December 28, 2022 Docket No. 52-050

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 12, "Radiation Protection," Revision 0

REFERENCES:
1. NuScale letter to NRC, "NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content," dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, "NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, 'NuScale Standard Design Approval Application (SDAA),'" dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, "Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application," Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, "NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application," dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 12 of the Standard Design Approval Application, "Radiation Protection," Revision 0. This chapter supports Part 2, "Final Safety Analysis Report," (FSAR) of the NuScale Standard Design Approval Application (SDAA) (Reference 1). NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 12. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 12 and technical content generally expected by the NRC; and 2) resolves identified technical issues that may have adversely impacted acceptance, docketing, or technical review of the application. Enclosure 2 provides NuScale's response to Reference 3 for the Chapter 12 observation.

Enclosure 1 contains Part 2 of the report entitled Chapter 12, "Radiation Protection," Revision 0, nonpublic version. As this version contains Sensitive Unclassified Nonsafeguards Information (SUNSI), NuScale requests that the nonpublic version (enclosure 1), be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. Enclosure 2 contains the public version.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 28, 2022.

Sincerely,

Carrie Fosaaen
Senior Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Brian Smith, NRC
Michael Dudek, NRC
Getachew Tesfaye, NRC
Bruce Bavol, NRC
David Drucker, NRC

Enclosure 1: SDAA Part 2 Chapter 12, “Radiation Protection,” Revision 0, (nonpublic)
Enclosure 2: SDAA Part 2 Chapter 12, “Radiation Protection,” Revision 0, (public)

Enclosure 1:

SDAA Part 2 Chapter 12, “Radiation Protection,” Revision 0, (nonpublic)

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 2:

SDAA Part 2 Chapter 12, “Radiation Protection,” Revision 0, (public)

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Contents

Section	Description
A	Chapter 12, “Radiation Protection,” Revision 0, public
B	Readiness Assessment Review responses for Chapter 12

Section A

NuScale US460 Plant
Standard Design Approval Application

Chapter Twelve
Radiation Protection

Final Safety Analysis Report

Revision 0

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF FIGURES

CHAPTER 12 RADIATION PROTECTION

12.1 Ensuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable

Plant design features, administrative programs, and procedures assist in maintaining occupational radiation exposure as low as reasonably achievable (ALARA) in accordance with 10 CFR 20.1101. The purpose of this chapter is to describe the policies, programs, and features for radiation protection.

12.1.1 Policy Considerations

12.1.1.1 Design and Construction Policies

The design incorporates ALARA principles and policies. Design reviews integrate facility layout, radiation shielding, building ventilation, material selection, and instrumentation design with layout requirements, flow of personnel, plant security, access control, and health physics considerations.

12.1.1.2 Operational Policies

Section 12.1.3 addresses implementation of the operational ALARA policy.

12.1.1.3 Compliance with 10 CFR 20 and Regulatory Guides 8.8, 8.10, and 1.8

The design complies with 10 CFR 20, and Regulatory Guides (RG) 8.8, Revision 3 and 8.10, Revision 1-R. Section 12.1.3 and Section 12.5 discuss plant operation policy considerations described in RG 8.8, RG 8.10, and RG 1.8, Revision 3 and demonstrated by an application that references the NuScale Power Plant US460 standard design. The design meets the guidance of RG 8.8, Sections C.2 and C.4 that address facility, equipment, and instrumentation design features. Section 12.3 addresses plant features that are examples of compliance with RG 8.8.

12.1.2 Design Considerations

This section describes the methods and features that allow the policy considerations of Section 12.1.1 to be implemented. Section 12.3 presents design features and attributes for maintaining ALARA personnel exposures.

12.1.2.1 General Design Considerations for Maintaining Exposures as Low as Reasonably Achievable

Experienced engineers use an inter-disciplinary ALARA design review process to ensure the design conforms to 10 CFR 20.1101(b) and RG 8.8. The design engineers receive training in radiation science and in ALARA design concepts that are incorporated into the facility design. During the design work, calculations are developed for the radiation source terms, radiation shielding, radiation zone maps, and occupational radiation exposure estimates. The results of the occupational radiation exposure estimates are provided as feedback to the design engineers during multi-discipline design phase reviews, which are used to develop additional ALARA features, as necessary.

The ALARA philosophy guiding the design ensures exposures are minimized by designing structures, systems, and components to achieve the following objectives:

- attain optimal reliability and maintainability, thereby reducing maintenance requirements for radioactive components.
- reduce radiation fields, thereby allowing operations, maintenance, and inspection activities to be performed in reduced radiation fields.
- reduce access, repair, and equipment removal times, thereby reducing the time spent in radiation fields.
- accommodate remote and semi-remote operation, maintenance, and inspection, thereby reducing the time spent in radiation fields.

To minimize production, distribution, and retention of activated corrosion products, the design considerations include appropriate material selection and proper water chemistry. Section 12.1.2.2 describes the design considerations in more detail. Section 12.3 describes design considerations and features addressing 10 CFR 20.1406 to minimize contamination of the facility, environment, waste generation, and decommissioning.

12.1.2.2 Equipment Design Considerations for Maintaining Exposures as Low as Reasonably Achievable

12.1.2.2.1 General Design Criteria

The engineering design process requires consideration of the applicable RGs, including RG 8.8, as part of the ALARA design criteria. The following paragraphs describe examples of using ALARA principles in design.

12.1.2.2.2 Equipment Design Considerations to Limit Time Spent in Radiation Areas

Equipment instrumentation and controls are located in accessible areas. Equipment is selected to minimize potential dose to personnel during maintenance. The selected equipment provides drainage capabilities to facilitate maintenance activities and smooth surfaces to reduce potential for contamination and facilitate decontamination efforts. Selection of components includes consideration for high reliability, ease of decontamination, maintenance and replacement, and accessibility. When possible, components located in higher dose rate areas are maintenance-free. During maintenance activities, worker doses are also reduced through the use of

- radiation-damage-resistant materials in high-radiation areas to reduce the need for frequent replacement and to reduce the probability of contamination from leakage.
- stainless steel for constructing or lining components, where it is compatible with the process, to reduce corrosion and provide options for decontamination methods.
- adequate working space for easy accessibility.
- locating valves in areas separated from potentially radioactive components.
- straight-through valve configurations to facilitate maintenance and avoid the buildup of accumulations in internal crevices.

Section 12.1.3 provides site-specific information describing how the plant implements the design consideration guidance in RG 8.8.

12.1.2.2.3 Equipment Design Considerations to Limit Component Radiation Levels

Equipment design considerations directed toward minimizing radiation levels near equipment or components requiring personnel attention include the following:

- The materials selected for components are chosen to meet service requirements and minimize cobalt-containing materials (e.g., Stellite) coming in contact with the primary coolant system.
- The design minimizes, to the extent possible, cobalt or nickel-containing alloys for the material to reduce the production of cobalt-60 and cobalt-58.
- Equipment is designed with provisions to limit leaks, and the plant is designed to collect and control leaked fluid by using sumps and drip pans piped to floor drains that are routed to the liquid radioactive waste system.
- Components requiring periodic servicing or maintenance are separated, when possible, from highly radioactive sources.
- Piping design avoids the creation of stagnant legs, uses sloping pipe runs, and locates connections above the pipe centerline.
- Crud deposition within the primary coolant system is reduced by smooth surfaces that minimize crud traps.
- The liquid radioactive waste system is provided with a clean-in-place skid that allows flushing system components with demineralized water, as needed.

12.1.2.2.4 Water Chemistry

Reactor coolant water chemistry is controlled during operation to minimize corrosion of surfaces in contact with the coolant and minimize the production of activated corrosion products. The chemical and volume control system removes impurities and suspended solids from the reactor coolant.

12.1.2.3 Facility Layout General Design Considerations for Maintaining Radiation Exposures as Low as Reasonably Achievable

The design utilizes operating experience and lessons learned from past plant designs to provide an efficient layout that provides reduced personnel exposures.

The radiation protection support facilities are located in the Radioactive Waste Building and include change rooms, offices, calibration facilities, counting room, hot machine shop, and equipment and personnel decontamination facilities. The Radioactive Waste Building also serves as the access portal to the radiologically controlled area, and includes dosimetry issue and personnel contamination monitors.

12.1.2.3.1 Minimizing Personnel Time Spent in Radiation Areas

Facility design considerations used in minimizing personnel time spent in radiation areas include

- space within cubicles for a laydown area for special tools and ease of servicing activities.
- instrumentation readouts, monitors, and control points located in low radiation zones.
- provisions for removing components and transporting them to low radiation zones where shielding and special tools are available.
- use of reach rods and remote operators for valves in high-radiation areas.

12.1.2.3.2 Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment

Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include the following:

- shielding between components.
- labyrinth entrances to reduce radiation streaming out of cubicle entrances.
- shield wall penetrations configurations to prevent "line-of-sight" streaming.
- pipe chases for pipes containing significant radioactive material.
- radiation areas where station personnel spend substantial time are designed to the lowest practical dose rates.
- curbing and sloped floors direct leakage to local drains or sumps to limit the spread of contamination from liquid systems.
- tanks containing radioactive liquids are designed with sloped bottoms toward outlets and flushing or cleaning features.

- spare connections on tanks and other components located in high radiation areas to allow for greater operational flexibility.
- pumps chosen to minimize leakage and provide housing drains.
- radiation sources are separated from occupied areas where practicable (e.g., pipes or ducts containing high radioactive fluids not passing through occupied areas).
- when permanent shielding is impractical, provisions for temporary shielding.
- instrumentation designed, selected, and located with consideration for long service life, ease and low frequency of calibration, and low crud accumulation.
- provisions to permit the rapid manipulation of shielding and insulation for equipment that requires periodic inspection or service.
- adequate space for moveable or temporary shielding for sources.
- means to control contamination and to facilitate decontamination of potentially contaminated areas.
- piping for "clean services" (e.g., station air, potable water, nitrogen) is located separate from piping for contaminated systems to avoid cross-contamination.
- features that permit remote removal, installation, inspection, or servicing of radioactive components.
- design features such as ventilation isolation or filtration and heating ventilation and air conditioning design such that air flows from areas of lower radioactivity to areas of higher radioactivity to protect against airborne contamination.

12.1.3 Operational Considerations

COL Item 12.1-1: An applicant that references the NuScale Power Plant US460 standard design will describe the operational program to maintain exposures to ionizing radiation as far below the dose limits as practical, as low as reasonably achievable (ALARA).

12.2 Radiation Sources

This section describes the sources of both contained and airborne radiation that provide input to:

- radiation shielding design calculations.
- ventilation systems design.
- radwaste systems design, including the classification of structures, systems, and components per Regulatory Guide (RG) 1.143.
- radiation protection assessment, including personnel protection.

12.2.1 Contained Sources

The contained radiation sources are developed for normal operation and shutdown conditions. The design basis primary coolant activity concentrations (Section 11.1) are the basis for the contained radiation sources. They are determined by propagating this radionuclide activity through various plant systems using the parameters and assumptions provided in this section. In order for the radiological source terms to be used in shielding calculations, the isotopic inventory is used to calculate the intensity and energy spectrum of the total emitted radiation. The ORIGEN-S code is used to bin the particle emissions into default energy bins based on the activity of each individual isotope. This section describes the radiation sources that provide part of the basis for the design of radiation shielding features. The radiation zone maps in Section 12.3 include drawings showing locations of contained sources.

12.2.1.1 Reactor Core

During normal reactor operations, neutron and gamma radiation are released from the reactor core and from the primary coolant. This radiation is attenuated by the reactor internals, the reactor vessel, the containment vessel, the water surrounding the NuScale Power Module (NPM), the reactor pool concrete walls, and by the bioshield. Table 12.2-1 provides the fission neutron source strength as well as neutron energy spectrum information. The n-gamma and fission gamma source strength is internally generated by MCNP6 using the neutron source strength as an input. The fission neutron source utilizes the Watt spectrum for U-235.

12.2.1.2 Reactor Coolant System

Radionuclides present in the reactor coolant system (RCS) are generated from the release of radioactive materials from postulated fuel clad defects and neutron activation of the primary coolant and impurities in the primary coolant.

The contribution of gamma radiation from the primary coolant is comprised of two components: the primary coolant at the steam generator entrance and at the core exit. The nitrogen-16 concentration is calculated at the core exit, the top of the upper riser/steam generator entrance, and at various points along the chemical and volume control system (CVCS) letdown line. Because of the low flow velocity of the primary coolant, the short half-life of nitrogen-16 causes it to decay by about one order of magnitude by the time it reaches the steam generator entrance. The corrosion and wear activation products (commonly referred to as CRUD) are uniformly modeled on a primary coolant mass basis. Tables 12.2-2 and 12.2-3 provide the primary coolant gamma spectra. Table 12.2-4 tabulates the nitrogen-16 concentration at several locations.

12.2.1.3 Chemical and Volume Control System

The CVCS processes a portion of the RCS through heat exchangers, demineralizers, and filters. The treated primary coolant is then returned to the RCS (Section 9.3.4). During this treatment process, components of the CVCS can become radiation sources due to soluble and non-soluble radionuclides in the primary coolant. The CVCS contained sources are determined using the design basis coolant source term from Section 11.1 (Table 11.1-4).

<u>Mixed-Bed and Cation Bed Demineralizers</u>

The CVCS mixed-bed demineralizers are modeled in continuous operation during the entire fuel cycle. The decontamination factors assumed are listed in Table 11.1-2.

The CVCS cation bed demineralizers are modeled in operation for one-half of the fuel cycle because they are operated intermittently during the operating cycle for lithium removal. Table 12.2-5 lists the assumed decontamination factors.

Tables 12.2-6 and 12.2-7 list the mixed-bed source terms and source strengths, respectively. These source terms and the associated analyses do not include short-term transients such as CRUD bursts associated with refueling outages. This results in the estimates of activity within some plant structures, systems, and components to not reflect the CRUD-burst related activity, including the CVCS mixed-bed demineralizer values (both columns) in Table 12.2-6 and Table 12.2-7. It is estimated that a CRUD burst could add up to 680 curies of CRUD isotopes to the CVCS mixed-bed demineralizer.

<u>Regenerative and Non-Regenerative Heat Exchangers</u>

The CVCS regenerative and non-regenerative heat exchangers are shell and tube type, as described in Section 9.3.4. To calculate the radiological source term, the heat exchangers are assumed to be completely filled with primary coolant. The major source term model assumptions are listed in Table 12.2-5. The source term for the RCS water is found in Table 11.1-4.

