including facility design features; radiation shielding material and quantities; ventilation components and flow rates; area radiation monitoring and airborne radiation monitoring equipment; and features of the design provided to minimize contamination of the environment, minimize the generation of waste, and facilitate decommissioning. The application describes the methods used and the resultant dose estimates expected for activities that are expected to occur during normal operation including during refueling, following accidents, and during construction.

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

12.1.1 Introduction

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

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

The ALARA principles are applied during the design process to identify and describe design features and specifications intended to limit and minimize the amount of radiation exposure to members of the public from contained radiation sources within the plant during operation;

radiation exposure from operating modules to workers constructing or installing additional modules; and radiation exposure to occupational workers during plant operation, AOOs, maintenance and inspection activities, and accidents. Operational program elements are used to complement design features and specifications to limit and minimize radiation exposure.

12.1.2 Summary of Application

DCA Part 2, Tier 1: There is no Tier 1 information related to radiation protection for this section.

DCA Part 2, Tier 2: The applicant provided DCA Part 2, Tier 2, Section 12.1, “Ensuring that Occupational Radiation Exposures Are As Low As Reasonably Achievable,” 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, inservice 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.

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

- 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 DCA Part 2, Tier 2, Section 12.1.

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

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

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

12.1.3 Regulatory Basis

The relevant requirements of the Commission’s regulations for assuring that occupation radiation exposure is ALARA, associated acceptance criteria, and review interfaces with other sections of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for

Nuclear Power Plants: LWR Edition" (SRP), appear in SRP Section 12.1, "Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable," Revision 4, issued September 2013 (ADAMS Accession No. ML13151A061). The following summarizes the regulatory requirements:

- 10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," as it relates to keeping workers who receive 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 52.47(b)(1), which requires a DCA Part 2 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 design certification (DC) has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act of 1954, as amended (AEA), and NRC regulations

- 10 CFR 20.1101, "Radiation Protection Programs," and the definition of ALARA in 10 CFR 20.1003, "Definitions," as they relate to 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

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

- Regulatory Guide (RG) 1.8, "Qualifications and Training for Nuclear Power Plant Personnel"

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

- RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable"

- RG 8.10, "Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As is Reasonably Achievable"

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

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

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 (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," issued October 2009 (ADAMS Accession No. ML093220530), and the associated NRC SER

- 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 DCA Part 2, Tier 2, Section 12.1, in accordance with the review procedures in SRP Section 12.1. The results of the NRC staff's review are provided below.

Policy Considerations

In DCA Part 2, 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 of 10 CFR 20.1101(b) and the training requirements of 10 CFR 19.12, "Instruction to Workers," 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 and the requirements of 10 CFR Part 20. DCA Part 2, 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 that encompasses the ALARA concept and includes provisions for maintaining radiation doses and intakes of radioactive materials ALARA for both occupational workers and members of the public. The detailed policy considerations for overall plant operations and implementation of such a radiation protection program are outside the scope of the DC review. 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 directs 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 applicant used an interdisciplinary team of experienced engineers to identify and evaluate existing operating plant experience for evaluating the guidance contained in RG 8.8 with respect to meeting the requirements, including ALARA, of 10 CFR Part 20. The application identified the types of design and operating considerations subsequently used to inform the design and specifications presented in the application.

Operational Considerations

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

Radiation Protection Considerations

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

12.1.5 Combined License Information Items

Table 12.1-1 lists the COL information item numbers and descriptions related to radiation protection from DCA Part 2, Tier 2, Section 12.1.3.

Table 12.1-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 12.1

COL Item No.	Description	DCA Part 2, 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, as low as reasonably achievable (ALARA).	12.1.3

12.1.6 Conclusion

Based on the information supplied by the applicant as described above, the NRC staff concludes that the general NuScale SMR design features described in DCA Part 2, Tier 2, Section 12.1, meet the acceptance criteria of SRP Section 12.1 and the applicable requirements of 10 CFR Part 19, 10 CFR Part 20, and 10 CFR 52.47(b)(1).

12.2 Radiation Sources

12.2.1 Introduction

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

12.2.2 Summary of Application

DCA Part 2, Tier 1: There is no Tier 1 information related to radiation protection for this section.

DCA Part 2, Tier 2: The applicant described onsite radiation sources primarily in DCA Part 2, Tier 2, Section 12.2, "Radiation Protection," which is summarized, in part, as follows:

- DCA Part 2, 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; plant sampling systems; and solid, liquid, and gaseous waste management systems. These sources emit gamma radiation, which requires shielding consideration and assessment of the dose to occupational workers and members of the public.

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

- In addition, DCA Part 2, Tier 2, Section 12.2, provides information on post-accident source terms in the NuScale design. DCA Part 2, Tier 2, Chapter 15, provides additional information on the post-accident source terms, and the accident source term methodology appears in NuScale TR 0915 17565-NP-A, "Accident Source Term

Methodology," Revision 4 (ADAMS Accession No. ML20057G132). The post-accident source terms are used to evaluate the doses in the main control room (MCR), the doses to initiate combustible gas monitoring, and the doses to equipment important to safety (i.e., DCA Part 2, 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 DCA Part 2, 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.

Technical Reports:

- NuScale TR-1116-52065, "Effluent Release (GALE Replacement) Methodology and Results," Revision 1

- NuScale TR-0116-20781-P, "Fluence Calculation Methodology and Results," Revision 1

Topical Reports:

- NuScale TR-0915-17565-NP-A, "Accident Source Term Methodology," Revision 4 (ADAMS Accession No. ML20057G132)

12.2.3 Regulatory Basis

The relevant requirements of the Commission's regulations for assuring that occupation radiation exposure is ALARA are described below. The associated acceptance criteria and the review interfaces with other SRP or design-specific review standard (DSRS) sections are described in Section 12.2, "Radiation Sources," of "Design Specific Review Standard for NuScale SMR Design" (ADAMS Accession No. ML15350A320).

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 the 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

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

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

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

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

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

- 10 CFR 52.47(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 .117, “Protection against Extreme Wind Events and Missiles for Nuclear Power Plants,” as they relate to the radiological criteria for classification and protection of nonradioactive waste SSCs that contain radioactive material

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

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

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

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

DSRS Section 12.1 specifies that the applicant modify the methods and data in ANSI/ANS Std. 18.1-1999 to reflect NuScale-specific design attributes. The methods and data in ANSI/ANS Std. 18.1-1999 were developed using relevant industry operating experience.

12.2.4 Technical Evaluation

The NRC staff evaluated the information in DCA Part 2, Tier 2, Section 12.2, against the applicable regulations and the guidance in DSRS Section 12.2 to verify that DCA Part 2 accurately described contained sources, including byproduct, source, and special nuclear materials, and other radiologically significant sources. The NRC staff reviewed the radiologically significant radiation sources described in the application that were expected to be generated during normal operations, during potential AOOs, and as a result of potential accidents. The NRC staff reviewed the methods, models, and assumptions used as the basis for establishing the kinds and quantities of radioactive materials or a radiation environment presented in the application. The specific areas of review include contained sources and airborne radioactive sources. The NRC staff used the kinds and quantities of radioactive materials or the radiation environment present, as described in DCA Part 2, Tier 2, Section 12.2, to evaluate the SSCs and design features described in DCA Part 2, Tier 2, Section 3.11 and Section 12.3, and other sections of DCA Part 2, Tier 2, that are provided to protect equipment, workers, and members of the public from the effects of radiation.

12.2.4.1 Contained Sources

In DCA Part 2, Tier 2, Section 12.2.1, “Contained Sources,” the applicant described the source terms used for determining the radiation shielding, facility design features, and radiation zoning during normal full-power operation, including AOOs, and evolutions such as refueling, as well as sources of radiation exposure to equipment following potential accidents. DCA Part 2, Tier 2, Section 12.2.1, describes the radiologically significant contained sources of radiation that are used as the basis for designing the radiation protection program and shield design calculations. For each of these contained sources, the applicant provided either the source strength by energy group or the associated activity levels listed by isotope (in most cases, the source terms were provided by activity level).

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

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 reactor coolant system (RCS) isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to the operation for a full fuel cycle at the TS limits for halogens (iodine (I)-131 dose equivalent) and noble gases (xenon (Xe)-133 dose equivalent).

The applicant used a DBFFF of 0.066 percent as the basis calculating the radioactive material content of contained source inventories for SSCs, such as liquid tanks, demineralizers, filters and charcoal beds, that are subsequently used as inputs for the normal operation radiation shielding design calculations, ventilation system design calculations, and normal operation personnel dose assessment. The NRC staff independently confirmed that the 0.066-percent failed fuel fraction for the DBFFF provides radionuclide concentrations that are consistent with the information in TS 3.4.8, “RCS Specific Activity,” which provides RCS coolant concentration limits on dose-equivalent I-131 and dose-equivalent Xe-133. Since the RCS specific activity

concentration limits of TS 3.4.8 correspond to the 0.066-percent failed fuel fraction value referenced by the applicant in DCA Part 2, Tier 2, Chapter 12, and the 0.066-percent failed fuel fraction was used to determine the kinds and quantities of radioactive material in plant systems, the NRC staff finds the use of this failed fuel fraction value to be acceptable.

The NRC staff reviewed and confirmed that the applicant provided source terms for the radiologically significant contained sources of radioactivity in DCA Part 2, Tier 2, Section 12.2. These sources include, but are not limited to, the reactor core, CVCS, reactor pool cooling, spent fuel pool (SFP) cooling, pool cleanup, pool surge control systems, liquid radioactive waste system (LRWS), gaseous radioactive waste system (GRWS), solid radioactive waste system (SRWS), spent fuel, and activated components (such as self-powered neutron detectors (SPNDs)).

During the review of DCA Revision 3, the NRC staff noticed that Part 2, Tier 2, Table 12.2-24, "In-Core Instrumentation Gamma Spectra," changed to 1 cycle of irradiation from 30 cycles, as listed in DCA Revision 2. Conversations with the applicant indicated that the applicant made these changes in response to staff questions about the neutron fluence values in the reactor core. The applicant further stated that the values present in DCA Part 2, Tier 2, Revision 3 Table 12.2-24, reflect limiting dose consequence condition (max energy/time) and the local neutron energy spectra and flux. However, the NRC staff found the gamma energy emission rates were uniformly higher for the 30-cycle irradiation duration for each of the time intervals (discharge, after 3 days, and after 30 days) than for the single-cycle irradiation period. Specifically, MCNP dose rate calculations indicated that the 30 irradiation cycles at 3 days dose rate was over 5 times higher than for the single irradiation at 3 days. The NRC staff review of industry data regarding the expected life time of the SPNDs indicated that the SPNDs should last five or more cycles. In a letter dated November 13, 2019 (ADAMS Accession No. ML19317D709), the applicant agreed to revise DCA Part 2, Tier 2, Revision 3, Table 12.2-24, to list the activity at the discharge for one cycle and the activity at 3 days following 30 cycles of irradiation in the core. This method of listing the radioactive content provides the maximum values that are expected to be present during use (i.e., one irradiation cycle immediately after discharge) and includes irradiation beyond the end of the expected life of the detectors (i.e., five cycles). Therefore, the NRC staff finds that the applicant adequately identified the kinds and quantities of radioactive material associated with the in-core neutron detectors, consistent with the requirements of 10 CFR 2.47(a)(5).