<u>Module Heatup System Heater</u>

The module heatup system heater is modeled with 100 percent design basis primary coolant using the parameters listed in Table 12.2-5.

Resin Transfer Pipe

A generic resin transfer line is modeled assuming it is 100 percent obstructed by resin beads from the CVCS mixed-bed demineralizer using a bulk dry resin density. Table 12.2-5 lists the parameters for modeled the generic resin transfer lines. The source term used for the spent resin transfer line is the CVCS mixed-bed demineralizer decayed for 48 hours, as provided in Table 12.2-6 and Table 12.2-7.

Reactor Coolant Filters

The reactor coolant filters are cartridge filters located downstream of the ion exchangers that clean the primary coolant in the CVCS and are modeled to remove CRUD particulate. Table 12.2-5 lists the assumed filter efficiency. Tables 12.2-6 and 12.2-7 list the filter source term and source strength, respectively.

12.2.1.4 Pool Cooling and Cleanup System

During normal conditions of operation, the pool cooling and cleanup system (PCWS) provides for water level control and temperature maintenance of the reactor pool, the refueling pool and the spent fuel pool. It also removes impurities to reduce radiation exposures and to maintain water chemistry and clarity. Section 9.1.3 further describes these system.

The pool cooling heat exchangers are conservatively assumed to be filled with reactor pool water even though the shell side is normally filled with site cooling water. Because the majority of the radioactivity consists of tritium, these heat exchangers do not represent a significant radiation source that requires radiation shielding. The PCWS demineralizers are assumed to collect the entire inventory of cleanable radioisotopes in the pool water as shown in Table 12.2-9. It is also assumed the PCWS demineralizers operate for two years, collecting the entire reactor pool water radionuclide inventory eight times (assuming a plant with 6 NPMs on an 18-month refueling cycle). The PCWS filters are assumed to collect CRUD particulates and are operated for one year.

Table 12.2-8 lists the input assumptions used to develop source terms. Tables 12.2-9 and 12.2-10 provide the radionuclide source terms and source strengths respectively, for the equipment.

12.2.1.5 Liquid Radioactive Waste System

The radionuclide inventory in the liquid radioactive waste system (LRWS) includes fission and activation products originating from the reactor core and the RCS. Table 11.2-3 lists the estimated input flows from various sources to the high-conductivity waste (HCW) collection tanks, and the low-conductivity waste (LCW) collection tanks. These inputs are processed in batches by the liquid radioactive waste processing skids and sent to the HCW and LCW sample tanks for final disposition. Table 12.2-11 lists the assumed values for the LRW processing equipment radionuclide collection efficiencies and geometries.

Tables 12.2-12a and 12.2-12b provide component source terms and Tables 12.2-13a and 12.2-13b provide source strengths.

12.2.1.6 Gaseous Radioactive Waste System

The radionuclide input to the gaseous radioactive waste system (GRWS) comes primarily from the LRWS degasifier, which strips the dissolved gases from the primary coolant that enters the degasifier from the CVCS. The gases from the degasifier are sent to the GRWS for conditioning and processing. Table 12.2-14 lists the assumed values pertaining to the GRWS source geometries and Table 11.3-1 describes the GRWS processing parameters. Tables 12.2-15 and 12.2-16 respectively, provide the GRWS component source terms and strengths.

The radioisotopic inventory shown in Table 12.2-15 for the GRWS guard bed and decay beds results in a lesser RG 1.143 safety classification than what is shown in Table 11.3-2. Because an end of operating cycle degasification evolution could result in a transient radioisotopic inventory that exceeds RW-IIb, the classification of the guard bed and decay beds are increased to RW-IIa to cover such transients, as reflected in Table 11.3-2.

12.2.1.7 Solid Radioactive Waste System

Table 12.2-17 lists the assumed values used to develop the solid radioactive waste system (SRWS) component sources modeling in shielding analysis. Table 12.2-18 lists the radionuclide inventory of the major SRWS components and Table 12.2-19 lists the SRWS component source strengths. For shielding design purposes, it is assumed that the Class A/B/C high integrity container (HIC) storage area contains five HICs loaded with Class B/C dewatered spent resins from the spent resin storage tank, which is decayed for approximately two years. Table 12.2-12b provides the radionuclide inventory of the drum dryer and Table 12.2-13b provides the drum dryer source strength. Storage areas are shielded to limit the radiation level to be compliant with the designated radiation zone.

12.2.1.8 Reactor Pool Water

There are two sources of radioactive material considered for the reactor pool water: primary coolant released during refueling outages and direct neutron activation. Because of the low power and low temperatures in the spent fuel pool, the radionuclide contribution to the pool water from defective fuel assemblies in the storage racks is considered negligible. The primary source of radionuclides in the reactor pool comes from the primary coolant system when an NPM is disassembled in the reactor pool during outages. During refueling outages, after the primary coolant is cleaned by the CVCS, the remaining quantities of radionuclides are released into the pool water during NPM disassembly. The post-CRUD burst cleanup of the primary coolant in the NPM by CVCS continues until the projected dose rate (after NPM disassembly) at 1 meter above the pool water is less than 5 mrem/hr. Table 12.2-8 lists the major input assumptions for the pool water source term and PCWS component shielding modeling.

The radionuclide contribution resulting from neutron activation of the reactor pool water contents is not significant due to the reduced neutron flux in the reactor pool water. The neutron flux at the outside edge of the containment vessel is many orders of magnitude less than the average neutron flux in the core and continues to decrease in the reactor pool borated water. The amount of neutron activation products in the reactor pool water is determined to be insignificant compared to the amount of primary coolant radionuclides released to the reactor pool water during refueling outages.

Table 12.2-8 lists the characteristics of the pool surge control storage tank, which is modeled as a vertical cylinder.

Tables 12.2-9 and 12.2-10 provide the source terms and source strengths respectively, for the pool water and pool surge control storage tank.

12.2.1.9 Spent Fuel

Spent fuel stored in the spent fuel racks presents a radiation source that is shielded by the water in the spent fuel pool as well as by the pool walls. The same methodology used to determine the maximum core isotopic source term in Section 11.1 is used to develop the spent fuel source term, resulting in the bounding assumption that the spent fuel racks are filled with irradiated fuel assemblies. Spent fuel gamma ray and neutron source strengths are considered in the evaluation of radiation levels for fuel handling and spent fuel storage.

Spent fuel gamma ray source strengths are presented in Table 12.2-20 for a spent fuel rack full of irradiated fuel assemblies. Spent fuel neutron source strengths are given in Table 12.2-21 for the same spent fuel rack.

12.2.1.10 In-Core Instruments

There are 12 fuel assemblies distributed in the reactor core that are instrumented with in-core instruments. Each of the 12 instruments contains self-powered neutron detectors and thermocouples. During reactor operations, the in-core instruments are irradiated, resulting in activation. Table 12.2-22 lists the major input assumptions and Table 12.2-23 provides the gamma spectra.

12.2.1.11 Control Rods and Secondary Source Rods

<u>Control Rod Assemblies</u>

Because the reactor core operates in an all-rods-out configuration, it is assumed that only the tip of the control rod is irradiated. This portion of the control rod assembly (CRA) consists of Ag-In-Cd neutron absorber. Table 12.2-24 lists the major input assumptions and Table 12.2-25 lists the CRA gamma spectra. The activated source from the CRA tips peaks early, therefore the source term reported is for end of Cycle 1.

Secondary Source Rod

The secondary source is antimony and beryllium (Sb-Be) and is irradiated for 40 cycles. Flux is the same as for the in-core instruments (Section 12.2.1.10).

Table 12.2-26 lists the gamma ray source strengths associated with the secondary source rods.

12.2.1.12 Secondary Coolant System

Primary-to-secondary leaks through the steam generator can introduce primary coolant activity into the secondary system with the resultant contamination level being dependent on the activity level in the primary coolant and the magnitude of the steam generator leak. Because the condensate polishing system is a full-flow system, the condensate polishers are evaluated for the radioactive material that could accumulate on the resins during the period between resin regenerations. Assuming the secondary coolant is at the design basis concentrations (Table 11.1-5), resin decontamination factors consistent with NUREG-0017, and a 10-day resin regeneration period, the accumulation of radioactive material is less than 100 mCi. The model excludes tritium from this activity because tritium does not accumulate on the resin.

12.2.1.13 Post-Accident Sources

The iodine spike design basis source term (Section 15.0.3 describes the maximum primary coolant activity released from the design bases accidents) is evaluated for equipment qualification (EQ) in and around an NPM. Three volumes associated with the NPM are modeled for design basis accident EQ dose consequences from a design basis accident: the reactor pressure vessel and containment vessel combined liquid sump volume, the containment vapor volume, and the bioshield envelope volume. The iodine spike design basis source term maximum post-accident activity concentrations used for EQ evaluation are provided on a mass basis in Table 12.2-31. The specific concentration values in Table 12.2-31 are in excess of the values that would be calculated using the methodology in TR-0915-17565-P-A (Reference 12.2-4), with the design inputs provided in Chapter 11 and Chapter 12. Plateout of activity onto containment surfaces is neglected due to the small containment volume and the lack of surface coatings inside containment. There is also no aerosol removal assumed. Other assumptions for the post-accident EQ source term are listed in Table 12.2-27. The three volumes are evaluated with conservative assumptions, including instantaneous and homogeneous releases into the volume of interest.

Table 12.2-28 lists the integrated post-accident source energy deposition versus time for both photons and electrons for the three evaluated volumes. Table 12.2-28 also tabulates the integrated doses for various times post-accident. Section 3.11 and Appendix 3C contain additional details on EQ. Consistent with 10 CFR 50.34(f)(2)(vii), areas that could contain core damage post-accident sources are evaluated for equipment protection. Section 19.2 addresses equipment protection from a core damage source term.

12.2.1.14 Other Contained Sources

There are no other identified contained sources that exceed 100 mCi.

COL Item 12.2-1: An applicant that references the NuScale Power Plant US460 standard design will describe additional site-specific contained radiation sources that exceed 100 millicuries (including sources for instrumentation and radiography) not identified in Section 12.2.1.

12.2.2 Airborne Radioactive Material Sources

This section describes the airborne radioactive material sources that form part of the basis for design of ventilation systems and personnel protective measures, and also are considered in personnel dose assessment.

12.2.2.1 Reactor Building Atmosphere

Airborne radioactivity may be present in the reactor building (RXB) atmosphere due to reactor pool evaporation or primary coolant leakage. The airborne concentration is modeled as a buildup to an equilibrium concentration given the production rate and removal rate, and on an evaporation rate calculated following the model in the Heating, Ventilating, and Air-conditioning Applications Handbook (Reference 12.2-3). The concentration of tritium in the reactor pool water is developed assuming the primary coolant letdown is recycled to the reactor pool. The concentration of tritium in the primary coolant leakage is developed assuming the primary coolant letdown is recycled back to the RCS. Each case maximizes the tritium concentration in the fluid of interest using the concentrations in Table 11.1-8. The airborne concentration in the air space above the reactor pool is determined by using the peak reactor pool water source term. Table 12.2-29 lists the input parameters.

$$A(\infty) = (C_{pool} \times p_f \times F_{evap}) / (\lambda + (F_{air}/V_{air}))$$

where,

$A(\infty)$ = equilibrium airborne concentration,

C_{pool} = pool water concentration,

p_f = partition fraction,

λ = decay constant,

F_{air}/V_{air} = air change rate, and

F_{evap} = pool evaporation rate = $A/Y(p_w - p_a)(95+0.425V)$, where,

A = area of pool surface, ft^2

Y = Latent heat of evaporation at surface water temperature, Btu/lb

p_w = saturation vapor pressure at surface water temperature, in. Hg

p_a = saturation vapor pressure at room air dew point, in. Hg

V = air velocity of water surface, fpm.

Primary coolant leaks can occur in the RXB from the CVCS. In areas that are routinely occupied, the RXB heating ventilation and air conditioning system provides sufficient air flow to maintain airborne concentrations to acceptable levels where CVCS leaks are a potential. The airborne concentrations in the RXB cubicles are determined using the same equilibrium model as the reactor pool area, but using CVCS leaks for the production term.

$$A(\infty) = (PCA \times p_{leak} \times p_f \times F_{leak}) / (\lambda + (F_{air}/V_{air}))$$

where,

$A(\infty)$ = equilibrium airborne concentration,

PCA = primary coolant activity concentration,

p_{leak} = leak flashing fraction,

p_f = partition fraction,

F_{leak} = primary coolant leak rate,

λ = radioactive decay constant, and

F_{air}/V_{air} = air change rate.

Table 12.2-30 lists the resultant airborne isotopic concentration in the RXB atmosphere. Section 12.3.4 describes monitoring airborne radioactivity within the air spaces of the facility. Section 11.5 describes monitoring gaseous effluents.

12.2.2.2 Turbine Building Atmosphere

Section 12.2.1.12 considers the secondary coolant to be clean for normal operating conditions. Therefore, the Turbine Building atmosphere contains minimal airborne radioactive material.

12.2.3 References

12.2-1 International Atomic Energy Agency, "Combined Methods for Liquid Radioactive Waste Treatment," IAEA-TECDOC-1336, Vienna, Austria, February 2003.

12.2-2 Electric Power Research Institute, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Vols. 1 and 2, EPRI #3002000505, EPRI, Palo Alto, CA, 2014.

12.2-3 American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE), “Handbook: Heating, Ventilating, and Air-conditioning Applications," Atlanta, GA, 2007.

12.2-4 NuScale Power, LLC, "Accident Source Term Methodology," TR-0915-17565-P-A, Revision 4.

Table 12.2-1: Core and Coolant Source Information

Parameter	Value	Units
Fission Neutron Source Strength	2.6E+19	Particles/s
Fission Neutron Energy Spectrum	U-235	-
Total Primary Coolant Source Strength	1.6E+14	Particles/s

Table 12.2-2: Primary Coolant at the Steam Generator Entrance Gamma Source Term

Energy Group	Energy (MeV)		Gamma Source
	Lower Bound	Upper Bound	(photons/sec/gram primary coolant)
1	1.00E-02	2.00E-02	8.9E+04
2	2.00E-02	3.00E-02	5.5E+04
3	3.00E-02	4.50E-02	1.8E+05
4	4.50E-02	6.00E-02	3.3E+04
5	6.00E-02	7.00E-02	1.7E+04
6	7.00E-02	7.50E-02	7.2E+03
7	7.50E-02	1.00E-01	1.4E+05
8	1.00E-01	1.50E-01	3.5E+04
9	1.50E-01	2.00E-01	2.8E+04
10	2.00E-01	2.60E-01	2.5E+04
11	2.60E-01	3.00E-01	8.1E+03
12	3.00E-01	4.00E-01	2.3E+04
13	4.00E-01	4.50E-01	8.2E+03
14	4.50E-01	5.10E-01	8.5E+03
15	5.10E-01	5.12E-01	5.4E+02
16	5.12E-01	6.00E-01	7.8E+03
17	6.00E-01	7.00E-01	1.0E+04
18	7.00E-01	8.00E-01	7.6E+03
19	8.00E-01	9.00E-01	6.1E+03
20	9.00E-01	1.00E+00	3.8E+03
21	1.00E+00	1.20E+00	6.4E+03
22	1.20E+00	1.33E+00	1.2E+04
23	1.33E+00	1.44E+00	2.8E+03
24	1.44E+00	1.50E+00	8.5E+02
25	1.50E+00	1.57E+00	9.4E+02
26	1.57E+00	1.66E+00	1.2E+03
27	1.66E+00	1.80E+00	3.8E+03
28	1.80E+00	2.00E+00	3.2E+03
29	2.00E+00	2.15E+00	1.1E+03
30	2.15E+00	2.35E+00	4.7E+02
31	2.35E+00	2.50E+00	2.7E+03
32	2.50E+00	2.75E+00	1.4E+04
33	2.75E+00	3.00E+00	3.5E+03
34	3.00E+00	3.50E+00	9.0E+02
35	3.50E+00	4.00E+00	1.0E+03
36	4.00E+00	4.50E+00	3.7E+02
37	4.50E+00	5.00E+00	3.4E+02
38	5.00E+00	5.50E+00	3.1E+02
39	5.50E+00	6.00E+00	1.5E+02
40	6.00E+00	6.50E+00	1.0E+06
41	6.50E+00	7.00E+00	6.6E+02
42	7.00E+00	7.50E+00	7.3E+04
43	7.50E+00	8.00E+00	2.1E+01
44	8.00E+00	1.00E+01	1.2E+03

Table 12.2-2: Primary Coolant at the Steam Generator Entrance Gamma Source Term (Continued)

Energy Group	Energy (MeV)		Gamma Source
	Lower Bound	Upper Bound	(photons/sec/gram primary coolant)
45	1.00E+01	1.20E+01	3.0E-01
46	1.20E+01	1.40E+01	0.0E+00
47	1.40E+01	2.00E+01	0.0E+00
Total			1.8E+06

Table 12.2-3: Primary Coolant at the Core Exit Gamma Source Term

Energy Group	Energy (MeV)		DB PCA Core Exit N-16
	Lower Bound	Upper Bound	(γ/s/g)
1	1.00E-02	2.00E-02	4.1E+05
2	2.00E-02	3.00E-02	2.4E+05
3	3.00E-02	4.50E-02	3.3E+05
4	4.50E-02	6.00E-02	1.6E+05
5	6.00E-02	7.00E-02	7.7E+04
6	7.00E-02	7.50E-02	3.4E+04
7	7.50E-02	1.00E-01	2.6E+05
8	1.00E-01	1.50E-01	1.6E+05
9	1.50E-01	2.00E-01	1.3E+05
10	2.00E-01	2.60E-01	8.4E+04
11	2.60E-01	3.00E-01	3.7E+04
12	3.00E-01	4.00E-01	1.0E+05
13	4.00E-01	4.50E-01	3.6E+04
14	4.50E-01	5.10E-01	3.9E+04
15	5.10E-01	5.12E-01	5.4E+02
16	5.12E-01	6.00E-01	2.3E+04
17	6.00E-01	7.00E-01	3.5E+04
18	7.00E-01	8.00E-01	2.7E+04
19	8.00E-01	9.00E-01	2.1E+04
20	9.00E-01	1.00E+00	1.6E+04
21	1.00E+00	1.20E+00	2.5E+04
22	1.20E+00	1.33E+00	2.2E+04
23	1.33E+00	1.44E+00	8.5E+03
24	1.44E+00	1.50E+00	3.3E+03
25	1.50E+00	1.57E+00	3.3E+03
26	1.57E+00	1.66E+00	5.4E+03
27	1.66E+00	1.80E+00	1.6E+04
28	1.80E+00	2.00E+00	1.3E+04
29	2.00E+00	2.15E+00	4.2E+03
30	2.15E+00	2.35E+00	4.7E+02
31	2.35E+00	2.50E+00	1.0E+04
32	2.50E+00	2.75E+00	6.2E+04
33	2.75E+00	3.00E+00	1.6E+04
34	3.00E+00	3.50E+00	4.2E+03
35	3.50E+00	4.00E+00	4.8E+03
36	4.00E+00	4.50E+00	1.7E+03
37	4.50E+00	5.00E+00	1.6E+03
38	5.00E+00	5.50E+00	1.5E+03
39	5.50E+00	6.00E+00	7.2E+02
40	6.00E+00	6.50E+00	4.7E+06
41	6.50E+00	7.00E+00	3.1E+03
42	7.00E+00	7.50E+00	3.5E+05
43	7.50E+00	8.00E+00	9.9E+01
44	8.00E+00	1.00E+01	5.5E+03
45	1.00E+01	1.20E+01	1.4E+00
46	1.20E+01	1.40E+01	0.0E+00

Table 12.2-3: Primary Coolant at the Core Exit Gamma Source Term (Continued)

Energy Group	Energy (MeV)		DB PCA Core Exit N-16
	Lower Bound	Upper Bound	(γ/s/g)
47	1.40E+01	2.00E+01	0.0E+00
Total			**7.5E+06**

Table 12.2-4: Nitrogen-16 Primary Coolant Concentrations at Full Power

Primary Coolant Location	µCi/g
Core exit	2.0E+02
Top of upper riser / entrance to SG	4.1E+01
CVCS letdown line	2.4E+00
CVCS heat exchanger	7.7E-01

Table 12.2-5: Chemical and Volume Control System Component Source Term Inputs and Assumptions

Model Parameter	Value
CVCS mixed bed:	-
CVCS mixed bed operation time	2 years
Decontamination Factors	Table 11.1-2
Geometry	Vertical Cylinder
Source dimensions of vessel	diameter=21.6 in, Height=95 in
Shielding thickness of steel shell	1.22 in
CVCS Cation bed:	-
CVCS Cation Bed Operation Time	1 year
CVCS Cation Bed Decontamination Factors:	-
Halogens	1
Cs, Rb	10
Others	10
Geometry	Vertical Cylinder
Source dimensions of vessel	diameter=21.6 in, Height=95 in
Shielding thickness of steel shell	1.22 in
Regenerative and Non-Regenerative Heat Exchangers:	-
Contents	100% Primary Coolant
Geometry	-
Regenerative heat exchanger	3x2 array of horizontal Cylinders
Source dimensions of each cylinder	diameter=12.8 in, Height=98.8 in
Shielding thickness of steel shell	1.6 in
Non-Regenerative heat exchanger	1x2 array of horizontal Cylinders
Source dimensions of each cylinder	diameter=11.5 in, Height=66.5 in
Shielding thickness of steel shell	1.25 in
Module Heating System Heater:	-
Contents	100% Primary Coolant No-N16
Geometry	Horizontal Cylinder
Source dimensions	diameter=20.4 in, Height=142.4 in
Shielding thickness of steel shell	1.81 in
CVCS flowrate	180 lbm/min
CVCS filter efficiency	9.1% (DF 1.1)
Geometry	Vertical Cylinder
Dimensions	diameter=10.1 in, Height= 65 in
Shielding thickness of steel shell	1.31 in
CVCS resin transfer line:	-
Pipe internal diameter	1.69 in
Pipe wall thickness	0.344 in
Pipe length	36.4 ft
Resin source term	CVCS MB Resin (48 hr Decay)
CVCS pipe inside vertical pipe chase	-
Contents	100% Primary Coolant with N-16 concentration of CVCS letdown line.
Geometry	Vertical Cylinder
Source length of pipe	43.5 ft
Shielding dimensions of pipe	diameter=2.13 in, Thickness= 0.375 in

Table 12.2-6: Chemical and Volume Control System Component Source Terms - Radionuclide Content

Radionuclide	CVCS Mixed Bed IX	CVCS Cation Bed IX	CVCS Particulate Filter	CVCS Mixed Bed Transferred - 48 hour decay
	(Ci)	(Ci)	(Ci)	(Ci)
Kr-83m	2.1E-02	-	-	2.0E-08
Kr-85m	2.3E-05	-	-	3.5E-09
Kr-85	6.2E-09	-	-	1.6E-09
Kr-87	-	-	-	-
Kr-88	-	-	-	-
Kr-89	-	-	-	-
Xe-131m	7.0E-01	-	-	1.7E-01
Xe-133m	2.8E-01	-	-	5.3E-02
Xe-133	9.5E+00	-	-	2.1E+00
Xe-135m	3.2E-01	-	-	5.3E-04
Xe-135	1.6E+00	-	-	3.2E-02
Xe-137	-	-	-	-
Xe-138	-	-	-	-
Br-82	5.3E-02	-	-	5.1E-03
Br-83	2.1E-02	-	-	4.9E-09
Br-84	2.1E-03	-	-	2.9E-31
Br-85	2.3E-05	-	-	-
I-129	3.0E-04	1.4E-10	-	7.5E-05
I-130	1.5E-01	-	-	2.6E-03
I-131	6.0E+01	9.4E-04	-	1.3E+01
I-132	2.8E+00	2.2E-02	-	4.1E-01
I-133	9.8E+00	1.9E-05	-	5.0E-01
I-134	7.7E-02	2.5E-05	-	6.7E-19
I-135	2.0E+00	-	-	3.1E-03
Rb-86m	3.1E-09	1.4E-09	-	-
Rb-86	4.8E-01	2.1E-01	-	1.1E-01
Rb-88	5.4E-02	2.4E-02	-	2.3E-51
Rb-89	2.1E-03	9.5E-04	-	3.1E-61
Cs-132	3.3E-03	1.5E-03	-	6.7E-04
Cs-134	1.4E+03	3.5E+02	-	3.4E+02
Cs-135m	1.1E-04	5.0E-05	-	1.2E-21
Cs-136	1.0E+01	4.7E+00	-	2.4E+00
Cs-137	9.3E+02	2.1E+02	-	2.3E+02
Cs-138	3.8E-02	1.7E-02	-	1.1E-28
P-32	2.0E-06	1.8E-08	-	4.5E-07
Co-57	2.4E-07	1.6E-09	-	6.1E-08
Sr-89	3.2E-01	3.9E-03	-	7.8E-02
Sr-90	4.9E-01	2.3E-03	-	1.2E-01
Sr-91	1.3E-03	1.2E-05	-	1.0E-05
Sr-92	2.0E-04	1.8E-06	-	1.8E-10
Y-90	4.9E-01	2.3E-03	-	1.2E-01
Y-91m	8.3E-04	7.7E-06	-	6.7E-06
Y-91	5.5E-02	5.0E-04	-	1.3E-02

Table 12.2-6: Chemical and Volume Control System Component Source Terms - Radionuclide Content (Continued)

Radionuclide	CVCS Mixed Bed IX	CVCS Cation Bed IX	CVCS Particulate Filter	CVCS Mixed Bed Transferred - 48 hour decay
	(Ci)	(Ci)	(Ci)	(Ci)
Y-92	4.2E-04	3.9E-06	-	2.0E-08
Y-93	3.0E-04	2.8E-06	-	2.9E-06
Zr-97	7.3E-04	6.7E-06	-	2.5E-05
Nb-95	1.1E+00	9.5E-03	-	1.0E+00
Mo-99	5.2E+00	4.8E-02	-	7.8E-01
Mo-101	7.3E-04	6.7E-06	-	8.4E-64
Tc-99m	5.0E+00	4.6E-02	-	7.6E-01
Tc-99	1.8E-02	8.3E-05	-	4.5E-03
Ru-103	7.1E-02	6.5E-04	-	1.7E-02
Ru-105	1.1E-04	1.0E-06	-	1.5E-08
Ru-106	3.1E-01	1.9E-03	-	7.8E-02
Rh-103m	7.0E-02	6.4E-04	-	1.7E-02
Rh-105	1.9E-03	1.8E-05	-	1.9E-04
Rh-106	3.1E-01	1.9E-03	-	7.8E-02
Ag-110	1.6E-01	1.1E-03	-	1.6E-01
Sb-124	1.6E-04	1.5E-06	-	4.0E-05
Sb-125	8.0E-03	4.1E-05	-	2.0E-03
Sb-127	3.9E-04	3.6E-06	-	6.8E-05
Sb-129	2.3E-05	2.1E-07	-	3.0E-09
Te-125m	1.7E-01	1.5E-03	-	4.1E-02
Te-127m	1.2E+00	1.0E-02	-	2.9E-01
Te-127	1.2E+00	9.9E-03	-	2.9E-01
Te-129m	1.1E+00	9.8E-03	-	2.6E-01
Te-129	6.7E-01	6.2E-03	-	1.6E-01
Te-131m	1.3E-01	1.2E-03	-	1.1E-02
Te-131	3.0E-02	2.7E-04	-	2.4E-03
Te-132	2.4E+00	2.2E-02	-	3.9E-01
Te-133m	2.5E-03	2.3E-05	-	1.4E-19
Te-134	2.7E-03	2.5E-05	-	1.2E-24
Ba-137m	8.8E+02	2.0E+02	-	2.2E+02
Ba-139	9.9E-05	9.1E-07	-	9.1E-16
Ba-140	1.2E-01	1.1E-03	-	2.7E-02
La-140	1.2E-01	1.1E-03	-	2.9E-02
La-141	8.7E-05	8.0E-07	-	4.5E-09
La-142	1.6E-05	1.5E-07	-	1.2E-15
Ce-141	4.7E-02	4.3E-04	-	1.1E-02
Ce-143	1.5E-03	1.4E-05	-	1.4E-04
Ce-144	2.9E-01	1.9E-03	-	7.1E-02
Pr-143	1.9E-02	1.7E-04	-	4.2E-03
Pr-144	2.8E-01	1.8E-03	-	7.1E-02
Np-239	5.4E-02	4.9E-04	-	7.4E-03
Na-24	1.4E+00	1.3E-02	2.6E-03	1.6E-01
Cr-51	3.6E+00	3.3E-02	6.6E-03	3.4E+00