The NRC staff reviewed the methods, models, and assumptions used by the applicant to determine the radionuclide concentrations in the RCS, the connected systems, and the downstream SSCs. The NRC staff used a combination of the information contained in DCA Part 2, Tier 2, Section 12.2; calculations performed by the NRC staff; and audits (see ADAMS Accession Nos. ML18124A182, ML18348A966, and ML19203A043) to review the methods, models, and assumptions used by the applicant to derive source terms used in DCA Part 2, Tier 2, Section 12.2. The NuScale facility design includes simultaneous operation of 12 NuScale Power Modules (NPM). SSCs, such as the spent resin storage tanks (SRSTs) and the phase separator tanks (PSTs), are designed to receive radioactive material from multiple modules. In addition, to allow for greater operational flexibility, some SSCs, like the SRSTs and PSTs, are relatively larger than the corresponding SSCs in the current operating fleet. Using a risk-informed approach, the NRC staff based its review of the radioactive material content of only a single NPM operating at the TS coolant specific activity limit, with the other 11 units operating with normally expected coolant activity concentrations. As a result, the methods the NRC staff used to evaluate radioactive material content of SSCs was adjusted to account for the dilution from other waste streams and radiological decay of the contents, as the SSCs were

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

The NRC staff reviewed the geometries, concentrations, branching ratios, daughter product concentrations and emitted radiation, decay half-lives, radiation emission spectra and fluences, and attenuation coefficients. The NRC staff verified that the applicant was accounting for the changes in radionuclide concentrations (such as cesium (Cs)-137 and barium (Ba)-137m) resulting from the decay and buildup of radionuclides. The NRC staff performed confirmatory calculations using programs such as Oak Ridge National Laboratory's (ORNL's) "Standardized Computer Analyses for Licensing Evaluation" (SCALE) (see "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design"), ORNL's "Oak Ridge Isotope GENeration" (ORIGEN), MicroShield, and the NRC's "RADionuclide Transport, Removal and Dose" (RADTRAD), to verify that the applicant either properly determined the concentration of radionuclides or, where it did not explicitly determine the radionuclide concentrations resulting from decay, that the applicant applied NRC-approved methods to compensate.

However, the NRC staff identified two areas in DCA Part 2, Tier 2, Revision 3, where the Ba-137m concentrations were not commensurate with the stated Cs-137 concentrations: the "Pool Surge Control Tank" column of Table 12.2-10, "Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup and Pool Surge Control System Component Source Terms—Radionuclide Content," and columns "LCW Collection Tank (Ci)" and "HCW Collection Tank (Ci)," of Table 12.2-13a, "Liquid Radioactive Waste System Component Source Terms—Radionuclide Content." The principle gamma emission from the decay of Cs-137 is actually emitted from the Ba-137m daughter. Based on the 2.55-minute half-life of Ba-137m, the concentration of Ba-137m should be approximately 95 percent of the Cs-137 concentration. Based on the low radiological significance of the total dose from the differences in reported radioactive concentration of Ba-137m, the NRC staff determined that the radiological assessment of the source terms in these SSCs is acceptable.

The NRC staff reviewed the geometric description of sources described in DCA Part 2, Tier 2, Section 12.2, to ensure that sources were appropriately distributed within the SSC that was expected to contain the source (e.g., in a volume the size and shape of a demineralizer within a room, versus the entire volume of the room containing the demineralizer bed). The NRC staff also reviewed the assumed contents of the SSC versus the stated capacity of the SSC. The geometric arrangement of the sources may have a significant impact on the resultant dose rates outside of the SSC. Based on this review, the NRC staff determined that the applicant is appropriately considering the geometrical arrangement of SSCs containing radioactive material.

During the NRC staff review of the source term used as the basis for the high-integrity container (HIC) storage room, an area identified as containing a sufficient amount of radioactive material to result in dose rates of 5 grays per hour (Gy/h) (500 radiation absorbed doses per hour (rad/h)), with no upper limit specified, the NRC staff noted that the radioactive content of the HIC storage room appeared to be inconsistent with the available volume of the room. Since this is a radiologically significant area of the RWB, the NRC staff reviewed the methods, models, and assumptions used by the applicant to determine the amount of radioactive material that could be present in the room. The NRC staff determined that the applicant based the radionuclide inventory in the room on the assumption that there would be a limited number of HICs in the center of the room that were all full of resin that resulted from the operation at the DBFFF.

Using the risk-informed approach, as described above, the NRC staff performed independent calculations using input streams to the HICs that were more representative of the probable inputs. The staff determined that the method used by the applicant to estimate the radiological significance of the radioactive material in the HIC storage room, while not considering the full storage volume of the room, was sufficient for the purpose of performing the required shielding analysis. The NRC staff determined that the description of the source term in the HIC storage room is consistent with the requirement of 10 CFR 52.47(a)(5) to identify the kinds and quantities of radioactive material sufficiently well to assure that design features for controlling radiation exposure to within the limits of 10 CFR Part 20 can be implemented. Therefore, the staff finds this to be acceptable.

DCA Part 2, Tier 2, Table 12.2-12, provides decontamination factors (DFs) for the liquid radioactive waste granulated activated charcoal (GAC) unit. As explained by the applicant, in the absence of regulatory guidance for the application of DFs for GAC, the applicant used DFs that were discussed in the paper, "The Volume Reduction of Liquid Radioactive Waste by Combined Treatment Methods," referenced in International Atomic Energy Agency (IAEA)-TECDOC-1336, "Combined Methods for Liquid Radioactive Waste Treatment." The applicant noted that the GAC filter is not designed nor intended to collect radionuclides; however, because it could collect radionuclides, it was conservatively (for the purpose of local radiation shielding assessment) assumed to collect radionuclides at the rate indicated by the DFs. The applicant also stated that to ensure that the amount of radioactive material sent downstream was not underestimated, the radionuclide concentration in the outlet stream from the GAC filter was assumed to not be reduced by the GAC filter. Therefore, the activity calculated as collected in the GAC filter is also available for collection on the downstream components. This ensures that shielding requirements and radiation zone designations of the downstream components are also conservatively established. The NRC staff 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.

The NRC staff reviewed the radionuclide concentrations listed in purification media, such as filters, resins, and charcoal beds, to ensure that the radioactive material content listed was based on the stated DFs and the flow rates through the components. As previously noted, some of the components are shared by as many as 12 NPMs, so the NRC staff adapted the review process to consider that only one NPM at a time would be operating at the TS coolant specification limit. The NRC staff considered information provided by the applicant to assess the concentrations in the other waste streams that are expected to contribute to the radioactive material content of the purification media. The NRC staff determined the methods, models, and assumptions used to determine the radionuclide content of the purification media were appropriate.

As part of the review to see how the applicant complied with 10 CFR 52.47(a)(5), the NRC staff examined how the applicant addressed the generation, distribution, and collection of activated corrosion products. 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 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 DCA Part 2, Tier 2, the NuScale Power Plant could also be classified as having a high-duty core.

In a response dated August 8, 2018 (ADAMS Accession No. ML18220B407), the applicant stated that it developed the crud burst model (i.e., crud burst factors) 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. In DCA Part 2, Tier 2, Section 12.2.1.8, "Reactor Pool Water," NuScale stated that the post-crud burst cleanup of the primary coolant in the NPM by the CVCS will operate until the projected dose rate (after NPM disassembly) to an operator on the refueling bridge is less than 6.5×10^{-7} coulombs per kilogram per hour (2.5 mR/hr). During an audit (see ADAMS Accession No. ML18348A966), the NRC staff reviewed the methods, models, and assumptions used by the applicant to determine the dose rate to the operator on the refueling bridge. The review conducted by the NRC staff confirmed that the crud burst factors assumed by the applicant were consistent with the use of industry operating experience, as required by 10 CFR 52.47(a)(22), and are, therefore, acceptable.

With respect to the accumulation of radioactive material in demineralizers, the applicant did not consider the accumulation of crud burst-related activity in DCA Part 2, Tier 2, Section 12.2, and stated that ANSI/ANS Std. 18.1-1999 coolant activity concentrations include crud bursts. Consistent with this position, DCA Part 2, Tier 2, Table 12.2-7, "Chemical and Volume Control System Component Source Terms—Radionuclide Content," and Table 12.2-8, "Chemical and Volume Control System Component Source Terms—Source Strengths," only reflect the radioactive material content of the SSCs resulting from normal operation and not the accumulation of radioactive material resulting from a crud burst.

On October 25, 2018, the NRC staff held a public meeting with the applicant (ADAMS Accession No. ML18327A104), during which the NRC staff clarified that ANSI/ANS Std. 18.1-1999 does not include activity accumulation as a result of crud burst clean up. Following further discussions with the NRC staff, the applicant added the following statements to DCA Part 2, Tier 2, Section 12.2.1.3, "Chemical and Volume Control System":

> The mixed-bed source terms and source strengths are listed in Table 12.2-7 and Table 12.2-8, respectively. These source terms and the associated analyses do not include short-term transients such as CRUD bursts associated with refueling outages. Based on an assumed Co-58 peaking factor of 10,000 and an assumed peaking factor of 1,000 for other CRUD isotopes, it is estimated that a CRUD burst could add up to 450 curies of CRUD isotopes to the CVCS mixed bed demineralizer. This results in the estimates of activity within some plant SSCs to not reflect the CRUD burst related activity, including the CVCS mixed bed demineralizer values (both columns) in Table 12.2-7 and Table 12.2-8.

While DCA Part 2, Tier 2, Table 12.2-7 and Table 12.2-8, do not contain crud burst-related activity, DCA Part 2, Tier 2, Section 12.2.1.3, does describe the potential accumulation of

crud-related activity in demineralizers from crud burst cleanup. Therefore, the NRC staff concludes that the applicant has adequately identified the kinds and quantities of radioactive material expected to be produced during operation, including refueling outages, and collected in the CVCS demineralizers, consistent with the requirements of 10 CFR 52.47(a)(5). Further, the staff finds that the method used to establish crud burst factors employed in refueling dose calculations is consistent with the requirement of 10 CFR 52.47(a)(22) to demonstrate how operating experience insights have been incorporated into the plant design.

The NRC staff reviewed assumptions related to calculating the amount of radioactive material that could be present in resin transfer lines, as discussed in DCA Part 2, Tier 2. The applicant's assumptions for the density of the resin in the resin transfer lines is consistent with operating experience at commercial nuclear power plants (see Report No. DE2004826015, "Studsvik Processing Facility Update," issued 2003, available at https://ntrl.ntis.gov/NTRL/). During normal resin transfer operations, a relatively high ratio of water to resin is required to ensure unobstructed flow of resin slurries through the resin transfer pipes, and fluctuations of water flow rates have resulted in resin transfer lines clogging. The density of the resin in these lines then becomes that of the originating source (e.g., SRSTs, demineralizer bed). The density of the material in the resin transfer line assumed by the applicant is consistent with this operating experience. However, as noted in the previous discussion about the crud burst content of resins, DCA Part 2, Tier 2, Table 12.2-7 and Table 12.2-8, are only representative of the source term accumulated in the listed components from normal operation without a crud burst. For resins that have collected radioactive material from a crud burst, the conditions outlined in DCA Part 2, Tier 2, Section 12.2.1.3, need to be considered by the facility operator when performing resin transfer operations. Since the radioactive material content of the resin may be higher than that assumed during the design of the shielding around the transfer lines, the NRC staff determined that the radiation protection program may need to consider additional actions for the protection of equipment or personnel should the resin transfer occur before the radioactive material has decayed to values commensurate with that listed in DCA Part 2, Table 12.2-7 and Table 12.2-8. However, implementation and specification of items for inclusion in a radiation protection program are beyond the scope of review conducted for a DCA and will have to be addressed at the COL stage.

The NRC staff compared information in DCA Part 2, Tier 2, Section 12.2, regarding the volume of contained sources of radioactive material to other sources of information provided by the applicant. During an audit (see ADAMS Accession No. ML18348A966), the NRC staff noted a difference in the size of the resin transfer line as stated in DCA Part 2, Tier 2, Section 12.2, and the dose calculation packages reviewed during the audit. Since the diameter of the resin transfer line is directly proportional to the amount of radioactive material contained within the line, and the amount of radioactive material in the line drives the shielding requirements in the RXB and the subsequent RXB radiation zone designations, the NRC staff requested clarification regarding the apparent difference. In a response dated April 16, 2019 (ADAMS Accession No. ML19106A454), the applicant explained that the transition to the larger diameter pipe occurred in the RWB and not the RXB. Based on the information provided by the applicant, the NRC staff determined that the radiation environment near the resin transfer line in the RXB described in DCA Part 2, Tier 2, Section 12.3, is consistent with the information provided in DCA Part 2, Tier 2, Section 12.2, and is, therefore, acceptable.