Table 12.2-6: Chemical and Volume Control System Component Source Terms - Radionuclide Content (Continued)

Radionuclide	CVCS Mixed Bed IX	CVCS Cation Bed IX	CVCS Particulate Filter	CVCS Mixed Bed Transferred - 48 hour decay
	(Ci)	(Ci)	(Ci)	(Ci)
Mn-54	1.7E+01	1.1E-01	2.1E-02	1.6E+01
Fe-55	2.0E+01	1.0E-01	2.1E-02	2.0E+01
Fe-59	5.5E-01	5.1E-03	1.0E-03	5.4E-01
Co-58	1.3E+01	1.2E-01	2.4E-02	1.3E+01
Co-60	9.7E+00	4.8E-02	9.6E-03	9.7E+00
Ni-63	5.5E+00	2.5E-02	5.1E-03	5.5E+00
Zn-65	4.5E+00	3.0E-02	6.2E-03	4.5E+00
Zr-95	1.0E+00	9.3E-03	1.9E-03	1.0E+00
Ag-110m	1.2E+01	7.8E-02	1.6E-02	1.2E+01
W-187	1.2E-01	1.1E-03	2.1E-04	2.8E-02
H-3	-	-	-	-
C-14	-	-	-	-
N-16	-	-	-	-
Ar-41	-	-	-	-
				9.0E+02

Table 12.2-7: Chemical and Volume Control System Component Source Terms - Source Strengths

Energy Group	Energy (MeV)		Design Basis Primary Coolant at CVCS Pipe Chase Photon Spectra N-16	CVCS Mixed-Bed	CVCS Cation Bed	CVCS Particulate Filter	CVCS Mixed Bed Transfer - 48 hour decay	CVCS Letdown - 71.3 second decay
	Lower Bound	Upper Bound	(photon/sec/ gram)	photons/sec	photons/sec	photons/sec	photons/sec	photons/sec
1	1.00E-02	2.00E-02	7.0E+03	2.7E+11	5.7E+10	4.2E+06	6.5E+10	5.3E+02
2	2.00E-02	3.00E-02	7.5E+03	3.2E+11	3.1E+10	8.5E+06	7.4E+10	1.1E+03
3	3.00E-02	4.50E-02	1.4E+05	2.8E+12	6.0E+11	1.6E+06	6.8E+11	3.4E+04
4	4.50E-02	6.00E-02	2.4E+03	8.0E+10	1.5E+10	1.5E+06	1.9E+10	1.3E+02
5	6.00E-02	7.00E-02	1.2E+03	5.0E+10	1.5E+10	2.3E+06	1.2E+10	6.1E+01
6	7.00E-02	7.50E-02	5.0E+02	1.3E+10	2.9E+09	1.4E+06	3.3E+09	2.4E+01
7	7.50E-02	1.00E-01	1.2E+05	2.4E+11	1.9E+10	8.7E+05	5.5E+10	2.8E+04
8	1.00E-01	1.50E-01	3.2E+03	2.4E+11	1.1E+10	2.8E+06	4.2E+10	2.2E+02
9	1.50E-01	2.00E-01	3.1E+03	1.0E+11	3.2E+10	1.8E+06	2.3E+10	3.9E+02
10	2.00E-01	2.60E-01	1.0E+04	1.7E+11	6.5E+09	1.2E+06	2.1E+10	2.3E+03
11	2.60E-01	3.00E-01	8.0E+02	1.9E+11	2.0E+10	3.6E+05	4.0E+10	3.5E+01
12	3.00E-01	4.00E-01	3.2E+03	2.1E+12	7.6E+10	2.4E+07	4.5E+11	1.1E+02
13	4.00E-01	4.50E-01	1.1E+03	2.9E+10	3.1E+08	2.2E+07	1.7E+10	1.2E+02
14	4.50E-01	5.10E-01	8.7E+02	7.6E+11	1.9E+11	2.4E+06	1.9E+11	4.7E+01
15	5.10E-01	5.12E-01	5.4E+02	1.6E+11	1.4E+09	2.7E+08	1.5E+11	1.2E+02
16	5.12E-01	6.00E-01	4.0E+03	1.2E+13	3.2E+12	7.6E+05	3.0E+12	2.5E+02
17	6.00E-01	7.00E-01	4.1E+03	7.6E+13	1.9E+13	6.8E+08	1.9E+13	2.8E+02
18	7.00E-01	8.00E-01	2.8E+03	4.8E+13	1.3E+13	3.2E+08	1.2E+13	2.0E+02
19	8.00E-01	9.00E-01	2.3E+03	3.9E+12	7.0E+11	2.1E+09	2.0E+12	1.6E+02
20	9.00E-01	1.00E+00	6.4E+02	1.7E+11	1.2E+09	2.0E+08	1.5E+11	3.2E+01
21	1.00E+00	1.20E+00	1.7E+03	2.2E+12	4.9E+11	5.2E+08	8.9E+11	9.9E+01
22	1.20E+00	1.33E+00	8.9E+03	3.2E+11	3.5E+10	2.0E+08	2.2E+11	7.9E+03
23	1.33E+00	1.44E+00	1.4E+03	1.8E+12	3.9E+11	4.1E+08	6.6E+11	1.0E+02
24	1.44E+00	1.50E+00	2.2E+02	2.7E+10	1.4E+08	2.5E+07	1.8E+10	3.0E+00
25	1.50E+00	1.57E+00	3.4E+02	6.5E+10	6.3E+08	8.6E+07	6.3E+10	6.4E+01
26	1.57E+00	1.66E+00	1.1E+02	4.8E+09	4.3E+07	1.4E+05	1.1E+09	4.2E+00

Table 12.2-7: Chemical and Volume Control System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy (MeV)		Design Basis Primary Coolant at CVCS Pipe Chase Photon Spectra N-16	CVCS Mixed-Bed	CVCS Cation Bed	CVCS Particulate Filter	CVCS Mixed Bed Transfer - 48 hour decay	CVCS Letdown - 71.3 second decay
	Lower Bound	Upper Bound	(photon/sec/gram)	photons/sec	photons/sec	photons/sec	photons/sec	photons/sec
27	1.66E+00	1.80E+00	6.7E+02	1.9E+10	3.4E+07	4.6E+06	2.6E+09	3.4E+01
28	1.80E+00	2.00E+00	6.5E+02	3.5E+09	2.2E+08	9.9E+04	3.8E+08	6.0E+01
29	2.00E+00	2.15E+00	3.5E+02	1.6E+09	1.6E+07	5.9E+03	1.3E+08	6.7E+01
30	2.15E+00	2.35E+00	4.7E+02	1.2E+09	1.0E+08	4.1E+03	8.5E+07	9.9E+01
31	2.35E+00	2.50E+00	8.0E+02	9.9E+08	3.4E+06	4.4E+00	3.8E+07	1.7E+02
32	2.50E+00	2.75E+00	1.2E+03	2.8E+10	3.3E+08	5.1E+07	3.1E+09	4.4E+01
33	2.75E+00	3.00E+00	4.4E+02	2.5E+10	2.3E+08	4.7E+07	2.7E+09	7.2E+00
34	3.00E+00	3.50E+00	7.0E+01	2.4E+07	7.5E+06	9.0E-02	3.5E+05	3.0E+00
35	3.50E+00	4.00E+00	6.4E+01	4.9E+07	1.6E+06	7.5E+04	4.4E+06	7.7E-01
36	4.00E+00	4.50E+00	2.2E+01	1.3E+06	2.8E+05	8.2E+02	4.8E+04	1.3E-01
37	4.50E+00	5.00E+00	2.3E+01	3.8E+06	1.7E+06	-	-	4.9E-01
38	5.00E+00	5.50E+00	1.8E+01	7.1E+03	3.2E+03	-	-	6.5E-03
39	5.50E+00	6.00E+00	8.7E+00	-	-	-	-	2.8E-03
40	6.00E+00	6.50E+00	5.7E+04	-	-	-	-	1.8E+01
41	6.50E+00	7.00E+00	3.7E+01	-	-	-	-	1.2E-02
42	7.00E+00	7.50E+00	4.2E+03	-	-	-	-	1.3E+00
43	7.50E+00	8.00E+00	1.2E+00	-	-	-	-	3.8E-04
44	8.00E+00	1.00E+01	6.7E+01	-	-	-	-	2.1E-02
45	1.00E+01	1.20E+01	1.7E-02	-	-	-	-	5.5E-06
46	1.20E+01	1.40E+01	-	-	-	-	-	-
47	1.40E+01	2.00E+01	-	-	-	-	-	-
Total			**3.9E+05**	**1.5E+14**	**3.7E+13**	**5.0E+09**	**4.0E+13**	**7.7E+04**

Table 12.2-8: Pool Cooling and Cleanup System Component Source Term Inputs and Assumptions

Model Parameter	Value
Pool cooling heat exchangers:	-
Contents	100% Pool Water
Source term mass of each heat exchanger	2.77E+06 g
Geometry	2x1 array of Horizontal Cylinders
Source dimensions of each cylinder	diameter=27 in, Height=299 in
Shielding thickness of steel shell	0.5 in
Pool cleanup system demineralizer:	-
Geometry	Vertical Cylinder
Source dimensions	diameter=130 in, Height=98 in
Shielding thickness of steel shell	0.25 in
Operation time	2 years
PCWS demineralizer decontamination factors	Halogens 100 Cs/Rb 2 Others 50
PCWS filters:	-
PCWS filter efficiency	9.1%
Geometry	Horizontal Cylinder
Source dimensions	diameter=38.9 in, Height=118 in
Shielding thickness of steel shell	0.25 in
PCWS flow rate	164 gpm
UHS water mass	1.18E+10 grams
PCWS filter operation time	1 years
Pool Surge Control Tank	-
Contents	Cleanup Up Pool Water
Geometry	Vertical Cylinder
Source dimensions	diameter=50 ft, Height=45 ft
Shielding thickness of steel wall	0.25 in
Source volume	8.8E+04 ft^3
Source Mass	2.5E+09 g

Note: Assumes the plant consists of 6 NPMs on an 18-month refueling cycle.

Table 12.2-9: Pool Cooling and Cleanup System Component Source Terms - Radionuclide Content

Radionuclide	PCWS Demineralizers	Reactor Pool Water	PCWS Surge Tank	PCWS Particulate Filter
	(Ci)	(Ci/g)	(Ci/g)	(Ci)
Kr-83m	9.6E-13	-	-	-
Kr-85m	-	-	-	-
Kr-85	-	-	-	-
Kr-87	-	-	-	-
Kr-88	-	-	-	-
Kr-89	-	-	-	-
Xe-131m	2.1E-02	-	-	-
Xe-133m	5.0E-04	-	-	-
Xe-133	1.7E-02	-	-	-
Xe-135m	3.6E-07	-	-	-
Xe-135	1.8E-06	-	-	-
Xe-137	-	-	-	-
Xe-138	-	-	-	-
Br-82	6.7E-06	2.4E-14	2.4E-16	-
Br-83	9.6E-13	5.2E-20	5.2E-22	-
Br-84	4.5E-38	1.1E-44	1.1E-46	-
Br-85	-	-	-	-
I-129	1.1E-07	1.2E-18	1.2E-20	-
I-130	2.6E-06	2.7E-14	2.7E-16	-
I-131	1.8E+00	1.2E-09	1.2E-11	-
I-132	5.9E-04	9.8E-13	9.8E-15	-
I-133	1.7E-02	1.1E-10	1.1E-12	-
I-134	3.9E-24	5.7E-31	5.7E-33	-
I-135	2.2E-06	4.3E-14	4.3E-16	-
Rb-86m	-	-	-	-
Rb-86	1.0E-03	3.1E-13	1.6E-13	-
Rb-88	1.7E-60	7.8E-67	3.9E-67	-
Rb-89	1.4E-71	7.1E-78	3.6E-78	-
Cs-132	6.2E-06	5.3E-15	2.6E-15	-
Cs-134	3.2E+00	4.8E-11	2.4E-11	-
Cs-135m	4.8E-26	7.3E-33	3.6E-33	-
Cs-136	2.2E-02	9.3E-12	4.7E-12	-
Cs-137	2.2E+00	2.5E-11	1.2E-11	-
Cs-138	1.7E-34	4.0E-41	2.0E-41	-
P-32	6.9E-10	2.6E-19	5.2E-21	-
Co-57	9.3E-11	2.2E-21	4.4E-23	-
Sr-89	1.2E-04	1.3E-14	2.6E-16	-
Sr-90	1.9E-04	2.0E-15	4.1E-17	-
Sr-91	9.7E-09	1.3E-16	2.6E-18	-
Sr-92	4.5E-14	2.2E-21	4.4E-23	-
Y-90	1.9E-04	1.2E-15	2.4E-17	-
Y-91m	6.2E-09	8.3E-17	1.7E-18	-
Y-91	2.1E-05	1.9E-15	3.8E-17	-
Y-92	8.1E-12	2.9E-19	5.8E-21	-

Table 12.2-9: Pool Cooling and Cleanup System Component Source Terms - Radionuclide Content (Continued)

Radionuclide	PCWS Demineralizers	Reactor Pool Water	PCWS Surge Tank	PCWS Particulate Filter
	(Ci)	(Ci/g)	(Ci/g)	(Ci)
Y-93	2.7E-09	3.5E-17	6.9E-19	-
Zr-97	2.9E-08	2.2E-16	4.5E-18	-
Nb-95	3.5E-01	1.4E-12	2.9E-14	-
Mo-99	1.1E-03	2.2E-12	4.4E-14	-
Mo-101	2.9E-75	1.5E-81	3.0E-83	-
Tc-99m	1.1E-03	2.1E-12	4.2E-14	-
Tc-99	7.0E-06	7.4E-17	1.5E-18	-
Ru-103	2.6E-05	3.6E-15	7.1E-17	-
Ru-105	7.9E-12	2.3E-19	4.6E-21	-
Ru-106	1.2E-04	2.3E-15	4.7E-17	-
Rh-103m	2.6E-05	3.5E-15	7.1E-17	-
Rh-105	2.5E-07	9.2E-16	1.8E-17	-
Rh-106	1.2E-04	2.3E-15	4.7E-17	-
Ag-110	5.4E-02	1.5E-12	3.0E-14	-
Sb-124	6.1E-08	5.4E-18	1.1E-19	-
Sb-125	3.1E-06	4.1E-17	8.3E-19	-
Sb-127	9.9E-08	1.4E-16	2.8E-18	-
Sb-129	1.5E-12	4.5E-20	9.0E-22	-
Te-125m	6.3E-05	5.8E-15	1.2E-16	-
Te-127m	4.5E-04	2.2E-14	4.5E-16	-
Te-127	4.4E-04	2.3E-14	4.7E-16	-
Te-129m	3.9E-04	6.3E-14	1.3E-15	-
Te-129	2.5E-04	3.9E-14	7.9E-16	-
Te-131m	1.4E-05	6.0E-14	1.2E-15	-
Te-131	3.2E-06	1.4E-14	2.7E-16	-
Te-132	5.7E-04	9.5E-13	1.9E-14	-
Te-133m	1.1E-24	1.6E-31	3.1E-33	-
Te-134	1.8E-30	3.3E-37	6.5E-39	-
Ba-137m	2.1E+00	2.3E-11	4.7E-13	-
Ba-139	4.2E-20	3.9E-27	7.8E-29	-
Ba-140	4.0E-05	1.7E-14	3.4E-16	-
La-140	4.4E-05	1.3E-14	2.6E-16	-
La-141	2.0E-12	6.6E-20	1.3E-21	-
La-142	7.7E-20	6.5E-27	1.3E-28	-
Ce-141	1.7E-05	2.8E-15	5.6E-17	-
Ce-143	1.8E-07	7.0E-16	1.4E-17	-
Ce-144	1.1E-04	2.5E-15	5.0E-17	-
Pr-143	6.4E-06	2.5E-15	4.9E-17	-
Pr-144	1.1E-04	2.5E-15	4.9E-17	-
Np-239	1.0E-05	2.4E-14	4.8E-16	-
Na-24	3.9E-05	3.7E-13	7.4E-15	4.3E-06
Cr-51	1.2E+00	2.5E-10	5.0E-12	1.3E-01
Mn-54	5.7E+00	1.4E-10	2.7E-12	4.4E-01
Fe-55	6.8E+00	1.0E-10	2.0E-12	4.3E-01

Table 12.2-9: Pool Cooling and Cleanup System Component Source Terms - Radionuclide Content (Continued)

Radionuclide	PCWS Demineralizers	Reactor Pool Water	PCWS Surge Tank	PCWS Particulate Filter
	(Ci)	(Ci/g)	(Ci/g)	(Ci)
Fe-59	1.8E-01	2.5E-11	4.9E-13	2.1E-02
Co-58	4.5E+01	3.8E-09	7.7E-11	5.0E+00
Co-60	3.4E+00	4.5E-11	9.0E-13	2.0E-01
Ni-63	1.9E+00	2.3E-11	4.5E-13	1.1E-01
Zn-65	1.5E+00	4.3E-11	8.6E-13	1.3E-01
Zr-95	3.5E-01	3.2E-11	6.5E-13	3.8E-02
Ag-110m	4.0E+00	1.1E-10	2.2E-12	3.3E-01
W-187	8.0E-03	4.8E-11	9.6E-13	9.0E-04
H-3	-	1.2E-07	1.2E-07	-
C-14	-	1.0E-12	1.0E-12	-
N-16	-	-	-	-
Ar-41	-	-	-	-

Table 12.2-10: Pool Cooling and Cleanup System Component Source Terms - Source Strengths

Energy Group	Energy (MeV)		PCWS Demineralizers	Reactor Pool Water	PCWS Surge Tank	PCWS Particulate Filter
	Lower Bound	Upper Bound	(γ/s)	(γ/s)	(γ/s)	(γ/s)
1	1.00E-02	2.00E-02	4.2E+09	6.0E+09	1.6E-02	3.6E+08
2	2.00E-02	3.00E-02	6.3E+09	2.1E+10	2.2E-02	3.3E+08
3	3.00E-02	4.50E-02	9.3E+09	1.3E+10	3.2E-02	1.5E+08
4	4.50E-02	6.00E-02	1.3E+09	2.7E+09	5.1E-03	1.2E+08
5	6.00E-02	7.00E-02	7.2E+08	5.7E+09	1.8E-02	6.2E+07
6	7.00E-02	7.50E-02	3.0E+08	3.4E+09	5.8E-03	2.8E+07
7	7.50E-02	1.00E-01	2.9E+09	1.4E+10	2.2E-02	9.3E+07
8	1.00E-01	1.50E-01	1.3E+09	4.8E+09	8.7E-03	1.2E+08
9	1.50E-01	2.00E-01	1.2E+09	3.3E+09	3.1E-02	8.9E+07
10	2.00E-01	2.60E-01	6.2E+08	9.7E+08	2.0E-03	5.2E+07
11	2.60E-01	3.00E-01	4.5E+09	3.4E+10	4.8E-02	2.1E+07
12	3.00E-01	4.00E-01	6.2E+10	4.7E+11	4.7E-01	5.0E+08
13	4.00E-01	4.50E-01	5.8E+09	2.5E+09	3.7E-03	4.8E+08
14	4.50E-01	5.10E-01	2.2E+09	8.1E+09	2.6E-02	2.2E+07
15	5.10E-01	5.12E-01	5.0E+11	5.0E+11	8.5E-01	5.5E+10
16	5.12E-01	6.00E-01	3.0E+10	4.6E+10	2.5E-01	1.0E+07
17	6.00E-01	7.00E-01	3.6E+11	1.3E+11	9.7E-01	1.4E+10
18	7.00E-01	8.00E-01	2.1E+11	6.7E+10	9.3E-01	6.7E+09
19	8.00E-01	9.00E-01	1.9E+12	1.7E+12	3.1E+00	2.0E+11
20	9.00E-01	1.00E+00	5.0E+10	1.7E+10	2.8E-02	4.1E+09
21	1.00E+00	1.20E+00	1.7E+11	4.0E+10	2.0E-01	1.1E+10
22	1.20E+00	1.33E+00	6.9E+10	1.8E+10	6.1E-02	4.3E+09
23	1.33E+00	1.44E+00	1.0E+11	2.2E+10	6.3E-02	6.7E+09
24	1.44E+00	1.50E+00	6.2E+09	2.0E+09	3.4E-03	5.1E+08
25	1.50E+00	1.57E+00	2.2E+10	7.0E+09	1.2E-02	1.8E+09
26	1.57E+00	1.66E+00	3.8E+07	1.9E+07	3.1E-05	3.0E+06
27	1.66E+00	1.80E+00	8.4E+09	8.4E+09	1.4E-02	9.3E+08
28	1.80E+00	2.00E+00	2.6E+07	1.7E+07	2.2E-05	2.1E+06
29	2.00E+00	2.15E+00	1.7E+06	4.4E+06	4.3E-06	1.2E+05
30	2.15E+00	2.35E+00	1.5E+06	2.1E+06	2.2E-06	8.6E+04
31	2.35E+00	2.50E+00	8.6E+04	1.1E+06	9.6E-07	7.7E+01
32	2.50E+00	2.75E+00	8.2E+05	8.5E+07	1.4E-04	8.4E+04
33	2.75E+00	3.00E+00	6.8E+05	7.7E+07	1.3E-04	7.7E+04
34	3.00E+00	3.50E+00	5.3E+02	1.6E+03	2.7E-09	1.5E-04
35	3.50E+00	4.00E+00	1.1E+03	1.2E+05	2.1E-07	1.2E+02
36	4.00E+00	4.50E+00	1.2E+01	1.4E+03	2.3E-09	1.3E+00
37	4.50E+00	5.00E+00	-	-	-	-
38	5.00E+00	5.50E+00	-	-	-	-
39	5.50E+00	6.00E+00	-	-	-	-
40	6.00E+00	6.50E+00	-	-	-	-
41	6.50E+00	7.00E+00	-	-	-	-
42	7.00E+00	7.50E+00	-	-	-	-
43	7.50E+00	8.00E+00	-	-	-	-

Table 12.2-10: Pool Cooling and Cleanup System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy (MeV)		PCWS Demineralizers	Reactor Pool Water	PCWS Surge Tank	PCWS Particulate Filter
	Lower Bound	Upper Bound	(γ/s)	(γ/s)	(γ/s)	(γ/s)
44	8.00E+00	1.00E+01	-	-	-	-
45	1.00E+01	1.20E+01	-	-	-	-
46	1.20E+01	1.40E+01	-	-	-	-
47	1.40E+01	2.00E+01	-	-	-	-
Total			**3.6E+12**	**3.2E+12**	**7.2E+00**	**3.1E+11**

Table 12.2-11: Liquid Radioactive Waste System Component Source Term Inputs and Assumptions

Model Parameter	Value
LRWS degasifier	-
Contents	CVCS Letdown
Geometry	Vertical Cylinder
Source dimensions	Diameter=138 in, Height=193 in
Shield thickness of steel shell	1.75 in
Volume	12500 Gallons
LCW and HCW collection tanks	-
Inputs	Table 11.2-3
Geometry	Vertical Cylinder
Source dimensions	Diameter=12 ft, Height=21 ft
Shield thickness of steel shell	0.25 in
Volume	14400 gallons
LRWS oil separator	-
Inputs	Table 11.2-3
Geometry	Horizontal Cylinder
Source dimensions	Diameter=51 in, Height=118 in
Shield thickness of steel shell	0.25 in
LCW and HCW granulated activated charcoal (GAC) units	-
Decontamination Factors	-
Cr-51	256
Mn-54	107
Co-58	13.2
Co-60	6.7
Ag-110m	3250
Antimony	7.1
Nb-95	639
Geometry	Vertical Cylinder
Source dimensions of vessel	Diameter=71 in, Height=71 in
Shield thickness of steel shell	0.25 in
LCW reverse osmosis (RO) unit	-
Decontamination factors	-
All nuclides	10
Geometry	Horizontal Cylinder
Source dimensions	Diameter=59 in, Length=102 in
Shield thickness of steel shell	0.25 in
LCW and HCW sample tanks	-
Geometry	Vertical Cylinder
Source dimensions	Diameter=12 ft, Height=21 ft
Shield thickness of steel shell	0.25 in
Drum dryer and holdup tank	-
Inputs	RO Rejects
Geometry	Vertical Cylinder
Source dimensions	Diameter=65.0 in, Height=118 in
Shield thickness of steel shell	0.25 in
LCW processing skid	-

Table 12.2-11: Liquid Radioactive Waste System Component Source Term Inputs and Assumptions (Continued)

Model Parameter	Value
Inputs	LCW Filters, Ixs, Accumulators, and Polishers total Accumulation (LCW Processing Skid)
Geometry	Horizontal Cylinder
Source Dimensions	Diameter=71 in, Length=299 in
Shield thickness of steel shell	0.25 in
Additional Shielding	1 in Steel

Note: Assumes the plant consists of 6 NPMs operating on an 18-month refueling cycle.

Table 12.2-12a: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content

Isotope	LCW Collection Tank (Ci/Tank)	HCW Collection Tank (Ci/Tank)	LCW Sample Tank (Ci/Tank)	HCW Sample Tank (Ci/Tank)
Kr83m	-	-	-	-
Kr85m	-	-	-	-
Kr85	-	-	-	-
Kr87	-	-	-	-
Kr88	-	-	-	-
Kr89	-	-	-	-
Xe131m	-	-	-	-
Xe133m	-	-	-	-
Xe133	-	-	-	-
Xe135m	-	-	-	-
Xe135	-	-	-	-
Xe137	-	-	-	-
Xe138	-	-	-	-
Br82	3.9E-05	3.6E-05	1.9E-13	9.4E-11
Br83	2.3E-04	2.1E-04	-	-
Br84	1.1E-04	9.6E-05	-	-
Br85	1.3E-05	1.2E-05	-	-
I129	6.5E-10	6.0E-10	6.5E-14	6.0E-14
I130	3.2E-04	3.0E-04	1.6E-20	8.8E-13
I131	8.2E-03	1.3E-02	1.3E-07	6.7E-07
I132	3.8E-03	3.5E-03	2.1E-09	1.5E-08
I133	1.2E-02	1.2E-02	6.0E-14	2.4E-09
I134	2.2E-03	2.0E-03	-	-
I135	7.9E-03	7.2E-03	-	2.2E-15
Rb86m	3.1E-07	2.3E-08	-	-
Rb86	1.8E-03	1.3E-04	4.8E-07	5.9E-08
Rb88	3.0E-01	2.2E-02	-	-
Rb89	1.4E-02	1.0E-03	-	-
Cs132	3.6E-05	2.6E-06	2.2E-09	6.8E-10
Cs134	2.6E-01	1.9E-02	1.5E-04	1.1E-05
Cs135m	2.1E-04	1.6E-05	-	-
Cs136	5.6E-02	4.2E-03	1.1E-05	1.6E-06
Cs137	1.3E-01	9.8E-03	7.7E-05	5.8E-06
Cs138	1.1E-01	8.4E-03	-	-
P32	3.6E-10	1.5E-10	1.3E-14	1.0E-14
Co57	2.7E-12	1.1E-12	2.6E-16	1.1E-16
Sr89	1.6E-05	6.6E-06	2.5E-09	7.1E-10
Sr90	2.6E-06	1.0E-06	2.6E-10	1.0E-10
Sr91	8.6E-06	3.4E-06	-	5.3E-16
Sr92	4.6E-06	1.8E-06	-	-
Y90	6.3E-07	2.5E-07	2.6E-10	9.3E-11
Y91m	4.6E-06	1.8E-06	-	3.4E-16
Y91	2.4E-06	9.6E-07	1.9E-10	8.9E-11
Y92	3.9E-06	1.6E-06	-	-
Y93	1.8E-06	7.3E-07	-	2.3E-16