The source terms described in DCA Part 2, Tier 2, Section 12.2, are used as the basis for determining the radiation dose rates used for demonstrating compliance with the requirements of GDC 4 and 10 CFR 50.49(e)(4). The NRC staff reviewed information provided in DCA Part 2, Tier 2, to determine how the applicant complied with the requirements of 10 CFR 50.49(e)(4)

and GDC 4. The NRC staff reviewed information contained in DCA Part 2, Tier 2, Section 12.2, to verify that it is consistent with DCA Part 2, Tier 2, total integrated dose (TID) for SSCs listed in DCA Part 2, Tier 2, Appendix 3C, Table 3C-1, “Environmental Qualification Zones—Reactor Building,” Table 3C-6, “Normal Operating Environmental Conditions,” and Table 3C-8, “Limiting Design Basis Accident EQ Radiation Dose.” These tables describe the integrated dose in and around the NuScale module containment vessel (CNV) caused by normal operations and radiation exposure following an accident. DCA Part 2, Tier 2, Section 12.2, contains tables that list the radionuclide concentrations and the resultant photon spectra and emission rates. In TR 0915 17565-NP-A, “Accident Source Term Methodology,” Revision 4, the applicant described how the source term used for the environmental qualification (EQ) dose analysis is derived. The EQ source term adopted by NuScale differs from past practice, in that it does not include a core-damage event as a contributor to the source term used for EQ. DCA Part 2, Tier 2, Section 12.2.1.13, “Post-Accident Sources,” states that the iodine spike design-basis source term maximum post-accident activity concentrations used for equipment qualification evaluation are provided in DCA Part 2, Tier 2, Section 12.2, tables; however, the specific concentration values in these tables are in excess of the values that would be calculated using the methodology in TR-0915-17565. Since the conditions and limitations associated with the safety evaluation for TR-0915-17565 address limiting the application of the design-basis analysis source term to inside the CNV and under the bioshield, the NRC staff determined that the information contained within DCA Part 2, Tier 2, Section 12.2, was acceptable. The NRC staff determined that questions regarding airborne and liquid activity source terms following an accident that were generated prior to the receipt of TR-0915-17565 are no longer relevant to the review of DCA Part 2, Tier 2, Section 12.2, as discussed in this SER. Further information regarding the NRC staff evaluation of these source terms is contained in Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment,” of this SER, and in the SER for TR-0915-17565.

The NRC staff reviewed information in DCA Part 2, Tier 2, Section 12.2, regarding the generation and transport of nitrogen-16 (N-16) through the RCS. Large quantities of N-16 are generated as reactor coolant passes through the neutron field present in the core during normal operation. The half-life of N-16 is about 7 seconds, so the transit time around the RCS flow loop has a significant impact on the amount of N-16 present at different points within the RCS. N-16 decays through the emission of a gamma photon with an energy of approximately 1.1×10^{-18} joules (7 MeV). Due to their abundance and their energy, these photons can be a significant contributor to the TID of equipment. DCA Part 2, Tier 2, Revision 3, Appendix 3C, Table 3C-6, provides the 60-year integrated gamma dose from normal operation. DCA Part 2, Tier 2, Table 12.2-3, “Cold Leg Primary Coolant Gamma Source Term,” and Table 12.2-4, “Hot Leg Primary Coolant Gamma Source Term,” provide the gamma spectra and fluence from the reactor coolant at two locations that are briefly described in DCA Part 2, Tier 2, Section 12.2.1.2, “Reactor Coolant System.” Using information reviewed during an audit (see ADAMS Accession No. ML18348A966), the NRC staff used the ORIGEN code to assess the concentration of radionuclides at various points within the RCS and the associated photon spectra and fluence. Using ORIGEN calculations, the NRC staff determined that there was a strong correlation between the RCS flow rate and the amount of N-16 decay that occurred. The faster the RCS flow rate, the higher the N-16 concentration at the top of the steam generator (SG) riser plate, and the higher the resultant photon dose rate at the top of the CNV. The applicant provided DCA Part 2, Tier 2, Appendix 3C, Table 3C-6, to reflect the current photon TID. The NRC staff determined that, based on the relative contribution of the gamma photons to the TID of equipment at the top of the CNV and under the bioshield, the basis of the photon source term for these SSCs was adequate and, therefore, acceptable.

Using information reviewed during an audit (see ADAMS Accession No. ML19203A043), the NRC staff was able to perform calculations using ORIGEN, SCALE, and MCNP and determined that the contribution of photons from the neutron activation of CNV and reactor pressure vessel (RPV) structural materials were only a minor contributor to the TID to the equipment at the top of the RPV and CNV and under the bioshield. Based on this confirmatory analysis, the NRC staff determined that the gamma radiation environments in the upper RPV, the upper CNV, and under the bioshield have been adequately described by the applicant in DCA Part 2, Tier 2, Appendix 3C.

The NRC staff reviewed the information provided in DCA Part 2, Tier 2, Appendix 3C, Table 3C-6, about the 60-year integrated N (neutron) dose (rad) for the area outside of the CNV and under the bioshield. 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 1, describes the neutron fluence to the RPV and CNV near the core but does not provide any neutron fluence information above the RPV flange area. Based on the lack of neutron spectral and fluence information in DCA Part 2, Tier 2, the NRC staff was not able to determine, with a reasonable degree of assurance, that the neutron dose rates provided by the applicant for the areas at the top of the RPV, top of the CNV, and under the bioshield were appropriate. Using information reviewed during an audit (see ADAMS Accession No. ML19203A043), the NRC staff identified a NuScale calculation package that provided neutron spectral and fluence information that addressed the NRC staff's concerns. In a proprietary response dated July 22, 2019 (ADAMS Accession No. ML19203A322), the applicant provided the neutron spectral and fluence data requested by the NRC staff. The neutron information provided by the applicant was generally consistent with neutron spectral and fluence data generated by the NRC staff performing calculations using ORIGEN, SCALE, and MCNP6.2. Since the information provided by the applicant and the calculations performed by the NRC staff were generally consistent with the neutron TID provided in DCA Part 2, Tier 2, Appendix 3C, Table 3C-6, the NRC staff concludes that the neutron radiation environments in the upper RPV, the upper CNV, and under the bioshield have been adequately described by the applicant.

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

The NRC staff reviewed the radioactive material content of the UHS pool provided in DCA Part 2, Tier 2, Section 12.2. Because of the depth of the UHS pool relative to the existing light-water reactor fleet, the NRC staff determined that the radionuclide concentration in the UHS pool water is the dominant source of radiation exposure during refueling outages. DCA Part 2, Tier 2, Section 12.2.3, "References," includes a reference to EPRI TR-3002000505. TR-3002000505, Volume 2, covers startup and shutdown chemistry. These documents state that deposition of particulates released during the shutdown evolution can lead to increased shutdown dose rates, elevated smearable activity levels in low-flow regions, and increases in personnel contamination risks. The documents further note that without operating reactor coolant pumps, the flow forces will be reduced, which could result in increased deposition of suspended material, less solubilization of system deposits, and an increased rate of deposition in low flow rate areas. 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 Chapter 12, Phase I audit (see ADAMS Accession No. ML18124A182), the NRC staff was able 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. DCA Part 2, Tier 2, Section 12.2.1.8, specifies that the post-crud burst cleanup of the primary coolant in the NPM by the CVCS will operate until the projected dose rate (after disassembly of the NPM) to an operator on the refueling bridge is less than 0.025 millisieverts per hour (mSv/h) (2.5 millirem per hour (mrem/h)). The proposed criteria of less than 0.025 mSv/h (2.5 mrem/h) is consistent with the criteria in ANSI/ANS Std. 57.2-1983 that the NRC staff uses. As such, the NRC staff determined that a licensee operating a NuScale Power Plant will have to operate the CVCS system during each outage until the projected dose rate to workers on the bridge is less than 0.025 mSv/h (2.5 mrem/h), consistent with ANSI/ANS Std. 57.2-1983. Based on the above, the NRC staff finds the design and proposed operation to be ALARA and, therefore, acceptable.

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, which are used for the design of personnel protective measures and for dose assessment. The NRC staff's review verified that the applicant has provided a tabulation of the calculated concentrations of radioactive material, by nuclides, expected during normal operation. The applicant provided tables in DCA Part 2, Tier 2, Section 12.2, that, when used with information provided in DCA Part 2, Tier 2, Section 11.1 and Chapter 15, facilitate the calculation of potential airborne activity concentrations in the RXB following an accident.

The NRC staff review indicates that the locations of most of the major potential sources of airborne radioactive material, such as filters, operating pumps, high-pressure fluid systems, or systems directly connected to the RCS (e.g., CVCS), were contained within the RXB. Although the NuScale design minimizes the potential for leakage of radioactive fluids, the applicant assumed that there was leakage in the CVCS pump/valve rooms on the 10.9-meter (35-foot, 8-inch) elevation of the RXB and in the degasifier rooms on the 7.3-meter (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.

Transuranic (TRU) nuclides, such as americium, plutonium, and curium, are formed in irradiated uranium fuel by neutron activation and neutron-induced fission and decay predominantly by alpha emission. These radionuclides are often radiologically 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 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). The NRC staff notes that the tabulations of airborne activity concentrations mentioned above did not contain TRU isotopes.

The NRC staff discussed the applicant's methods for considering TRU airborne activity during an audit (see ADAMS Accession No. ML18124A182). The NRC staff noted the potential for the generation of TRU particulate airborne activity from activities such as drying of the RXB NPM dry dock walls, drying of the exterior portions of the upper CNV while it is in the dry dock area, and from operation and maintenance of the drum dryer in the RWB. The NRC staff disagreed with the applicant's responses to questions dated February 19, 2018 (ADAMS Accession No. ML18050A029), and May 9, 2018 (ADAMS Accession No. ML18129A415), regarding airborne TRU activity in the RXB and RWB. Therefore, the NRC staff decided to perform an independent assessment of the potential airborne TRU concentrations that could be present in the RXB and RWB. The NRC staff assessment considered available operating experience and was based on the applicant's plant-specific design information and information about escape coefficients in other NRC guidance documents. Specifically, 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. ML18086A196), to extrapolate potential airborne equilibrium values for major TRU wastes. The independent analysis performed by the NRC staff found that the TRU wastes did not appear to be significant contributors to airborne radioactivity for the NuScale-specific design. Based on the above analysis, the NRC staff determined that the potential airborne concentration of TRU isotopes will not be significant, given the specific characteristics of the NuScale design, and is, therefore, acceptable.

The NRC staff reviewed the sources of tritium that contribute to the equilibrium concentration of tritium in the UHS pool. One of the potential sources of tritium in the UHS pool is direct activation of water by neutrons escaping the CNV. Other major sources of tritium in the UHS pool water include neutron absorption by lithium, which is used for pH control, and boron, which is used for reactivity control. Through independent analysis performed using physical plant parameters available in DCA Part 2, Tier 2, the NRC staff compared the activation rate of hydrogen into deuterium and then into tritium, as well the activation of the naturally occurring deuterium into tritium. The NRC staff analysis compared the amount of naturally occurring deuterium to the amount of deuterium produced through neutron activation of water and determined that the total increase in tritium production from activation of water was negligible. The NRC staff determined that the tritium production rates from neutron absorption in lithium and boron were consistent with those stated by the applicant. The NRC staff finds that DCA Part 2, Tier 2, contains sufficient information about the concentration of tritium in the UHS pool water resulting from the activation of the pool water by fission neutrons, and is, therefore, acceptable.

The NRC staff reviewed how the applicant considered sources of airborne radioactive material within the RWB. DCA Part 2, Tier 2, Revision 3, Section 12.2.2, "Airborne Radioactive Material Sources," 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. Tanks and vessels located within the RWB are vented to the building ventilation system. The NRC staff found that the stated assumptions provided reasonable assurance that airborne activity concentrations within the RWB, from those sources, would be maintained within the limits of 10 CFR Part 20. As noted above, the NRC staff discussed with the applicant the drum dryer, which is located in the RWB, as a specific potential source of airborne radioactive material. The NRC staff reviewed the description of the drum dryer in DCA Part 2, Tier 2, Section 11.2.2.1.4, "Chemical Waste," and through discussions held with the applicant, the NRC

staff determined that the applicant provided sufficient information about the drum dryer as a specific potential source of radioactive material. The NRC staff determined that the applicant adequately described the potential sources of airborne radioactive material within the RWB, and, therefore, the staff finds the description of the sources of airborne radioactive material in the RWB acceptable.