Table 12.2-12a: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	LCW Collection Tank (Ci/Tank)	HCW Collection Tank (Ci/Tank)	LCW Sample Tank (Ci/Tank)	HCW Sample Tank (Ci/Tank)
Zr97	2.7E-06	1.1E-06	2.2E-19	5.0E-14
Nb95	4.9E-06	2.8E-05	5.3E-09	1.1E-08
Mo99	4.9E-03	2.0E-03	2.4E-09	2.8E-08
Mo101	1.9E-04	7.4E-05	-	-
Tc99m	4.5E-03	1.8E-03	2.3E-09	2.7E-08
Tc99	9.2E-08	3.7E-08	9.3E-12	3.7E-12
Ru103	4.7E-06	1.9E-06	3.2E-10	1.6E-10
Ru105	1.5E-06	6.2E-07	-	-
Ru106	2.9E-06	1.2E-06	2.8E-10	1.2E-10
Rh103m	4.6E-06	1.9E-06	3.2E-10	1.6E-10
Rh105	3.2E-06	1.3E-06	1.7E-14	3.5E-12
Rh106	2.9E-06	1.2E-06	2.8E-10	1.2E-10
Ag110	9.1E-06	3.1E-05	7.1E-10	3.0E-09
Sb124	7.0E-09	2.8E-09	5.5E-13	2.6E-13
Sb125	5.2E-08	2.1E-08	5.1E-12	2.1E-12
Sb127	2.6E-07	1.1E-07	5.9E-13	2.6E-12
Sb129	3.3E-07	1.3E-07	-	-
Te125m	7.6E-06	3.0E-06	5.9E-10	2.8E-10
Te127m	2.9E-05	1.2E-05	2.5E-09	1.1E-09
Te127	1.1E-04	4.5E-05	2.5E-09	1.1E-09
Te129m	8.2E-05	3.3E-05	5.3E-09	2.8E-09
Te129	1.2E-04	4.7E-05	3.4E-09	1.8E-09
Te131m	2.7E-04	1.1E-04	2.3E-13	1.5E-10
Te131	1.3E-04	5.3E-05	5.1E-14	3.3E-11
Te132	2.0E-03	7.9E-04	2.1E-09	1.5E-08
Te133m	1.7E-04	6.7E-05	-	-
Te134	2.4E-04	9.5E-05	-	-
Ba137m	9.0E-03	3.7E-03	7.3E-05	5.4E-06
Ba139	4.5E-06	1.8E-06	-	-
Ba140	2.4E-05	9.7E-06	7.7E-10	6.3E-10
La140	7.0E-06	2.8E-06	8.8E-10	7.0E-10
La141	1.4E-06	5.5E-07	-	-
La142	6.6E-07	2.6E-07	-	-
Ce141	3.7E-06	1.5E-06	2.4E-10	1.3E-10
Ce143	2.8E-06	1.1E-06	7.0E-15	2.3E-12
Ce144	3.1E-06	1.3E-06	3.0E-10	1.2E-10
Pr143	3.3E-06	1.3E-06	1.2E-10	9.7E-11
Pr144	3.1E-06	1.2E-06	3.0E-10	1.2E-10
Np239	5.9E-05	2.4E-05	1.2E-11	2.4E-10
Na24	2.4E-02	9.4E-03	1.6E-16	1.7E-10
Cr51	1.3E-03	5.1E-03	7.9E-08	4.2E-07
Mn54	6.8E-04	2.8E-03	6.5E-08	2.7E-07
Fe55	5.1E-04	2.1E-03	5.1E-08	2.1E-07
Fe59	1.3E-04	5.1E-04	9.3E-09	4.5E-08
Co58	2.0E-03	7.1E-02	1.6E-07	6.6E-06

Table 12.2-12a: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	LCW Collection Tank (Ci/Tank)	HCW Collection Tank (Ci/Tank)	LCW Sample Tank (Ci/Tank)	HCW Sample Tank (Ci/Tank)
Co60	2.3E-04	9.2E-04	2.2E-08	9.2E-08
Ni63	1.1E-04	4.6E-04	1.1E-08	4.6E-08
Zn65	2.2E-04	8.8E-04	2.1E-08	8.6E-08
Zr95	1.7E-04	6.6E-04	1.3E-08	6.1E-08
Ag110m	5.6E-04	2.2E-03	5.2E-08	2.2E-07
W187	1.2E-03	1.4E-03	4.6E-14	6.0E-10
H3	1.2E+02	1.2E+01	1.2E+02	1.2E+01
C14	1.2E-02	7.7E-04	1.2E-02	7.7E-04
N16	-	-	-	-
Ar41	-	-	-	-
Total	**1.2E+02**	**1.2E+01**	**1.2E+02**	**1.2E+01**

Table 12.2-12b: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content

Isotope	LCW Reverse Osmosis Unit	LCW Solids Collection Filter	HCW Granulated Activated Charcoal	Drum Dryer
	(Ci)	(Ci)	(Ci)	(Ci/Drum)
Kr83m	-	-	-	-
Kr85m	-	-	-	-
Kr85	-	-	-	-
Kr87	-	-	-	-
Kr88	-	-	-	-
Kr89	-	-	-	-
Xe131m	-	-	-	-
Xe133m	-	-	-	-
Xe133	-	-	-	-
Xe135m	-	-	-	-
Xe135	-	-	-	-
Xe137	-	-	-	-
Xe138	-	-	-	-
Br82	1.5E-07	-	-	1.5E-07
Br83	5.9E-08	-	-	5.9E-08
Br84	6.1E-09	-	-	6.1E-09
Br85	6.7E-11	-	-	6.7E-11
I129	2.3E-11	-	1.2E-11	4.3E-10
I130	4.3E-07	-	-	4.3E-07
I131	2.2E-04	-	2.5E-05	2.7E-04
I132	5.1E-06	-	5.9E-04	5.2E-06
I133	2.9E-05	-	5.0E-07	2.9E-05
I134	2.2E-07	-	6.5E-07	2.2E-07
I135	5.6E-06	-	-	5.6E-06
Rb86m	1.1E-12	-	-	1.1E-12
Rb86	8.4E-05	-	3.8E-12	1.6E-04
Rb88	1.8E-05	-	-	1.8E-05
Rb89	7.1E-07	-	-	7.1E-07
Cs132	9.8E-07	-	-	1.1E-06
Cs134	1.7E-02	-	-	2.7E-01
Cs135m	3.8E-08	-	-	3.8E-08
Cs136	2.3E-03	-	-	3.6E-03
Cs137	8.6E-03	-	-	1.6E-01
Cs138	1.3E-05	-	-	1.3E-05
P32	4.8E-12	-	-	7.9E-12
Co57	5.5E-14	-	-	6.9E-13
Sr89	4.3E-07	-	2.6E-06	1.8E-06
Sr90	5.3E-08	-	-	9.8E-07
Sr91	5.2E-09	-	-	5.2E-09
Sr92	7.7E-10	-	-	7.7E-10
Y90	2.1E-08	-	1.8E-04	4.3E-07
Y91m	1.6E-09	-	1.9E-07	1.6E-09
Y91	4.4E-08	-	1.5E-07	2.1E-07
Y92	1.2E-09	-	4.8E-08	1.2E-09

Table 12.2-12b: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	LCW Reverse Osmosis Unit	LCW Solids Collection Filter	HCW Granulated Activated Charcoal	Drum Dryer
	(Ci)	(Ci)	(Ci)	(Ci/Drum)
Y93	1.2E-09	-	-	1.2E-09
Zr97	2.9E-09	-	-	2.9E-09
Nb95	9.5E-07	-	1.0E-02	9.0E-06
Mo99	2.0E-05	-	-	2.0E-05
Mo101	2.9E-09	-	-	2.9E-09
Tc99m	9.5E-06	-	1.1E-03	9.6E-06
Tc99	1.9E-09	-	2.6E-09	3.6E-08
Ru103	8.1E-08	-	-	2.8E-07
Ru105	4.3E-10	-	-	4.3E-10
Ru106	5.9E-08	-	-	8.2E-07
Rh103m	3.5E-08	-	1.7E-05	1.2E-07
Rh105	7.3E-09	-	2.7E-08	7.3E-09
Rh106	2.6E-08	-	9.0E-05	3.6E-07
Ag110	6.6E-08	-	1.7E-03	8.0E-07
Sb124	1.3E-10	-	3.4E-08	6.3E-10
Sb125	1.1E-09	-	2.3E-06	1.8E-08
Sb127	1.5E-09	-	8.2E-08	1.5E-09
Sb129	9.0E-11	-	4.8E-09	9.0E-11
Te125m	1.4E-07	-	5.8E-07	6.5E-07
Te127m	5.5E-07	-	1.7E-08	4.3E-06
Te127	3.0E-07	-	2.9E-04	1.9E-06
Te129m	1.4E-06	-	1.3E-09	4.2E-06
Te129	3.9E-07	-	1.7E-04	1.2E-06
Te131m	5.1E-07	-	-	5.1E-07
Te131	5.3E-08	-	7.0E-06	5.3E-08
Te132	9.4E-06	-	-	9.6E-06
Te133m	9.8E-09	-	-	9.8E-09
Te134	1.1E-08	-	-	1.1E-08
Ba137m	6.6E-03	-	1.6E+00	1.2E-01
Ba139	3.9E-10	-	-	3.9E-10
Ba140	3.0E-07	-	-	4.7E-07
La140	1.4E-07	-	2.9E-05	2.2E-07
La141	3.4E-10	-	-	3.4E-10
La142	6.3E-11	-	-	6.3E-11
Ce141	6.2E-08	-	2.1E-08	1.8E-07
Ce143	5.8E-09	-	-	5.8E-09
Ce144	6.3E-08	-	-	8.0E-07
Pr143	4.4E-08	-	3.6E-07	7.0E-08
Pr144	2.7E-08	-	7.8E-05	3.5E-07
Np239	2.1E-07	-	-	2.1E-07
Na24	2.2E-05	2.5E-03	-	2.2E-05
Cr51	1.3E-04	3.6E-02	3.3E-02	3.3E-04
Mn54	8.3E-05	1.8E-01	1.8E-01	1.1E-03
Fe55	6.4E-05	2.1E-01	-	1.1E-03

Table 12.2-12b: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	LCW Reverse Osmosis Unit	LCW Solids Collection Filter	HCW Granulated Activated Charcoal	Drum Dryer
	(Ci)	(Ci)	(Ci)	(Ci/Drum)
Fe59	1.3E-05	5.7E-03	-	5.1E-05
Co58	1.9E-03	1.2E+00	1.1E+00	1.1E-02
Co60	2.8E-05	1.0E-01	1.2E-01	5.0E-04
Ni63	1.4E-05	5.9E-02	-	2.7E-04
Zn65	2.6E-05	4.8E-02	-	3.2E-04
Zr95	1.8E-05	1.1E-02	-	9.5E-05
Ag110m	6.7E-05	1.2E-01	1.3E-01	8.2E-04
W187	3.7E-06	4.1E-04	-	3.7E-06
H3	-	-	-	-
C14	-	-	-	-
N16	-	-	-	-
Ar41	-	-	-	-
Total	**3.7E-02**	**2.0E+00**	**3.2E+00**	**5.7E-01**

Table 12.2-12c: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content

Isotope	LCW Processing Skid	Oil Separator
	(Ci)	(Ci)
Kr83m	-	-
Kr85m	-	-
Kr85	-	-
Kr87	-	-
Kr88	-	-
Kr89	-	-
Xe131m	-	-
Xe133m	-	-
Xe133	-	-
Xe135m	-	-
Xe135	-	-
Xe137	-	-
Xe138	-	-
Br82	1.7E-05	4.1E-06
Br83	6.6E-06	2.3E-05
Br84	6.8E-07	1.1E-05
Br85	7.4E-09	1.3E-06
I129	9.6E-08	6.7E-11
I130	4.8E-05	3.3E-05
I131	3.0E-02	8.4E-04
I132	5.7E-04	3.9E-04
I133	3.2E-03	1.3E-03
I134	2.4E-05	2.3E-04
I135	6.3E-04	8.1E-04
Rb86m	2.0E-11	9.6E-10
Rb86	3.1E-03	5.6E-06
Rb88	3.5E-04	9.6E-04
Rb89	1.4E-05	4.4E-05
Cs132	2.1E-05	1.1E-07
Cs134	8.7E+00	8.0E-04
Cs135m	7.2E-07	6.7E-07
Cs136	6.8E-02	1.7E-04
Cs137	6.0E+00	4.1E-04
Cs138	2.4E-04	3.6E-04
P32	8.7E-10	1.6E-11
Co57	1.1E-10	1.2E-13
Sr89	1.5E-04	7.1E-07
Sr90	2.1E-04	1.1E-07
Sr91	5.8E-07	3.7E-07
Sr92	8.6E-08	2.0E-07
Y90	9.4E-05	2.7E-08
Y91m	1.8E-07	2.0E-07
Y91	2.4E-05	1.0E-07
Y92	1.3E-07	1.7E-07
Y93	1.3E-07	7.9E-08

Table 12.2-12c: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	LCW Processing Skid (Ci)	Oil Separator (Ci)
Zr97	3.2E-07	1.2E-07
Nb95	1.0E-03	2.1E-07
Mo99	2.3E-03	2.1E-04
Mo101	3.2E-07	8.1E-06
Tc99m	1.1E-03	2.0E-04
Tc99	7.9E-06	4.0E-09
Ru103	3.1E-05	2.0E-07
Ru105	4.8E-08	6.7E-08
Ru106	1.4E-04	1.3E-07
Rh103m	1.4E-05	2.0E-07
Rh105	8.1E-07	1.4E-07
Rh106	6.1E-05	1.3E-07
Ag110	1.2E-04	3.9E-07
Sb124	7.1E-08	3.0E-10
Sb125	3.5E-06	2.3E-09
Sb127	1.7E-07	1.1E-08
Sb129	1.0E-08	1.4E-08
Te125m	7.4E-05	3.3E-07
Te127m	5.2E-04	1.2E-06
Te127	2.3E-04	4.9E-06
Te129m	4.7E-04	3.6E-0
Te129	1.3E-04	5.1E-06
Te131m	5.7E-05	1.2E-05
Te131	5.9E-06	5.8E-06
Te132	1.1E-03	8.5E-05
Te133m	1.1E-0	7.3E-06
Te134	1.2E-06	1.0E-05
Ba137m	4.6E+00	3.9E-04
Ba139	4.3E-08	1.9E-07
Ba140	5.2E-05	1.0E-06
La140	2.5E-05	3.0E-07
La141	3.8E-08	6.0E-08
La142	7.0E-09	2.8E-08
Ce141	2.0E-05	1.6E-07
Ce143	6.5E-07	1.2E-07
Ce144	1.3E-04	1.4E-07
Pr143	7.8E-06	1.4E-07
Pr144	5.5E-05	1.3E-07
Np239	2.3E-05	2.5E-06
Na24	2.5E-03	1.0E-03
Cr51	3.6E-02	5.8E-05
Mn54	1.8E-01	3.0E-05
Fe55	2.1E-01	2.2E-05
Fe59	5.7E-03	5.6E-06
Co58	1.2E+00	8.5E-05