The NRC staff reviewed concentrations of particulate (other than from TRU wastes), iodine, and noble gas airborne radioactive material within the RXB, as discussed in DCA Part 2, Tier 2, Section 12.2. The NRC staff review included comparing the NRC staff's estimates of airborne particulate activity to those contained in DCA Part 2, Tier 2, Section 12.2. The NRC staff review showed that based on the concentrations of radionuclides in the UHS pool water and the resuspension rates of particulates due to the evaporation of the UHS pool water, the principle contributor to dose from airborne activity within the RXB was airborne tritium. The regulations at 10 CFR 20.1003 use the following two criteria to define an airborne activity area: (1) the total derived air concentration exceeding the values of Appendix B to 10 CFR Part 20 or (2) the presence of airborne activity concentrations in the area such that an individual without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent of the ALI or 12 DAC-hours. As such, the NRC staff finds that the particulate activity concentration contained within the RXB atmosphere is unlikely to result in the area being classified as an airborne activity area, in accordance with 10 CFR 20.1003. The concentration of airborne particulate material is also unlikely to be a significant contributor to the ORE of workers in the RXB. Therefore, the NRC staff finds the probable concentrations of particulate, iodine, and noble gas airborne radioactive material within the RXB to be acceptable.

DSRS Section 12.2 states, in part, that for nuclear power plants designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations should be based on a primary coolant concentration of 130 kilobecquerels per gram (kBq/g) (3.5 microcuries per gram (μCi/g)) or an alternate value for which the methods, models, and assumptions have been provided in the application. This value is an important consideration for evaluating the concentration of airborne tritium, as described above. The NRC staff used a combination of the information contained in DCA Part 2, Tier 2, Section 12.2; results of calculations performed by the NRC staff; and information reviewed during audits (see ADAMS Accession Nos. ML18124A182, ML18348A966, and ML19203A043) to assess how the applicant determined the RCS tritium concentration. The NRC staff determined that the method used by the applicant to calculate RCS tritium concentration included the use of tritiated water for all RCS makeup. DCA Part 2, Tier 2, Table 11.1-8, contains the primary coolant average concentration of tritium, and the applicant provided a footnote to the table noting the maximum calculated peak primary coolant tritium activity of 130 kBq/g (3.5 μCi/g). Because the NRC staff's review of the applicant's calculations verified that the RCS tritium activity was at this peak value for a short period of time, and because the value used by the applicant for RCS tritium calculations was reasonable, the method proposed to calculate tritium concentrations is acceptable to the staff.

The NRC staff evaluated the amount of airborne tritium that could be present in the RXB atmosphere, and the methods, models, and assumptions used by the applicant to determine the concentration of airborne tritium. The NRC staff and the applicant recognized that the amount of airborne tritium was related to how much tritium was added to the UHS pool water, how much tritium evaporated from the pool, and finally how much tritium was removed by the RXB ventilation system (RBVS). The amount of tritium produced in the coolant is dependent on RCS chemistry (e.g., how much boron is in the coolant) and the reactor physics (e.g., reactor power level, fuel configuration). Since the processes used to clean waste water do not remove tritium,

the concentration of tritium in any water removed from the RCS and recycled back to the RCS will remain unchanged. The RCS coolant tritium concentration is dependent on how much tritium is produced in the coolant, the amount of tritium removed via letdown, and the amount of tritium added as a result of recycling-processed RCS wastewater, as makeup feed water for the CVCS, during boron dilution. This is the amount of tritium that will be added to the pool water when the CNV is disassembled for refueling. Using recycled RCS water as makeup water for the UHS pool will increase the concentration of tritium in the UHS pool. Tritium is removed from the UHS pool primarily through evaporation. The amount of evaporation from the UHS pool is dependent on the air flow rate across the surface of the pool water, the surface area of the pool water, and the temperature difference between the pool water and the building air temperature.

Based on information made available to the NRC staff during the Chapter 12, Phase I, audit (see ADAMS Accession No. ML18124A182), the NRC staff determined that the applicant evaluated the evaporation rate of the reactor pool water at 37.8 degrees Celsius (°C) (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. In response to a question dated January 28, 2019 (ADAMS Accession No. ML19028A413), the applicant reduced the TS allowed pool water temperature to 43.3 °C (110 °F) (see Chapter 6 of this SER for the staff evaluation of the response to this question). The NRC staff used interactions with the applicant during an audit (see ADAMS Accession No. ML19203A043) and additional information provided in the applicant's response dated December 17, 2018 (ADAMS Accession No. ML18351A390), to examine the effects of changes to the exposed UHS pool surface area available for evaporation into the RXB pool area room volume and changes to analysis assumptions for tritium production rates in the RCS. The NRC staff used this additional information to determine that even if the UHS pool water temperature were to rise to 43.3 °C (110 °F), the concentrations of airborne tritium in the RXB pool area would be less than the values required to designate the area as an airborne radioactivity area, as defined in 10 CFR 20.1003. Therefore, the NRC staff has reasonable assurance that the airborne tritium concentrations will be maintained in accordance with the requirements of 10 CFR Part 20, Subpart C, "Occupational Dose Limits," 10 CFR Part 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas," and 10 CFR 20.1101(b).

12.2.5 Combined License Information Items

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

Table 12.2-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 12.2

COL Item No.	Description	DCA Part 2, 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 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-NP-A, “Accident Source Term Methodology,” Revision 4 provides the source terms. Further information regarding the NRC staff evaluation of these source terms is contained in Section 3.11 of this SER and in the SER for TR-0915-17565.

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 pool and leakage from system components located in the equipment compartments. The applicant has tabulated the maximum expected routine radioactive airborne concentrations for areas where airborne radioactive material may be present, such as in the CVCS pump/valve room, the degasifier room, and the airspace above the reactor pool.

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. In turn, the NRC staff deemphasized its review of sections of the application for which aspects of the NuScale design were less radiologically significant than the currently licensed fleet. Areas of increased focus for the staff review included the custom failed fuel fraction for the basis of the design of the shielding and ventilation systems; the 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 areas of reduced NRC staff focus included the amount of water shielding above irradiated fuel bundles and the 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 GDC 4, 19, and 61, as they relate to the information on radiation sources provided by the applicant; and 10 CFR 20.1406(a) and 10 CFR 52.47(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. The NRC staff determined that the applicant provided adequate data about the kinds and quantities of radioactive material expected to be present in the plant, and that the applicant adequately described the methods, models, and assumptions used to determine the quantities of

radioactive material expected to be present, in accordance with the regulatory requirements listed above.

12.3 Radiation Protection Design Features

This section covers both Section 12.3 and Section 12.4 of DCA Part 2, 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 inservice inspection, provide a measure of the effectiveness of the proposed design features in reducing overall area dose rates.

12.3.2 Summary of Application

DCA Part 2, Tier 1: The DCA Part 2, Tier 1, information associated with this section includes DCA Part 2, 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”; and Section 3.14, “Equipment Qualification—Shared Equipment.” This section includes descriptions of design features that demonstrate compliance with the occupational and public radiation safety requirements of 10 CFR Part 20; the requirements to protect SSCs from the effects of the potential radiation environment that may be present during normal operation and potential accidents; the monitoring of radiation levels in the plant following potential accidents; and monitoring of radiation levels in areas where fuel is stored or handled, including those 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.

DCA Part 2, Tier 2: The applicant described radiation protection design features in DCA Part 2, Tier 2, Sections 12.3 and 12.4, which are summarized, in part, as follows:

- The RXB, which provides shielding for the protection of MCR operators and members of the public, contains 12 individual reactor modules in a common pool of water, which also provides shielding during normal operation, refueling, and accident conditions.

- The RPV contains an integral pressurizer and SGs. RCS fluid is circulated through the core and SGs through natural convection. The RPV is located inside of a steel CNV that is evacuated to near 0 pascals (absolute) (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.

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

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

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

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

- Radiation monitoring equipment is provided for monitoring areas where fuel is stored or handled (including the movement of NPMs during refueling), normal operation, and post-accident conditions.

ITAAC:

DCA Part 2, Tier 1, Sections 2.7, 2.8, 3.3, 3.9, 3.11, 3.12, 3.14, 3.17, and 3.18 provide the ITAAC associated with DCA Part 2, Tier 2, Section 12.3, and are discussed in the following SER sections as indicated:

- Section 2.7—SER Section 14.3.8, “Radiation Protection—Inspections, Tests, Analyses, and Acceptance Criteria”

- Section 2.8—SER Section 14.3.6, “Electrical Systems—Inspections, Tests, Analyses, and Acceptance Criteria”

- Section 3.3—SER Section 14.3.8

- Section 3.4—SER Section 14.3.7, “Plant Systems—Inspections, Tests, Analyses, and Acceptance Criteria”

- Section 3.9—SER Section 14.3.8

- Section 3.11—SER Section 14.3.8

- Section 3.12—SER Section 14.3.8

- Section 3.14—SER Section 14.3

- Section 3.17, “Radiation Monitoring—NuScale Power Modules 1–6”—SER Section 14.3.8

- Section 3.18, “Radiation Monitoring—NuScale Power Modules 7–12”—SER Section 14.3.8

Technical Specifications: DCA Part 2, Tier 2, Chapter 16, “Technical Specifications,” Section 5.7, “High Radiation Area,” addresses TS for the control of high-radiation areas.

Technical Reports:

- NuScale TR-0116-20781-P

Topical Reports:

- NuScale TR-0915-17565

12.3.3 Regulatory Basis

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

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

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

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

- 10 CFR 20.1301 and 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public,” as they relate to the facility design features that affect the radiation exposure to a member of the public from noneffluent sources associated with normal operations and AOOs

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

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

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

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

- 10 CFR 50.68, "Criticality Accident Requirements," or 10 CFR 70.24, "Criticality Accident Requirements," as they relate to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled

- GDC 14, "Reactor Coolant Pressure Boundary," and GDC 30, "Quality of Reactor Coolant Pressure Boundary," as they relate to the ability to detect RCS pressure boundary leakage with radiation detectors

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

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

- GDC 63, "Monitoring Fuel and Waste Storage," as it relates to detecting excessive radiation levels in the facility

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

- 10 CFR 52.47(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 DCA Part 2 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 DSRS 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, 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, Office of Nuclear Reactor Regulation, to Regional Administrators, dated August 16, 1982, as they relate to methods acceptable to the NRC staff for complying with NRC regulations to provide and calibrate, or verify the calibration of, safety-related instrumentation for radiation monitoring following an accident in a nuclear power plant

- RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," as it relates to actions taken to address the guidance in RG 8.8, Regulatory Position C.2(d), during facility design, engineering, construction, and decommissioning to maintain 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 DCA Part 2, Tier 2, Chapter 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 DCA Part 2, Tier 2, 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 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 HPSSC-6.8.1-1981, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors,” as it relates to criteria for the establishment of locations for fixed continuous area gamma-radiation monitors and for design features and ranges of measurement

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

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

- Memorandum from L.W. Camper to D.B. Matthews and E.E. Collins, “List of Decommissioning Lessons Learned in Support of the Development of a Standard Review Plan for New Reactor Licensing,” dated October 10, 2006 (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 DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Sections 12.3 and 12.4, and in other related sections of DCA Part 2, Tier 1 and 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 DCA Part 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, "Domestic Licensing of Special Nuclear Material," have been met. The sections below present the NRC staff's findings.

Several aspects of the radiation protection design features of the NuScale SMR design differ from those traditionally found in PWRs. The RXB contains up to 12 reactors that are each enclosed in a separate containment vessel, which are submerged in a common pool of water that is used as the UHS. The UHS, instead of concrete, provides the primary shielding (i.e., the shielding immediately around the reactor vessel). During refueling, the containment vessel, including the contained reactor vessel and all the fuel for that reactor, is moved as an integral unit to the attached refueling pool. In the refueling portion of the pool, the containment and reactor vessels are disassembled, and fuel is removed and placed in the attached SFP. The three connected pools (the reactor cooling pool, the refueling pool, and the SFP) are interconnected and enclosed by a single RXB. While one reactor is being refueled, up to 11 other reactors, located in the contiguous reactor cooling pool, may continue to operate. The SGs utilize helical coil tubes, with the secondary coolant on the inside of the tubes and reactor coolant on the outside of the tubes. Control rods, and the associated drive mechanisms, are all fully contained within the reactor vessel. The decay heat removal heat exchanger, which is mounted on the exterior of the containment vessel, is normally submerged in the joint reactor shielding/refueling pool and is subject to drying during refueling. Spent resin and liquid waste storage tanks and related processing SSCs are in a separate building located adjacent to the RXB.

12.3.4.1 Radiation Protection Design Features

The facility 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), and to maintain radiation exposures for workers and members of the public to within the limits of 10 CFR Part 20. The facility design includes provisions for minimizing contamination of the facility and the environment, minimizing the amount of waste generated, and minimizing the cost of decommissioning. Design features are provided to monitor radiation fields within the facility for the protection of workers, the assessment of potential accident conditions, and the radiation fields where fuel is stored or handled. These design features include facility design, shielding, ventilation, and area and airborne radiation monitors.

12.3.4.1.1 Facility Design Features

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

The NRC found, in its review of DCA Part 2, Tier 2, Section 12.3.1, “Facility Design Features,” that the NuScale design incorporates many features of large light-water reactors that have been shown to be effective at reducing radiation exposures to workers, consistent with ALARA, minimizing contamination of the facility, minimizing the generation of waste, and facilitating decommissioning.

Based on information provided in DCA Part 2, Tier 2, Section 12.3.1, the types of materials used in the construction of the facility are specified to reduce corrosion rates and improve equipment reliability. Stainless steel is used for reactor coolant pressure boundary applications. Thermally treated Alloy 690 base metal is used for SG tubing material. Stainless steels and thermally treated Alloy 690 metal are used to reduce the possibility of intergranular stress-corrosion cracking, which could reduce equipment failure rates and, therefore, reduce worker dose resulting from maintenance activities. Stainless steel is used for the CNV, which is in continuous contact with the UHS pool water. In DCA Part 2, Tier 2, Section 12.3.1, the applicant stated that the amount of cobalt impurities allowed in SSCs wetted by the RCS or exposed to neutron fluxes will be limited to reduce activated corrosion products in plant systems. The NRC staff finds that the design appropriately limits cobalt, consistent with the guidance in RG 8.8 and RG 4.21, thereby meeting the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406.

The NRC staff looked at the expected radiation environment of components, particularly those that are not safety related, located within the CNV. The NRC staff was interested in these components because they would not be within the EQ program (10 CFR 50.49) and may not be considered within the scope of GDC 4. The NRC staff’s concern was that radiation-induced degradation of these components might result in a failure mechanism during an accident, resulting in unexpected debris generation and reduced core cooling during an accident. Some examples of the types of components the NRC staff considered include the reactor component cooling water (RCCW) flexible hoses to the control rod drive mechanism (CRDM) motors and magnets, the RCCW thermal relief valves, and power and signal cables for the individual rod position indicator system coils. This type of information was made available to the NRC staff during a Chapter 12, Phase I, audit (see ADAMS Accession No. ML18124A182), and in a CRDM audit (see ADAMS Accession No. ML17331A357). As part of these audits, the NRC staff looked at the radiation environment inside of the CNV and the location of components that are not safety related. As a result of discussions with the staff, the applicant revised DCA Part 2, Tier 2, Section 3.11.6, to state that there would be no environmentally induced debris inside the CNV that would interfere with the proper functioning of the emergency core cooling system. The NRC staff evaluated the information provided by the applicant and finds it to be acceptable because the applicant revised the DCA to state that there will not be equipment that is not safety related within the CNV that could interfere with the operation of the emergency core cooling.

Due to the radiological significance and high specific activity of used resins, the NRC staff reviewed submitted information about used resin-handling SSCs, such as sealed pumps, valves, and the provisions (air and water supplies, piping arrangement) for sparging tanks containing used resin. The NRC staff review found that the applicant implemented design features that should reduce ORE through the elimination of concentrations of radioactive material and by reducing the frequency and duration of corrective maintenance. The provision of design features, such as sealed pumps, also reduces the potential for leaks, which could cause contamination of the facility or the generation of waste. The design of the resin transfer system allows for the reuse of water provided for fluidizing (i.e., fluffing) the resins in tanks or demineralizers and for transferring the resins between vessels. This reduces the amount of

water required and, thus, reduces the amount of radioactive waste generated. Since the design is consistent with the requirement of 10 CFR 52.47(a)(22) to demonstrate how operating experience insights have been incorporated into the plant design, the requirements of 10 CFR 20.1406(a) to provide design features for minimizing the amount of waste generated, and the requirements of 10 CFR 20.1101(b) to provide design features for maintaining ORE ALARA, the NRC staff finds this approach acceptable.

The NRC staff reviewed the provisions for limiting dose to members of the public from direct sources (i.e., contained sources) of radioactive material. The applicant has a large pool surge controls system (PSCS) tank located outside of the RXB. The PSCS storage tank is designed to temporarily store cleaned up pool water that is displaced during dry dock operations. The NRC staff used information from audits (see ADAMS Accession Nos. ML18124A182, ML18348A966, and ML19203A043) to understand how the applicant determined the amount of radioactive material contained in the PSCS tank and identified what design features are available for controlling and limiting the amount of radioactive material in the tank. The NRC staff confirmed that the design of the facility has provisions for removing radioactive material from the water in the dry dock, from water being pumped to the PSCS tank, and from water in the PSCS tank. The staff finds this acceptable, since the design is consistent with the requirement of 10 CFR 52.47(a)(22) to demonstrate how operating experience insights have been incorporated into the plant design; the requirements of 10 CFR 20.1406(a) to provide design features for minimizing the amount of waste generated; the requirements of 10 CFR Part 20, Subpart D, “Radiation Dose Limits for Individual Members of the Public,” for limiting dose to members of the public; and the requirements of 10 CFR 20.1101(b) to provide design features for maintaining ORE ALARA.

Consistent with the guidance in RG 8.8, the applicant considered the radiation levels, the access frequency, and the duration of access of personnel when establishing radiation zone maps. The applicant based the radiation zones on the maximum dose rate in the area, which is consistent with the guidance in RG 8.8. Based on this approach, the applicant provided normal operation radiation zone maps in DCA Part 2, Tier 2, Section 12.3, for portions of the RXB and RWB. The applicant also provided radiation zones related to EQ (see Section 3.11 of this SER for the NRC staff’s evaluation of EQ) in DCA Part 2, Tier 2, Section 3.11. The applicant provided airborne radioactivity zones for those portions of the facility with the potential for airborne radioactivity. Based on the absence of identified vital area missions (i.e., no expectation for personnel to be in the area) for non-core-damage accidents, the applicant did not provide post-accident radiation zone maps. There are no direct requirements for radiation zone maps in 10 CFR Part 20, 10 CFR Part 50, or 10 CFR Part 52. As such, the NRC staff evaluated the proposed use of radiation zone maps to help keep ORE ALARA and to identify radiation environments and determined that they are consistent with the requirements of 10 CFR 20.1101(b), 10 CFR 50.49(e)(4), and GDC 4 and are, therefore, acceptable.

DCA Part 2, Tier 2 Section 12.3.1.3.1, provides information on controls and design features for VHRAs 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. DCA Part 2, Tier 2, Table 12.3-3, only identifies one VHRA in NuScale DCA Part 2, 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 the facility design should ensure that an individual is not able to gain unauthorized or inadvertent access to areas in which radiation levels could be encountered at 5 Gy (500 rad) or more in 1 hour at 1 meter (3.3 feet) from a radiation source or any surface through which the radiation penetrates (e.g., those adjacent to operating reactors or irradiated portions of reactors or CNVs

of shutdown reactors). The applicant stated that the door to enter the HIC storage room is locked to prevent unauthorized access and that egress from the area is not impeded. Therefore, the NRC staff concluded that the identification of the VRHAs and the description of the design features specified for VHRAs is consistent with the requirements of 10 CFR 42.47(a)(5) for providing controls to maintain radiation exposures within the limits of 10 CFR Part 20. Further, this is consistent with the requirements of 10 CFR Part 20, Subpart G, "Control of Exposure from External Sources in Restricted Areas," and 10 CFR 20.1101(b) to maintain ORE ALARA. Therefore, the staff finds the provided information on controls and design features for VHRAs to be acceptable.

12.3.4.1.2 Shielding

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

Using a risk-informed approach, the NRC staff's evaluation of radiation shielding focused on areas of the facility that could contain high concentrations of radioactive materials during normal operation or following accidents. Using this approach, the NRC staff focused the shielding review on areas like neutron and gamma radiation from the NPMs as a result of operation; gamma radiation emitted from the NPMs following accidents; plant components, such as demineralizers, filters, and charcoal beds, that concentrate radioactive material; irradiated components, such as the SPNDs; large masses of radioactive material, such as the UHS pool; and areas of the facility where high concentrations of radioactive material may accumulate, such as SRSTs, PSTs, and HIC storage areas. For the identified areas above, the NRC staff compared the amount of radioactive material in the SSCs to the shielding provided and the resultant dose rates. The NRC staff used SCALE, ORIGEN, MicroShield, and MCNP to perform confirmatory or scoping calculations, which can be seen in the analysis below.

DCA Part 2, Tier 2, Table 12.3-6, "Reactor Building Shield Wall Geometry," and Table 12.3-7, "Radioactive Waste Building Shield Wall Geometry," provide concrete shielding thicknesses for rooms and cubicles containing significant radiation sources, which require shielding. These shielding thicknesses are based on the source terms provided in DCA Part 2, Tier 2, Chapter 11 and Section 12.2, and core damage source terms consistent with TR-0915-17565, where appropriate. The applicant used SCALE and MCNP to perform source-term and radiation shielding and zoning dose calculations. For most shielding applications in the NuScale design, concrete shielding is designed in accordance with NRC-endorsed ANSI/ANS 6.4-2006, using material descriptions from Pacific Northwest National Laboratory-25870, "Compendium of Material Composition for Radiation Transport Modeling," Revision 1, or the SCALE material descriptions. The NRC staff finds that methods described by the applicant for performing shielding evaluations are consistent with those identified in NRC guidance and are, therefore, acceptable.

DCA Part 2, Tier 2, Revision 3, Section 12.3.2, states that while concrete is the material used for a significant portion of plant shielding, other types of materials, such as steel, water, tungsten, and polymer composites, are considered for both permanent and temporary shielding. DCA Part 2, 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.” DCA Part 2, Tier 2, Section 12.3, Tables 12.3-6 and 12.3-7, identify the type of shielding material provided, the nominal thickness of the concrete of the walls and floors used for shielding in the RXB, the RWB parameters such as boron enrichment, and the locations of radiation shield doors (see DCA Part 2, Tier 2, Table 12.3-8 and Table 12.3-9). The applicant’s specifications for radiation shield doors states that these shield doors are designed to have a radiation attenuation capability that meets or exceeds that of the wall in which they are installed. The NRC staff concluded that the types of radiation material identified were appropriate for the specified areas, and that the use of radiation shield doors with the same or more attenuating capability as the walls in which they are installed is consistent with the requirements of 10 CFR 42.47(a)(5) for providing controls to maintain radiation exposures within the limits of 10 CFR Part 20 and is consistent with the requirement of 10 CFR 20.1101(b) to maintain ORE ALARA. Therefore, the staff finds that the type of radiation material identified is acceptable.

DCA Part 2, Tier 2, Section 12.3, describes the dimensions and locations of shielding. During audits (see ADAMS Accession Nos. ML18124A182 and ML18348A966), the NRC staff identified some areas where the shielding calculations performed by the applicant assumed the use of metal plates that were not described in DCA Part 2, Tier 2, Section 12.3, Tables 12.3-6 and 12.3-7. As a result of the audit, the applicant modified DCA Part 2, Tier 2, Table 12.3-7, to include information about the dimensions of additional shielding plates as assumed in the shielding calculations. As such, the NRC staff determined that the applicant’s description of the use of the metal shield plates to reduce radiation exposure is consistent with the requirements of 10 CFR 42.47(a)(5) for providing controls to maintain radiation exposures within the limits of 10 CFR Part 20 and is consistent with the requirement of 10 CFR 20.1101(b) to maintain ORE ALARA. Therefore, the staff finds the applicant’s description to be acceptable.