Table 12.2-12c: Liquid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	LCW Processing Skid	Oil Separator
	(Ci)	(Ci)
Co60	1.0E-01	9.8E-06
Ni63	5.9E-02	4.9E-06
Zn65	4.8E-02	9.4E-06
Zr95	1.1E-02	7.2E-06
Ag110m	1.2E-01	2.4E-05
W187	4.1E-04	5.2E-05
H3	-	6.1E-01
C14	-	2.0E-05
N16	-	-
Ar41	-	-
Total	**2.1E+01**	**6.2E-01**

Table 12.2-13a: Liquid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Lower Bound	Upper Bound	LCW Collection Tank	HCW Collection Tank	LCW Sample Tank	HCW Sample Tank
	(MeV)	(Mev)	(γ/s)	(γ/s)	(γ/s)	(γ/s)
1	1.00E-02	2.00E-02	6.8E+08	7.5E+07	3.8E+06	3.7E+05
2	2.00E-02	3.00E-02	4.5E+08	8.2E+07	1.2E+05	9.1E+03
3	3.00E-02	4.50E-02	6.0E+08	7.3E+07	3.1E+05	2.5E+04
4	4.50E-02	6.00E-02	2.5E+08	2.9E+07	3.4E+04	2.5E+03
5	6.00E-02	7.00E-02	2.3E+08	3.0E+07	3.1E+04	3.8E+03
6	7.00E-02	7.50E-02	5.7E+07	1.2E+07	4.2E+03	3.2E+02
7	7.50E-02	1.00E-01	3.2E+08	4.0E+07	3.0E+04	4.5E+03
8	1.00E-01	1.50E-01	5.0E+08	1.1E+08	5.4E+03	1.7E+03
9	1.50E-01	2.00E-01	5.6E+08	5.3E+07	6.5E+04	9.9E+03
10	2.00E-01	2.60E-01	2.5E+08	4.9E+07	2.7E+03	7.7E+02
11	2.60E-01	3.00E-01	3.2E+08	6.8E+07	4.4E+04	8.1E+03
12	3.00E-01	4.00E-01	1.3E+09	5.2E+08	1.7E+05	4.8E+04
13	4.00E-01	4.50E-01	2.9E+08	5.4E+07	1.4E+02	3.5E+02
14	4.50E-01	5.10E-01	1.5E+09	1.4E+08	8.5E+04	6.9E+03
15	5.10E-01	5.12E-01	3.2E+07	8.0E+08	1.8E+03	7.3E+04
16	5.12E-01	6.00E-01	3.2E+09	6.5E+08	1.3E+06	1.0E+05
17	6.00E-01	7.00E-01	9.3E+09	1.1E+09	7.4E+06	5.7E+05
18	7.00E-01	8.00E-01	9.3E+09	9.1E+08	5.2E+06	4.0E+05
19	8.00E-01	9.00E-01	3.6E+09	3.1E+09	5.8E+05	3.2E+05
20	9.00E-01	1.00E+00	9.2E+08	1.3E+08	6.8E+02	2.9E+03
21	1.00E+00	1.20E+00	3.4E+09	4.4E+08	4.2E+05	5.8E+04
22	1.20E+00	1.33E+00	8.0E+08	1.8E+08	7.9E+04	1.4E+04
23	1.33E+00	1.44E+00	4.7E+09	6.8E+08	1.6E+05	1.6E+04
24	1.44E+00	1.50E+00	8.5E+07	3.6E+07	8.3E+01	3.5E+02
25	1.50E+00	1.57E+00	4.7E+07	2.2E+07	7.4E+02	1.3E+03
26	1.57E+00	1.66E+00	1.7E+07	5.2E+06	3.1E+01	2.7E+01
27	1.66E+00	1.80E+00	1.3E+08	7.8E+07	3.1E+01	1.2E+03
28	1.80E+00	2.00E+00	2.5E+09	1.9E+08	1.9E+00	1.2E+01
29	2.00E+00	2.15E+00	1.0E+08	1.1E+07	6.6E-01	4.7E+00
30	2.15E+00	2.35E+00	7.2E+08	5.5E+07	5.1E-01	2.3E+00
31	2.35E+00	2.50E+00	1.8E+07	4.5E+06	3.1E-01	1.3E+00
32	2.50E+00	2.75E+00	1.1E+09	2.3E+08	1.2E+00	4.5E+00
33	2.75E+00	3.00E+00	4.4E+08	1.7E+08	2.3E-02	3.1E+00
34	3.00E+00	3.50E+00	7.7E+07	5.9E+06	9.2E-03	7.2E-03
35	3.50E+00	4.00E+00	1.5E+07	1.6E+06	8.0E-08	4.9E-03
36	4.00E+00	4.50E+00	2.9E+06	2.3E+05	4.9E-11	5.3E-05
37	4.50E+00	5.00E+00	2.1E+07	1.6E+06	-	-
38	5.00E+00	5.50E+00	4.0E+04	3.0E+03	-	-
39	5.50E+00	6.00E+00	-	-	-	-
40	6.00E+00	6.50E+00	-	-	-	-
41	6.50E+00	7.00E+00	-	-	-	-
42	7.00E+00	7.50E+00	-	-	-	-
43	7.50E+00	8.00E+00	-	-	-	-

Table 12.2-13a: Liquid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Lower Bound	Upper Bound	LCW Collection Tank	HCW Collection Tank	LCW Sample Tank	HCW Sample Tank
	(MeV)	(Mev)	(γ/s)	(γ/s)	(γ/s)	(γ/s)
44	8.00E+00	1.00E+01	-	-	-	-
45	1.00E+01	1.20E+01	-	-	-	-
46	1.20E+01	1.40E+01	-	-	-	-
47	1.40E+01	2.00E+01	-	-	-	-
Total			**4.8E+10**	**1.0E+10**	**2.0E+07**	**2.0E+06**

Table 12.2-13b: Liquid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Lower Bound (MeV)	Upper Bound (MeV)	LCW RO (γ/s)	LCW SCF (γ/s)	HCW GAC (γ/s)	Drum Dryer (γ/s)
1	1.00E-02	2.00E-02	3.0E+06	9.2E+07	8.9E+07	4.4E+07
2	2.00E-02	3.00E-02	2.1E+06	1.0E+08	9.9E+07	2.4E+07
3	3.00E-02	4.50E-02	2.7E+07	3.9E+07	3.8E+09	3.9E+08
4	4.50E-02	6.00E-02	8.4E+05	3.0E+07	2.7E+07	1.2E+07
5	6.00E-02	7.00E-02	4.5E+06	1.7E+07	1.3E+07	1.2E+07
6	7.00E-02	7.50E-02	1.7E+05	8.1E+06	5.5E+06	2.2E+06
7	7.50E-02	1.00E-01	5.1E+06	2.3E+07	2.2E+07	1.5E+07
8	1.00E-01	1.50E-01	1.1E+06	3.2E+07	6.8E+07	7.7E+06
9	1.50E-01	2.00E-01	1.4E+07	2.3E+07	1.5E+07	2.5E+07
10	2.00E-01	2.60E-01	6.8E+05	1.5E+07	1.5E+07	4.8E+06
11	2.60E-01	3.00E-01	9.7E+06	5.7E+06	6.0E+06	1.6E+07
12	3.00E-01	4.00E-01	4.3E+07	1.4E+08	1.3E+08	6.7E+07
13	4.00E-01	4.50E-01	1.6E+05	1.8E+08	1.8E+08	1.3E+06
14	4.50E-01	5.10E-01	1.0E+07	7.6E+06	6.0E+06	1.5E+08
15	5.10E-01	5.12E-01	2.1E+07	1.3E+10	1.2E+10	1.2E+08
16	5.12E-01	6.00E-01	1.5E+08	3.5E+06	6.4E+06	2.4E+09
17	6.00E-01	7.00E-01	7.9E+08	5.3E+09	6.0E+10	1.3E+10
18	7.00E-01	8.00E-01	5.9E+08	2.4E+09	2.4E+09	9.5E+09
19	8.00E-01	9.00E-01	1.7E+08	5.3E+10	4.9E+10	9.6E+08
20	9.00E-01	1.00E+00	9.4E+05	1.6E+09	1.6E+09	1.0E+07
21	1.00E+00	1.20E+00	7.5E+07	5.1E+09	4.6E+09	4.0E+08
22	1.20E+00	1.33E+00	1.7E+07	2.1E+09	2.3E+09	3.7E+07
23	1.33E+00	1.44E+00	2.1E+07	3.1E+09	3.2E+09	3.1E+08
24	1.44E+00	1.50E+00	1.3E+05	1.9E+08	2.0E+08	1.3E+06
25	1.50E+00	1.57E+00	4.7E+05	6.7E+08	6.8E+08	4.6E+06
26	1.57E+00	1.66E+00	7.5E+03	1.1E+06	2.2E+06	1.7E+04
27	1.66E+00	1.80E+00	4.0E+05	2.2E+08	2.0E+08	2.0E+06
28	1.80E+00	2.00E+00	1.6E+05	7.8E+05	1.2E+06	1.6E+05
29	2.00E+00	2.15E+00	1.0E+04	4.6E+04	2.3E+05	1.0E+04
30	2.15E+00	2.35E+00	7.8E+04	4.4E+04	1.5E+05	7.8E+04
31	2.35E+00	2.50E+00	3.7E+03	4.0E+01	5.2E+04	3.7E+03
32	2.50E+00	2.75E+00	4.9E+05	4.8E+07	4.8E+04	4.9E+05
33	2.75E+00	3.00E+00	4.0E+05	4.4E+07	9.3E+02	4.0E+05
34	3.00E+00	3.50E+00	5.7E+03	8.4E-02	3.6E+02	5.7E+03
35	3.50E+00	4.00E+00	1.6E+03	7.0E+04	2.4E-02	1.6E+03
36	4.00E+00	4.50E+00	2.2E+02	7.7E+02	-	2.2E+02
37	4.50E+00	5.00E+00	1.3E+03	-	-	1.3E+03
38	5.00E+00	5.50E+00	2.4E+00	-	-	2.4E+00
39	5.50E+00	6.00E+00	-	-	-	-
40	6.00E+00	6.50E+00	-	-	-	-
41	6.50E+00	7.00E+00	-	-	-	-
42	7.00E+00	7.50E+00	-	-	-	-
43	7.50E+00	8.00E+00	-	-	-	-
44	8.00E+00	1.00E+01	-	-	-	-
45	1.00E+01	1.20E+01	-	-	-	-

Table 12.2-13b: Liquid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Lower Bound (MeV)	Upper Bound (MeV)	LCW RO (γ/s)	LCW SCF (γ/s)	HCW GAC (γ/s)	Drum Dryer (γ/s)
46	1.20E+01	1.40E+01	-	-	-	-
47	1.40E+01	2.00E+01	-	-	-	-
Total			**2.0E+09**	**8.7E+10**	**1.4E+11**	**2.8E+10**

Table 12.2-13c: Liquid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Lower Bound	Upper Bound	LCW Processing Skid	Oil Separator
	(MeV)	(Mev)	(γ/s)	(γ/s)
1	1.00E-02	2.00E-02	1.6E+09	4.4E+06
2	2.00E-02	3.00E-02	9.5E+08	6.0E+06
3	3.00E-02	4.50E-02	1.4E+10	4.6E+06
4	4.50E-02	6.00E-02	4.3E+08	1.7E+06
5	6.00E-02	7.00E-02	3.2E+08	1.3E+06
6	7.00E-02	7.50E-02	8.2E+07	5.2E+05
7	7.50E-02	1.00E-01	4.4E+08	2.2E+06
8	1.00E-01	1.50E-01	3.2E+08	1.0E+07
9	1.50E-01	2.00E-01	5.4E+08	2.9E+06
10	2.00E-01	2.60E-01	1.9E+08	4.5E+06
11	2.60E-01	3.00E-01	3.7E+08	4.6E+06
12	3.00E-01	4.00E-01	2.2E+09	3.1E+07
13	4.00E-01	4.50E-01	1.9E+08	4.4E+06
14	4.50E-01	5.10E-01	4.8E+09	6.7E+06
15	5.10E-01	5.12E-01	1.3E+10	1.9E+06
16	5.12E-01	6.00E-01	7.8E+10	5.6E+07
17	6.00E-01	7.00E-01	4.5E+11	6.7E+07
18	7.00E-01	8.00E-01	3.1E+11	4.7E+07
19	8.00E-01	9.00E-01	6.8E+10	3.6E+07
20	9.00E-01	1.00E+00	1.6E+09	7.8E+06
21	1.00E+00	1.20E+00	1.6E+10	2.6E+07
22	1.20E+00	1.33E+00	2.6E+09	1.4E+07
23	1.33E+00	1.44E+00	1.3E+10	5.2E+07
24	1.44E+00	1.50E+00	1.9E+08	3.4E+06
25	1.50E+00	1.57E+00	6.7E+08	1.1E+06
26	1.57E+00	1.66E+00	2.1E+06	5.3E+05
27	1.66E+00	1.80E+00	2.3E+08	6.9E+06
28	1.80E+00	2.00E+00	4.4E+06	9.0E+06
29	2.00E+00	2.15E+00	5.8E+05	7.7E+05
30	2.15E+00	2.35E+00	1.7E+06	2.5E+06
31	2.35E+00	2.50E+00	3.1E+05	4.3E+05
32	2.50E+00	2.75E+00	4.9E+07	2.2E+07
33	2.75E+00	3.00E+00	4.4E+07	1.8E+07
34	3.00E+00	3.50E+00	1.1E+05	2.7E+05
35	3.50E+00	4.00E+00	9.0E+04	1.0E+05
36	4.00E+00	4.50E+00	4.8E+03	1.1E+04
37	4.50E+00	5.00E+00	2.4E+04	6.7E+04
38	5.00E+00	5.50E+00	4.6E+01	1.3E+02
39	5.50E+00	6.00E+00	-	-
40	6.00E+00	6.50E+00	-	-
41	6.50E+00	7.00E+00	-	-
42	7.00E+00	7.50E+00	-	-
43	7.50E+00	8.00E+00	-	-
44	8.00E+00	1.00E+01	-	-

Table 12.2-13c: Liquid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Lower Bound	Upper Bound	LCW Processing Skid	Oil Separator
	(MeV)	(Mev)	(γ/s)	(γ/s)
45	1.00E+01	1.20E+01	-	-
46	1.20E+01	1.40E+01	-	-
47	1.40E+01	2.00E+01	-	-
Total			**9.8E+11**	**4.6E+08**