During the NRC staff review of the shielding for the HIC storage room, an area identified as containing a sufficient amount of radioactive material to result in dose rates of 5 Gy/h (500 rad/h), with no upper limit specified, the NRC staff found that the applicant’s arrangement of the radioactive material in the room (e.g., a limited number of containers in the center of the room away from the shield walls and the ceiling) did not appear to conservatively represent the arrangement of radioactive material that could be in the room. The NRC staff performed an independent analysis that considered the specific activity of differing resin and filtered waste streams and the expected decay of isotopes during the collection of the large number of HICs capable of fitting in the room. The NRC analysis considered actions that may be taken as part of the operational radiation protection program to arrange the HICs so that the HICs with the highest radioactive material content would be located away from sensitive portions of the room. Although the applicant stated that HIC stacking was not allowed because HIC stacking rings are commercially available and DCA Part 2 does not preclude the use of stacking rings, the NRC staff analysis included the use of HIC stacking to assess the effects on dose rates in the RWB, including adjacent rooms and the truck bay, which is located above the HIC storage room. The results of the NRC staff’s conservative analysis showed that the zoning designations reported by the applicant were reasonable. The NRC staff concluded that its analysis using a room filled with HICs was reasonably close to the applicant’s values by comparison. Therefore, the staff determined that the radiation zoning for the HIC storage room and the surrounding areas was acceptable.

DCA Part 2, Tier 2, Section 12.3.2.2, states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. The acceptance criteria in DSRS Section 12.3–12.4 state that an assessment of design features provided to protect shielding material subject to degradation, such as through the effects of radiation, temperature extremes (e.g., degradation of concrete caused by high temperature), and density changes (e.g., due to drying), should be provided. The guidance in RG 1.69 discusses the use of American Concrete Institute (ACI) 349-06, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary”; ACI 349.1R-07, “Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures”; and the associated environmental constraints on concrete shielding material. The NRC staff confirmed that the applicant has committed to following the guidance of RG 1.69, which endorses ACI 349. Therefore, the NRC staff finds that the specifications for the design of the concrete shielding wall is consistent with the requirements of 10 CFR 52.47(a)(5) to provide controls to maintain radiation exposures within the limits of 10 CFR Part 20 for the life of the plant and, as such, is acceptable.

The NRC staff used information obtained during an audit (see ADAMS Accession No. ML19203A043) to verify that design temperature specifications for the borated HDPE were sufficient to ensure the integrity of the borated neutron radiation shielding capability for the design life of the plant. The applicant stated that since degradation of the borated HDPE radiation shielding material used on the face of the NPM bioshield bay could potentially occur if the exhaust ventilation provided for the reactor module bays does not maintain air temperatures under the bioshield less than 82.2 °C (180 °F) (e.g., due to damper failure), conditions in which the air temperature under the bioshield exceeds 82.2 °C (180 ° F) require an evaluation of the continued efficacy of the bioshield polyethylene material’s radiation shielding properties. Therefore, the NRC staff finds that the specifications for the design of the concrete shielding wall are consistent with the requirements of 10 CFR 52.47(a)(5) to provide controls to maintain radiation exposures within the limits of 10 CFR Part 20 and are consistent with the requirements of 10 CFR 20.1101(b) to maintain ORE ALARA for the life of the plant. Therefore, the staff finds the specifications to be acceptable.

DCA Part 2, Tier 2, Revision 3, Section 12.3.2.4, “Major Component Shielding Design Description,” describes the use of borated HDPE for the bioshield neutron shielding. The NRC staff used information obtain during an audit (see ADAMS Accession No. ML19203A043) to verify that design specifications for the borated HDPE related to boron content were appropriate to ensure adequate neutron radiation shielding capability for the design life of the plant, and that provisions were made to allow the venting of gases generated through the absorption of neutrons by boron in the shielding material. The NRC staff verified that the values in DCA Part 2, Tier 2, Table 12.3-6, specifying material thickness and boron content were consistent with the values used in the applicant’s shielding calculation input files and were sufficient to bound the total neutron radiation fluence expected during the design life of the plant. The NRC staff also verified that the shielding panels included vents to allow the release of gases that will be generated by the boron adsorption of neutrons. As such, the NRC staff finds that the material characteristics of the borated HDPE shielding material is compatible with the radiation environment expected during normal operation over the life of the plant, consistent with the requirements of 10 CFR 52.47(a)(5) to provide controls to maintain radiation exposures within the limits of 10 CFR Part 20, consistent with the requirements of GDC 4 to ensure that SSCs are compatible with the environmental conditions, and consistent with the requirements of 10 CFR 20.1101(b) to maintain ORE ALARA for the life of the plant. As such, the staff finds the design specifications for the borated HDPE related to boron content to be acceptable.

The acceptance criteria in DSRS Section 12.3–12.4 state that accessible portions of the facility that are capable of having radiation levels greater than 1 Gy/h (100 rad/h) should be shielded and clearly marked with a sign stating that potentially lethal radiation fields are possible. DCA Part 2, Tier 2, Revision 3, identifies a number of areas (e.g., resin demineralizers, filters, SRSTs) that may contain quantities of radioactive material resulting in radiation dose rates exceeding 1 Gy/h (100 rad/h). The locations of the shield plugs are identified on the radiation zone map figures contained in DCA Part 2, Tier 2, Section 12.3. The applicant also specified that the shield plugs provide an equivalent radiation attenuation to the shield floor that contains the plug. The NRC staff evaluated the information provided and determined that the use of shield plugs that have a shielding value equivalent to the floor in which they are installed provides reasonable assurance that radiation exposure to workers in the areas around the shield plugs will be maintained within the limits of 10 CFR Part 20 and ORE will be ALARA. Therefore, the staff finds this to be acceptable.

In DCA Part 2, Tier 2, Section 12.3.1, the applicant discussed general design features for minimizing radiation streaming through penetrations. The NRC staff review identified the presence of large penetrations, such as the main steam lines (MSLs), main feed water lines (MFWLs), and NPM bay HVAC lines, in the radiation shield wall between the NPM bay and the RXB steam gallery area. The NRC staff noted that DCA Part 2, Tier 2, Chapter 12, Table 12.3-6, does not contain a description of the shielding for these penetrations. However, DCA Part 2, Tier 2, Figure 12.3-1g, "Reactor Building Radiation Zone Map—100' Elevation," depicts radiation zones in the steam gallery that would correspond to the presence of shielding that is not evident in DCA Part 2, Tier 2, Table 12.3-6. In its responses to questions dated May 8, 2018 (ADAMS Accession No. ML18128A390) and May 15, 2019 (ADAMS Accession No. ML19137A287), the applicant stated that the penetrations and penetration shielding design were not finalized and would be completed in a future phase of the design that will be the responsibility of the COL applicant. In a public meeting conducted on July 22, 2019 (ADAMS Accession No. ML19302E460), the NRC staff presented potential options available to the applicant to resolve this issue (ADAMS Accession No. ML19204A277).

Because the NuScale application neither describes nor analyzes the shielding for the penetrations in the shield wall, the NRC staff does not have a basis for making findings on the adequacy of a future shielding design to achieve the necessary safety function for limiting dose to values as specified in the application. The NRC staff determined that the COL item proposed by the applicant was not sufficient for the NRC staff to conclude its review. Furthermore, the applicant stated that it did not intend to revise DCA Part 2, Tier 1, Chapter 4, to include appropriate design interface items pursuant to 10 CFR 52.47(a)(26), nor to propose design acceptance criteria for the penetration shielding, nor to complete the shielding design. Therefore, the NRC staff recommends that the Commission include language in the proposed rule stating that the NRC is not making a finding on the adequacy of the necessary shielding to limit dose in the steam gallery to be consistent with the radiation zones specified in the application. Specifically, Section VI, "Issue Resolution," of Appendix G to 10 CFR Part 52 for the DC for the NuScale SMR will state that the design of the shielding for the penetrations between the NPM module and the RXB steam gallery is not considered resolved within the meaning of 10 CFR 52.63(a)(5) by the DC, and Section IV, "Additional Requirements and Restrictions," of Appendix G will require the COL applicant to complete the design and analysis of the shielding for the penetrations between the NPM bay and the RXB steam gallery.

The NRC staff reviewed the radiation zone designations for areas in the RXB near the UHS pool. The radiation zone designations were based on the NPM as a neutron and gamma source term. Shielding above the NPM is provided by the concrete mass of the bioshield cover.

Shielding for the front of the NPM bay is provided by the borated HDPE that is encapsulated in steel plates. The NRC staff determined that based on the inaccessibility of the area of the RXB corresponding to the UHS pool level near the NPM bay entrances by personnel during normal operation (i.e., the need to use a boat or a personnel basket suspended from a crane), the radiation zone designations for those portions of the RXB provide reasonable assurance that ORE will be controlled by the radiation protection program, will be maintained ALARA, and will be kept within the limits of 10 CFR Part 20. Therefore, the staff finds these radiation zone designations to be acceptable.

DCA Part 2, Tier 2, Section 12.3.2.4.4, “Radioactive Waste Building,” states that additional shielding is provided for the drum dryer. The drum dryer collects the effluent waste streams from other liquid radioactive waste processing system components. The water from these waste streams is routed to a 208-liter (55-gallon) nominally 7.4-cubic foot or 208.2-literdrum, 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. Due to the potential for high specific activity to be present in the drum, the NRC staff used audits (see ADAMS Accession Nos. ML18124A182 and ML18348A966) to evaluate the methods, models, and assumptions used by the applicant to determine the amount of shielding for the drum dryer. The radiation zone maps contained in DCA Part 2, Tier 2, Section 12.3, reflect the radioactive material content of the drums and the shielding provided. The NRC staff reviewed the provided information and concluded that the applicant had appropriately identified the kinds and quantities of radioactive material expected to be present in the drum, and that the shielding provided was appropriate for maintaining worker exposure within the limits of 10 CFR Part 20 and for keeping ORE ALARA. Therefore, the staff found the shielding provided for the drum dryer to be acceptable.

The information (source term, shielding geometries, shield thicknesses) provided by the applicant allowed the NRC staff to perform confirmatory analyses of the applicant’s radiation zone designations. The NRC staff’s analyses concluded that the radiation zoning specified in the figures in DCA Part 2, Tier 2, Section 12.3, is accurate and appropriate with the specified shielding. Except as noted for the penetration shielding, the NRC staff has reasonable assurance that the radiation zone designations will allow workers to support the operation of the facility within the radiation exposure limits of 10 CFR Part 20 and will allow workers to maintain ORE ALARA. The radiation zone designations are also consistent with the information provided in DCA Part 2, Tier 2, Section 3.11, regarding the radiation environments for EQ and GDC 4, with the specified shielding.

12.3.4.1.3 Ventilation

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

In general, for the NuScale design, ventilation pathways in radiologically controlled areas flow from areas anticipated to have lower levels of airborne activity to areas anticipated to contain higher levels of radioactivity. DCA Part 2, 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 documents the NRC staff’s assessment of the airborne activity concentrations.

DCA Part 2, Tier 2, Section 12.3.3, “Ventilation,” describes ventilation system design features provided to minimize ORE. In areas subject to airborne activity, the ventilation systems are designed to collect, process, and exhaust airborne radioactive material, including directing airflow to processed exhausts; the building ventilation systems are designed to maintain an airflow inside the buildings from areas of low airborne potential to areas of higher airborne potential; ducting interior and exterior surfaces have relatively smooth finishes to reduce the localized collection of radioactive contamination; differential pressures across filter media are limited to ensure adequate filtering and to prevent damage; and duct sizes and flowrates are designed to prevent settling of radioactive material in the ducts. However, the NRC staff’s review of DCA Part 2, Tier 2, Section 9.4, “Air Conditioning, Heating, Cooling, and Ventilation Systems,” showed that there was little or no additional information on the physical parameters of the HVAC systems in the RXB or RWB that support these statements. In addition, there was no discussion on the design features (e.g., flow balancing dampers) provided to support establishing the flowrates required to sweep air from areas containing lower contamination to areas containing higher contamination. In the response to questions dated May 23, 2018 (ADAMS Accession No. ML18143B390), the applicant stated that the HVAC system details are not finalized and therefore not described in the DCA to the level of detail requested by the NRC staff. The applicant noted that ITAAC in DCA Part 2, Tier 1, Section 3.3, “Reactor Building Heating Ventilation and Air Conditioning System,” verified the design commitment to maintain building pressures negative relative to the outside environment. The applicant also noted that the design would be completed in conformance with the specifications of the Sheet Metal and Air Conditioning Contractors’ National Association (SMACNA) referenced in DCA Part 2, Tier 2, Section 9.4. The applicant stated that conformance with the specifications of the referenced SMACNA standards would be verified through the initial test program. The applicant stated that the COL applicant is responsible for implementing the initial test program and establishing the periodic testing and inspection requirements in accordance with RG 1.140. Section 14.2 of this SER describes the NRC staff’s review of the initial test program for the applicant’s implementation of the SMACNA standards for the design of the HVAC system. The referenced SMACNA standards, while not specifically endorsed by the NRC, contain appropriate guidance that the NRC staff finds acceptable for establishing design criteria for HVAC systems. Therefore, the NRC staff has reasonable assurance that the design of the HVAC system will be consistent with the requirements of 10 CFR Part 20, Subpart H, regarding the use of engineering controls to limit radiation exposure.

The NuScale application does not describe the instruments and controls necessary for closing the dampers on a signal other than smoke or fire (e.g., high radiation) and does not state that the operators will perform a manual action to shut the fire dampers following an accident. The NRC staff reviewed the source terms and design features for managing airborne radioactive material in the RXB following an accident. In its response to questions dated May 10, 2018 (ADAMS Accession No. ML18136A871), the applicant stated that the RXB HVAC RBVS system uses smoke dampers to minimize leakage between the pool area and other portions of the building. DCA Part 2, Tier 2, Section 9.4.2.2, states that each NPM bay has an exhaust air vent that incorporates a fire damper and a blast damper. The NRC staff reviewed several sections of the application, including DCA Part 2, Tier 2, Chapter 7 and Sections 9.4, 9.5.1, and 12.3, and was unable to find any discussion related to the instruments or controls that would shut the NPM fire dampers on a signal other than smoke or fire (e.g., high radiation). Using a risk-informed approach, the NRC staff considered the circumstances that could create a safety concern following an accident. The primary risk to operators or equipment survivability involves a core damage event with a failure of the RBVS exhaust fans as well as an open NPM bay exhaust damper. The NRC staff concludes that there is a low risk of these events occurring

concurrently. Therefore, the design and operational procedure for the RBVS system provide reasonable assurance of adequate protection of SSCs and occupational workers.

The NRC staff reviewed how the applicant complied with the requirements of 10 CFR 50.34(f)(2)(xxvi) regarding the design of components to minimize leakage from systems outside of containment. In its response to questions dated March 16, 2018 (ADAMS Accession No. ML18075A285), the applicant responded to staff questions about potential leakage from systems outside containment that could contain highly radioactive fluid, consistent with the requirements of 10 CFR 50.34(f)(2)(xxvi). Specifically, 10 CFR 50.34(f)(2)(xxvi) requires applicants to provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term (i.e., a core damage source term) radioactive materials following an accident. However, the applicant has not provided information necessary for the staff to address the leakage control and detection in systems outside containment, such as the maximum allowable total leakage from the systems used to perform combustible gas monitoring. Therefore, the staff concludes that the applicant does not meet the requirements of 10 CFR 50.34(f)(2)(xxvi) at the DC stage, and a future COL applicant will need to address this issue.

Further, the NRC staff is unable to reach a conclusion that the system can be re-isolated in order to mitigate potential leakage from these systems that may impact the ability to demonstrate compliance with the requirements of 10 CFR 50.34(f)(2)(xxviii) to limit radiation exposure to control room operators; the requirements of 10 CFR 52.47(a)(2)(iv) to limit exposure to members of the public; and the requirements of 10 CFR 50.34(f)(2)(vii) to perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. The applicant stated that it did not intend to revise DCA Tier 1, Chapter 4, to include design interface items pursuant to 10 CFR 52.47(a)(26), nor propose design acceptance criteria for the system, nor complete the design.

Therefore, the NRC staff recommends that the Commission include language in the proposed rule stating that the NRC is not making a finding on the design of components to minimize and control leakage from systems outside containment. This includes potential leakage from these systems that could impact the offsite dose analyses, the dose analyses for the MCR, and, if necessary, the ability to safely re-isolate these systems after monitoring has been initiated. Specifically, Section VI of Appendix G to 10 CFR Part 52 for the DC for the NuScale SMR will state that the design and evaluation of the leakage from the combustible gas monitoring loop are not considered resolved within the meaning of 10 CFR 52.63(a)(5), and Section IV to Appendix G will state that the COL applicant is responsible for providing sufficient design information demonstrating that the requirements of 10 CFR 50.34(f)(2)(xxviii) are met with respect to potential radiation releases under accident conditions from the systems used for post-accident hydrogen and oxygen monitoring. The COL applicant is to provide assurance that post-accident leakage from these systems does not result in the total MCR dose exceeding the dose criteria (i.e., 50 mSv (5 rem)) for the surrogate event with significant core damage or include design features in accordance with 10 CFR 50.34(f)(2)(xxvi) and 10 CFR 50.34(f)(2)(xxviii) to provide assurance that the dose criteria are not exceeded, or both. The COL applicant will also provide information to verify, as appropriate, that post-accident leakage from these systems does not result in the total dose for the surrogate event with significant core damage exceeding the offsite dose criteria, as required by 10 CFR 52.47(a)(2)(iv). In addition, if manual actuation is required to re-isolate the system in

order to contain potential leakage, the COL applicant will demonstrate that this can be done safely and within the requirements of 10 CFR 50.34(f)(2)(vii).

The NRC staff evaluations documented in Chapters 6 and 15 of this SER contain additional information regarding the methods used by the applicant to determine core damage source terms and the evaluation of dose consequences for those core damage source terms.

12.3.4.1.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

All plant radiation-monitoring equipment is designed to alert operators and other station personnel to changing or abnormally high radiation conditions in the plant to prevent possible personnel overexposures, to aid health physics personnel in keeping worker doses ALARA, and to limit releases to the environment and public. The area radiation monitors supplement the personnel and area radiation survey provisions of the health physics program, which DCA Part 2, Tier 2, Section 12.5, "Operational Radiation Protection Program," directs the COL applicant to describe (see COL Item 12.5-1). The area radiation monitors must comply with the applicable requirements of 10 CFR Part 20 and 10 CFR Part 50 and should conform to the personnel radiation protection guidelines in RGs 1.97, 8.2, and 8.8 and the guidance of DSRS Section 12.3–12.4.

Radiation indications from the fixed airborne and area monitors can be read locally and in the MCR. Alarms are also provided both locally and in the MCR; some monitors also alarm in the waste management control room. ANSI/ANS HPSSC-6.8.1-1981, referenced by DCA Part 2, Tier 2, and DSRS Sections 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.

DCA Part 2, 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 DCA Part 2, Tier 2, Table 12.3-11 and Table 12.3-12, "Fixed Area Radiation Monitors." DCA Part 2, 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. 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. The under-the-bioshield radiation monitors in the NuScale design are high-range radiation monitors that are designed to provide radiation dose information under the bioshield following accident conditions. Therefore, the under-the-bioshield monitors meet the intent of 10 CFR 50.34(f)(2)(xvii) to monitor radiological conditions during an accident, despite being outside containment. These monitors meet the range and placement criteria specified in NUREG-0737, Task Action Plan Item II.F.1.3. The NRC staff verified that the applicant identified these monitors as environmentally qualified for source terms corresponding to non-core-damage accidents and are post-accident monitoring Type B and Type C variables in accordance with the guidance in RG 1.97, Revision 3. The design-basis analysis source term adopted by NuScale is consistent with the methodology outlined in SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," dated August 16, 2019 (ADAMS Accession No. ML19107A455). The applicant stated that these radiation monitors are relied upon to assess the presence of core damage. The source terms to be used for equipment survivability, and the criteria to be applied to equipment expected to be able to perform a function following a core damage event, are not included in the criteria for the NRC staff review of Chapter 12 of the application. The NRC staff's evaluation of the radiological conditions associated with core damage source terms that

are used to assess equipment survivability is discussed in Chapter 15 of this SER. The NRC staff evaluation of the survivability of equipment, as discussed in SECY-19-0079, is contained in Section 19.2, "Severe Accident Evaluation," and Section 15.0.3, "Radiological Consequences of Design Basis Accidents," of this SER. Further information regarding the NRC staff's evaluation of the high-range containment radiation monitors with respect to 10 CFR 50.49(e)(4) is contained in Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," of this SER and in the SER for TR-0915-17565. Therefore, the staff finds that the containment high-radiation monitors comply with the requirements of 10 CFR Part 20, Subpart F, "Surveys and Monitoring," and the requirement of 10 CFR 50.34(f)(2)(xvii) with regard to the requirement to provide radiation monitoring instrumentation for normal operation and non-core-damage events.

DCA Part 2, Tier 2, states that the NuScale design complies with the requirements of 10 CFR 50.68(b) in lieu of 10 CFR 70.24, as allowed by 10 CFR 70.24(d)(1). DCA Part 2, 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 DCA Part 2, Tier 2, Section 12.3. In addition, a local area radiation monitor is mounted on the refueling bridge with a local and MCR alarm function that monitors refueling activities. In DCA Part 2, Tier 2, Section 9.1.1.1, "Design Basis," the applicant stated that the radiation monitors described in DCA Part 2, Tier 2, Section 12.3.4, were provided for compliance with 10 CFR 50.68(b)(6). The applicant also stated in DCA Part 2, Tier 2, Section 12.3.4.1, 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). DCA Part 2, Tier 2, Revision 1, Table 12.3-12, shows that there are 10 reactor pool gamma radiation monitors on the 38.4-meter (126-foot) elevation of the RXB. DCA Part 2, 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 DCA Part 2, 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 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, and is, therefore, acceptable.

DCA Part 2, 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). DCA Part 2, 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. DCA Part 2, 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 DCA Part 2, Tier 2, Section 7.2, discusses the ERDS connection. 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). Section I of Appendix E to 10 CFR Part 50 states that each applicant for a COL under 10 CFR Part 52, Subpart C, "Combined Licenses," is required by

10 CFR 52.79, “Contents of Applications; Technical Information in Final Safety Analysis Report,” to include in the application plans for coping with emergencies. COL Item 13.3-3 directs a COL applicant to follow NUREG-1394, “Emergency Response Data System (ERDS) Implementation,” Revision 1, issued June 1991 (ADAMS Accession No. ML0807900380), to satisfy these requirements. 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 acceptable because the applicant 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).

12.3.4.1.5 Minimization of Contamination

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

DCA Part 2, 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.

DCA Part 2, Tier 2, Revision 3, Section 12.3.6.1, “Facility Design Objectives for 10 CFR 20.1406,” describes four design objectives and two operational program objectives used by the applicant during the design phase and specified for use by COL 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 NRC staff determined that the general design features described by the applicant are in accordance with this design philosophy and demonstrate compliance with the requirements of 10 CFR 20.1406. These features include measures to minimize facility contamination and contamination of the environment and features to facilitate decommissioning. DCA Part 2, Tier 2, Table 12.3-13, “NuScale Power Plant Systems with NRC RG 4.21 Evaluation,” lists systems that were evaluated using the guidance in RG 4.21. DCA Part 2, 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, DCA Part 2, Tier 2, includes COL Item 12.3-7, which directs the COL applicant to develop a plantwide RG 4.21 program to address the operational and programmatic considerations and site radiological environmental monitoring aspects of the minimization of contamination program 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. The NRC staff does not review operational programs during the design phase; therefore, it is acceptable for COL applicants to address the operational considerations as described in the COL Item 12.3-7.

The NRC staff reviewed the design features of the pool leakage detection system (PLDS). The NRC staff examined this area of the plant design based on the guidance contained in RG 4.21. The guidance contained in RG 4.21 addresses leakage from SFPs. As part of the review, the NRC staff also considered information contained in the "Liquid Radioactive Release Lessons Learned Task Force Final Report," dated September 1, 2006 (ADAMS Accession No. ML062650312), related to releases from SFPs and the causes for those releases. The report notes that the potential exists for unplanned and unmonitored releases of radioactive liquids to migrate off site undetected, including those portions of SFPs that are not visible to operators. The task force identified leakage from SFPs as one of the main components resulting in ground water contamination. The report also describes an event in which the liner leakage detection system became clogged with boric acid precipitate. The PLDS is provided to detect leakage from the UHS and the other connected pools, such as the SFP. The NRC staff conducted an audit of the PLDS for the UHS (see "Audit Summary for the Phase II Regulatory Audit of the Pool Leakage Detection System for the Ultimate Heat Sink for NuScale Power, LLC") to review specific information about the design of the PLDS and the basis of the PLDS design. The PLDS works in conjunction with the radioactive waste drain system to collect and quantify leakage from the UHS pool. The NRC staff reviewed the design features that would divert waste to the LRWS based on indications of water level or radiation detection and alert the operators of UHS pool leakage. As result of the PLDS audit, the applicant added descriptions of the leakage detection capability of the PLDS, the design features provided for ensuring that leakage detection channels remain clear, the design features provided to prevent cross-contamination of PLDS samples, and information about leakage detection channels on vertical walls of the UHS.

The NRC staff reviewed the design features of SSCs provided for minimization of contamination described in DCA Part 2, Tier 2, Tables 12.3-14 through 12.3-44. These tables discuss the specific features in the NuScale design provided to show how the applicant addressed the guidance in RG 4.21 and the requirements in 10 CFR 20.1406. The NRC staff also reviewed the layouts for these systems to verify the containment and control of radioactive materials. Based on the discussion above and the information provided by the applicant, the NRC staff concludes that there is reasonable assurance that the design features described by the applicant will assure compliance with the design requirements of 10 CFR 20.1406 and are, therefore, acceptable.

12.3.4.2 Dose Assessment

This section provides information on the dose assessment for both normal operations, including refueling, and post-accident actions. The NRC staff reviewed DCA Part 2, 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

DCA Part 2, Tier 2, adheres to these RGs and NRC staff positions, the NRC staff can conclude that the relevant requirements of 10 CFR Part 20 and other applicable regulations have been met. In addition, the NRC staff selectively compared, based on the radiological significance of the task, the applicant's dose assessment for specific functions and activities against the experience of operating PWRs. Radiation exposures to operating personnel shall not exceed the occupational dose limits specified in 10 CFR 20.1201, and doses 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 DCA Part 2, Tier 2, Sections 12.3 and 12.4, and other related sections of DCA Part 2, 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 DCA Part 2, Tier 2, is consistent with the guidance in these RGs and NRC staff positions, the NRC staff concludes 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.

12.3.4.2.1 Post-Accident Sampling

The NRC staff notes that by letter dated January 31, 2019 (ADAMS Accession No. ML19031C975), as amended by the RAI response dated July 26, 2019 (ADAMS Accession No. ML19207A534), NuScale submitted an exemption request from 10 CFR 50.34(f)(2)(viii). NuScale's request for an exemption to 10 CFR 50.34(f)(2)(viii) is evaluated in Section 9.3.2 of the SER. As discussed in Section 9.3.2 of this SER, since the NuScale design will be exempt from 10 CFR 50.34(f)(2)(viii) and the applicant has removed all information from the DCA indicating that post-accident samples will be taken, the NRC staff did not assess the radiological dose consequences to a worker obtaining and analyzing post-accident RCS and containment atmosphere samples following an accident.

12.3.4.2.2 Post-Accident Combustible Gas Monitoring

The regulation in 10 CFR 50.44(c)(4) requires that equipment for monitoring hydrogen and oxygen must be functional, reliable, and capable of continuously measuring the concentrations of these gases following a significant beyond-design-basis accident. The process of performing post-accident hydrogen and oxygen monitoring in the NuScale SMR design requires the containment to be unisolated and the potentially highly radioactive containment atmosphere to be pumped through the containment evacuation system, sampling system (where the monitors are located), and containment flood and drain system. The NRC staff has reviewed a scoping calculation provided by the applicant in its response to questions dated September 5, 2019 (ADAMS Accession No. ML19248D680), and has reasonable assurance that operators would be able to initiate combustible gas monitoring while remaining compliant with 10 CFR Part 20 and 10 CFR 50.34(f)(2)(vii).

Some of the containment isolation valves that allow passage of the gas for the combustible gas monitoring system can only be closed locally. Consideration of the potential operator radiation exposure that may occur should isolation of the combustible gas monitoring system be required has not been evaluated by the applicant or the NRC staff. The NRC staff discusses leakage from systems outside of containment during combustible gas monitoring in Section 12.3.4.1.3 of this SER.

12.3.4.2.3 Operations and Maintenance Exposure Estimates:

The NRC staff reviewed the applicant's estimates for radiation exposures to plant personnel who perform work activities involving normal operations, maintenance and inspections, refueling activities, and waste handling and whether the method of estimating those doses employed the guidance in RG 8.19. RG 8.19 notes that the dose assessment process should establish an objective to develop a systematic process for considering and evaluating possible dose-reducing design changes and associated operating procedure changes as part of the comprehensive ongoing design review and should identify principal ALARA-related changes resulting from the dose assessment. To allow for the expected buildup of radioactive material on and in SSCs, the occupational dose assessment should be based on anticipated radiation conditions after at least 5 years of plant operation. Analysis of the elements of the dose estimate (e.g., radiation levels, task duration, and frequency), treated qualitatively, can be significantly valuable in making engineering judgments on design changes for ALARA purposes. RG 8.19 states that plant experience, which is available from industry groups like EPRI, provides useful information for performing the dose assessment. It is the view of the NRC staff that the use of this type of information is consistent with 10 CFR 52.47(a)(22), which requires applicants to demonstrate how the plant design incorporates operating experience insights. The NRC staff considered NuScale's specific design features and relevant operating experience (e.g., EPRI TR-1015119, "Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction Final Report," issued November 2007) when performing its review. Although the 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. Therefore, the staff finds the proposed exposure estimates acceptable.

To estimate the OREs for the NuScale facility, the applicant identified various work activities and work durations along with the expected significant (greater than 0.001 mSv/h (0.1 mrem/h) radiation fields that would be encountered. The NRC staff used information from Chapter 12 of the Phase I audit (see ADAMS Accession No. ML18124A182) in its review of the dose estimates provided in DCA Part 2, Tier 2, Section 12.4, and the bases for the estimated doses for the work activities described.

The NRC staff also reviewed the dose estimates provided in DCA Part 2, Tier 2, Section 12.4, and compared them to what would be anticipated given the NRC staff operating experience with large light-water reactors. As stated in RG 8.19, an analysis of the elements of the man-rem estimate (e.g., radiation levels, task duration, and frequency), treated qualitatively, can be of significant value in making engineering judgments regarding design changes for ALARA purposes. An expected result of the dose assessment process described in the guidance is that various dose-reducing design changes and innovations will be incorporated into the design. Therefore, the NRC staff discussed with the applicant the dose estimates stated in DCA Part 2, Tier 2, Section 12.4, and the methods used by the applicant to perform the dose estimates. In its response to questions dated March 21, 2018 (ADAMS Accession No. ML18080A113), the applicant stated that using operating experience documented in EPRI TR-1015119, it conducted a study that demonstrated that the estimated dose rates for an operator performing SG maintenance activities are appropriate for the NuScale design. 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 RPV steel wall or through the RPV and CNV steel walls, the radiation exposure is significantly reduced. Therefore, the applicant concluded that it had appropriately accounted for the radioactive cobalt buildup in the SGs and that the results of the

ORE estimates in DCA Part 2, Section 12.4, are appropriate. The NRC staff performed confirmatory calculations utilizing the MicroShield modeling code that used input values for source terms, materials, and dimensions provided by the applicant, resulting in dose rate estimates consistent with the dose rates provided by the applicant. Although the NRC staff disagrees with some of the information in the applicant's response—specifically, with respect to determining applicable operating experience—the NRC staff's confirmatory calculations demonstrated that DCA Part 2, Tier 2, adequately accounts for outage ORE and that the outage ORE meets the applicable regulatory requirements and is, therefore, acceptable.

The NRC staff reviewed whether the applicant assessed the impact of constructing additional NPMs while existing NPMs were operating. The NRC staff determined that the applicant had provided a construction worker dose estimate. Since the exposure estimate was below the 10 CFR 20.1301 exposure limits for members of the public, the proposed exposure estimate was acceptable to the NRC staff. The applicant also provided COL Item 12.4-1, which states that the COL applicant is responsible for providing a dose estimate for construction of a NuScale facility near an existing operating nuclear power plant that is not a NuScale-designed power plant. Based on the information provided by the applicant, the NRC staff concluded that there is reasonable assurance that the construction worker doses can be controlled to within the limits of 10 CFR Part 20 during the construction of NuScale Power Plants.

12.3.5 Combined License Information Items

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

Table 12.3-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 12.3

COL Item No.	Description	DCA Part 2, 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	DCA Part 2, 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
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

Note: For COL Item 12.3-7, Objectives 5 and 6 are applicant-defined terms and are described in SER Section 12.3.4.1.5 regarding features for compliance with 10 CFR 20.1406 and 10 CFR 52.47(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); GDC 4, 19, and 61, as they relate to the information on radiation sources provided by the applicant; 10 CFR 50.34(f)(2)(xvii), as it relates to radiation monitoring; 10 CFR 50.68 and 10 CFR 70.24, as they relate to radiation monitoring where fuel is stored or handled; GDC 63 and 10 CFR Part 50, Appendix E, as they relate to monitoring for excessive radiation levels in the facility; GDC 14 and 30, as they relate to RCS pressure boundary radiation monitoring; 10 CFR 52.47(b)(1), as it relates to the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that a facility that incorporates the 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.

Based on the above, the NRC staff concludes that the portions of the design that were evaluated, as described in NuScale DCA Part 2, Tier 2, adequately address radiation protection design features. Portions of the design that are not considered resolved by the DC within the

meaning of 10 CFR 52.63(a)(5) and must be addressed by the COL applicant include the following:

- radiation shielding for the NPM to RXB steam gallery penetrations

- design features and analysis for leakage of systems outside of containment (containment evacuation system, containment flood and drain system, and process sampling system) used for combustible gas monitoring, which may have highly radioactive fluid following an accident

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12.4 Dose Assessment

The staff's review of this section of the DCA is documented in Section 12.3.4.2 of this SER.

12.5 Operational Radiation Protection Program

12.5.1 Introduction

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

12.5.2 Summary of Application

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

DCA Part 2, 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: DCA Part 2, Tier 2, Chapter 16, Section 5.7, addresses TS for the control of high-radiation areas.

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

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

12.5.3 Regulatory Basis

The relevant requirements of NRC regulations for the operational radiation protection program, and the associated acceptance criteria, are described in DSRS Section 12.5, "Operational Radiation Protection Program" (ADAMS Accession No. ML15350A341). The guidance in DSRS

Section 12.5 and the applicable regulatory requirements will be addressed by the staff during the review of a potential future COL application.

12.5.4 Technical Evaluation

NuScale DCA Part 2, Tier 2, Section 12.5, states that the COL applicant must provide the radiation protection program. DCA Part 2, Tier 2, Section 12.1, states that the COL applicant must provide the ALARA program. DCA Part 2, Tier 2, Section 12.3, states that the COL applicant must provide programs to minimize contamination of the facility. The review of these programs is beyond the scope of review conducted for a DCA. As described in DSRS Section 12.5, the COL applicant is also responsible for providing the description of the operational program and proposed implementation milestones for the leakage control program required by 10 CFR 50.34(f)(2)(xxvi) and the ground water protection program and procedures required by 10 CFR 20.1406, to the extent that they are not described in other sections of the application.

12.5.5 Combined License Information Items

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

Table 12.5-1 NuScale COL Information Items for DCA Part 2, Tier 2, Section 12.5

COL Item No.	Description	DCA Part 2, 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 low as reasonably achievable in accordance with 10 CFR 20.1101.	12.5

12.5.6 Conclusion

The NRC staff does not review operational programs during the DC 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, 10 CFR 50.34(f)(2), and other regulations applicable to these areas during the COL review.