Table 12.2-14: Gaseous Radioactive Waste System Component Source Term Inputs

Model Parameter	Value
GRWS guard bed	-
Contents	Guard Bed Buildup
Geometry	Vertical Cylinder
Source dimensions of vessel	Diameter=2 ft, Height=3.2 ft
Shield thickness of steel shell	0.25 in
GRWS decay bed (four each per train; two trains)	-
Inputs	Sum of Decay Bed 5a, 6a, 7a, and 8a
Geometry	Vertical Cylinder
Source dimensions of vessel	Diameter=6.5 ft, Height=4.4 ft
Shield thickness o fsteel shell	0.25 in

Table 12.2-15: Gaseous Radioactive Waste System Component Source Term Radionuclide Content

Isotope	Guard Bed (Ci)	Charcoal Decay Bed 5A (Ci)	Charcoal Decay Bed 6A (Ci)	Charcoal Decay Bed 7A (Ci)	Charcoal Decay Bed 8A (Ci)
Kr83m	4.6E-04	5.6E-04	1.4E-06	6.4E-08	6.3E-09
Kr85m	3.0E-03	5.3E-03	3.6E-04	2.4E-05	1.7E-06
Kr85	7.6E-01	2.8E+00	2.8E+00	2.8E+00	2.8E+00
Kr87	8.2E-04	8.8E-04	6.7E-08	5.1E-12	3.9E-16
Kr88	3.9E-03	5.7E-03	8.1E-05	1.2E-06	1.7E-08
Kr89	2.4E-06	2.4E-06	-	-	-
Xe131m	3.7E-01	9.8E-01	3.6E-01	1.3E-01	4.7E-02
Xe133m	1.9E-01	2.4E-01	1.0E-03	5.2E-06	1.1E-07
Xe133	2.0E+01	3.9E+01	4.1E+00	4.2E-01	4.3E-02
Xe135m	1.2E-04	1.2E-04	9.8E-06	9.8E-07	9.8E-08
Xe135	8.3E-02	8.3E-02	1.8E-04	1.6E-05	1.6E-06
Xe137	9.7E-06	9.7E-06	-	-	-
Xe138	1.2E-04	1.2E-04	-	-	-
Br82	1.5E-06	1.5E-06	1.5E-07	1.5E-08	1.5E-09
Br83	5.7E-07	5.7E-07	5.7E-08	5.7E-09	5.7E-10
Br84	5.8E-08	5.8E-08	5.8E-09	5.8E-10	5.8E-11
Br85	6.4E-10	6.4E-10	6.4E-11	6.4E-12	6.4E-13
I129	2.5E-07	2.5E-07	2.5E-08	2.5E-09	2.5E-10
I130	4.2E-06	4.2E-06	4.2E-07	4.2E-08	4.2E-09
I131	1.6E-03	1.6E-03	1.6E-04	1.6E-05	1.6E-06
I132	9.1E-06	9.1E-06	9.1E-07	9.1E-08	9.1E-09
I133	2.7E-04	2.7E-04	2.7E-05	2.7E-06	2.7E-07
I134	2.0E-06	2.0E-06	2.0E-07	2.0E-08	2.0E-09
I135	5.4E-05	5.4E-05	5.4E-06	5.4E-07	5.4E-08
Rb86m	-	-	-	-	-
Rb86	-	-	-	-	-
Rb88	3.9E-03	5.7E-03	8.1E-05	1.2E-06	1.7E-08
Rb89	2.4E-06	2.4E-06	-	-	-
Cs132	-	-	-	-	-
Cs134	-	-	-	-	-
Cs135m	-	-	-	-	-
Cs136	-	-	-	-	-
Cs137	7.2E-06	7.2E-06	-	-	-
Cs138	1.2E-04	1.2E-04	-	-	-
P32	-	-	-	-	-
Co57	-	-	-	-	-
Sr89	2.4E-06	2.4E-06	-	-	-
Sr90	-	-	-	-	-
Sr91	-	-	-	-	-
Sr92	-	-	-	-	-
Y90	-	-	-	-	-
Y91m	-	-	-	-	-
Y91	-	-	-	-	-
Y92	-	-	-	-	-
Y93	-	-	-	-	-

Table 12.2-15: Gaseous Radioactive Waste System Component Source Term Radionuclide Content (Continued)

Isotope	Guard Bed (Ci)	Charcoal Decay Bed 5A (Ci)	Charcoal Decay Bed 6A (Ci)	Charcoal Decay Bed 7A (Ci)	Charcoal Decay Bed 8A (Ci)
Zr97	-	-	-	-	-
Nb95	-	-	-	-	-
Mo99	-	-	-	-	-
Mo101	-	-	-	-	-
Tc99m	-	-	-	-	-
Tc99	-	-	-	-	-
Ru103	-	-	-	-	-
Ru105	-	-	-	-	-
Ru106	-	-	-	-	-
Rh103m	-	-	-	-	-
Rh105	-	-	-	-	-
Rh106	-	-	-	-	-
Ag110	-	-	-	-	-
Sb124	-	-	-	-	-
Sb125	-	-	-	-	-
Sb127	-	-	-	-	-
Sb129	-	-	-	-	-
Te125m	-	-	-	-	-
Te127m	-	-	-	-	-
Te127	-	-	-	-	-
Te129m	-	-	-	-	-
Te129	-	-	-	-	-
Te131m	-	-	-	-	-
Te131	-	-	-	-	-
Te132	-	-	-	-	-
Te133m	-	-	-	-	-
Te134	-	-	-	-	-
Ba137m	6.8E-06	6.8E-06	-	-	-
Ba139	-	-	-	-	-
Ba140	-	-	-	-	-
La140	-	-	-	-	-
La141	-	-	-	-	-
La142	-	-	-	-	-
Ce141	-	-	-	-	-
Ce143	-	-	-	-	-
Ce144	-	-	-	-	-
Pr143	-	-	-	-	-
Pr144	-	-	-	-	-
Np239	-	-	-	-	-
Na24	-	-	-	-	-
Cr51	-	-	-	-	-
Mn54	-	-	-	-	-
Fe55	-	-	-	-	-
Fe59	-	-	-	-	-
Co58	-	-	-	-	-

Table 12.2-15: Gaseous Radioactive Waste System Component Source Term Radionuclide Content (Continued)

Isotope	Guard Bed (Ci)	Charcoal Decay Bed 5A (Ci)	Charcoal Decay Bed 6A (Ci)	Charcoal Decay Bed 7A (Ci)	Charcoal Decay Bed 8A (Ci)
Co60	-	-	-	-	-
Ni63	-	-	-	-	-
Zn65	-	-	-	-	-
Zr95	-	-	-	-	-
Ag110m	-	-	-	-	-
W187	-	-	-	-	-
H3	-	-	-	-	-
C14	-	-	-	-	-
N16	-	-	-	-	-
Ar41	1.4E-02	3.8E-02	1.4E-02	5.2E-03	1.9E-03
Total	2.1E+01	4.3E+01	7.3E+00	3.4E+00	2.9E+00

Table 12.2-15a: Degasifier Radiological Content

Isotope	Activity (Ci)
Kr83m	8.8E-02
Kr85m	3.7E-01
Kr85	6.8E+01
Kr87	2.0E-01
Kr88	5.9E-01
Kr89	1.0E-02
Xe131m	1.6E+00
Xe133m	1.3E+00
Xe133	9.9E+01
Xe135m	1.2E-01
Xe135	2.7E+00
Xe137	3.6E-02
Xe138	1.4E-01
Br82	2.5E-05
Br83	1.4E-04
Br84	6.4E-05
Br85	5.9E-06
I129	4.0E-10
I130	2.0E-04
I131	5.1E-03
I132	2.4E-03
I133	7.7E-03
I134	1.4E-03
I135	4.9E-03
Rb86m	1.3E-07
Rb86	1.7E-03
Rb88	3.1E-01
Rb89	1.3E-02
Cs132	3.4E-05
Cs134	2.5E-01
Cs135m	2.0E-04
Cs136	5.4E-02
Cs137	1.3E-01
Cs138	1.1E-01
P32	1.9E-10
Co57	1.5E-12
Sr89	8.9E-06
Sr90	1.4E-06
Sr91	4.6E-06
Sr92	2.4E-06
Y90	3.3E-07
Y91m	2.4E-06
Y91	1.3E-06
Y92	2.1E-06
Y93	9.8E-07
Zr97	1.5E-06
Nb95	2.6E-06

Table 12.2-15a: Degasifier Radiological Content (Continued)

Isotope	Activity (Ci)
Mo99	2.6E-03
Mo101	9.4E-05
Tc99m	2.4E-03
Tc99	4.9E-08
Ru103	2.5E-06
Ru105	8.2E-07
Ru106	1.6E-06
Rh103m	2.5E-06
Rh105	1.7E-06
Rh106	1.6E-06
Ag110	4.1E-06
Sb124	3.7E-09
Sb125	2.8E-08
Sb127	1.4E-07
Sb129	1.7E-07
Te125m	4.0E-06
Te127m	1.5E-05
Te127	6.1E-05
Te129m	4.4E-05
Te129	6.2E-05
Te131m	1.4E-04
Te131	7.0E-05
Te132	1.0E-03
Te133m	8.9E-05
Te134	1.3E-04
Ba137m	3.7E-02
Ba139	2.4E-06
Ba140	1.3E-05
La140	3.7E-06
La141	7.3E-07
La142	3.5E-07
Ce141	2.0E-06
Ce143	1.5E-06
Ce144	1.7E-06
Pr143	1.7E-06
Pr144	1.7E-06
Np239	3.1E-05
Na24	1.3E-02
Cr51	7.1E-04
Mn54	3.6E-04
Fe55	2.7E-04
Fe59	6.9E-05
Co58	1.1E-03
Co60	1.2E-04
Ni63	6.0E-05
Zn65	1.2E-04
Zr95	8.9E-05

Table 12.2-15a: Degasifier Radiological Content (Continued)

Isotope	Activity (Ci)
Ag110m	3.0E-04
W187	6.4E-04
H3	1.2E+02
C14	1.2E-02
N16	3.5E-02
Ar41	9.6E+00

Note: The radiological content of the liquid in the degasifier is primary coolant that has been processed through the CVCS demineralizers using the shared systems fuel failure fraction, and decayed for 10 N-16 half-lives (71.3 seconds). The CVCS demineralizer decontamination factors are provided in Table 11.1-2.

Table 12.2-16: Gaseous Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Energy (MeV)		Guard Bed	Charcoal Decay Bed 5A	Charcoal Decay Bed 6A	Charcoal Decay Bed 7A	Charcoal Decay Bed 8A
	Lower Bound	Upper Bound	photons/sec	photons/sec	photons/sec	photons/sec	photons/sec
1	1.00E-02	2.00E-02	1.3E+09	2.7E+09	7.0E+08	4.8E+08	4.6E+08
2	2.00E-02	3.00E-02	8.1E+09	1.8E+10	5.1E+09	2.0E+09	8.7E+08
3	3.00E-02	4.50E-02	3.1E+11	6.0E+11	6.6E+10	7.7E+09	1.2E+09
4	4.50E-02	6.00E-02	2.8E+08	6.0E+08	1.8E+08	1.4E+08	1.3E+08
5	6.00E-02	7.00E-02	1.2E+08	2.6E+08	8.1E+07	6.2E+07	6.0E+07
6	7.00E-02	7.50E-02	4.8E+07	1.0E+08	3.4E+07	2.6E+07	2.5E+07
7	7.50E-02	1.00E-01	2.6E+11	5.1E+11	5.4E+10	5.7E+09	6.6E+08
8	1.00E-01	1.50E-01	1.6E+08	3.4E+08	1.1E+08	8.7E+07	8.5E+07
9	1.50E-01	2.00E-01	8.2E+08	1.8E+09	3.9E+08	1.4E+08	7.9E+07
10	2.00E-01	2.60E-01	3.7E+09	4.0E+09	3.3E+07	2.2E+07	2.1E+07
11	2.60E-01	3.00E-01	2.9E+07	5.5E+07	1.4E+07	9.6E+06	9.1E+06
12	3.00E-01	4.00E-01	1.2E+08	1.8E+08	2.7E+07	1.4E+07	1.3E+07
13	4.00E-01	4.50E-01	2.8E+07	3.1E+07	2.0E+06	1.9E+06	1.9E+06
14	4.50E-01	5.10E-01	4.1E+06	5.8E+06	1.1E+06	1.0E+06	9.9E+05
15	5.10E-01	5.12E-01	6.2E+07	2.3E+08	2.3E+08	2.3E+08	2.3E+08
16	5.12E-01	6.00E-01	7.0E+07	2.2E+08	2.1E+08	2.1E+08	2.1E+08
17	6.00E-01	7.00E-01	9.2E+07	9.3E+07	8.6E+05	1.9E+05	1.2E+05
18	7.00E-01	8.00E-01	5.2E+06	6.0E+06	2.3E+05	2.5E+04	4.6E+03
19	8.00E-01	9.00E-01	3.7E+07	5.2E+07	7.6E+05	2.2E+04	2.2E+03
20	9.00E-01	1.00E+00	1.3E+07	1.9E+07	2.9E+05	6.6E+03	6.7E+02
21	1.00E+00	1.20E+00	8.8E+06	1.1E+07	2.1E+05	1.1E+04	1.2E+03
22	1.20E+00	1.33E+00	5.4E+08	1.4E+09	5.3E+08	2.0E+08	7.3E+07
23	1.33E+00	1.44E+00	8.2E+06	1.0E+07	9.2E+04	1.9E+03	1.1E+02
24	1.44E+00	1.50E+00	4.7E+05	5.6E+05	2.4E+04	2.0E+03	2.1E+02
25	1.50E+00	1.57E+00	1.9E+07	2.8E+07	4.0E+05	6.1E+03	1.4E+02
26	1.57E+00	1.66E+00	1.0E+06	1.4E+06	1.9E+04	3.6E+02	2.5E+01
27	1.66E+00	1.80E+00	3.8E+06	5.1E+06	3.4E+05	1.0E+05	3.6E+04
28	1.80E+00	2.00E+00	3.3E+07	4.7E+07	6.7E+05	9.9E+03	1.8E+02
29	2.00E+00	2.15E+00	1.4E+07	2.0E+07	2.6E+05	4.0E+03	7.8E+01
30	2.15E+00	2.35E+00	2.5E+07	3.6E+07	5.1E+05	7.4E+03	1.2E+02

Table 12.2-16: Gaseous Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy (MeV)		Guard Bed	Charcoal Decay Bed 5A	Charcoal Decay Bed 6A	Charcoal Decay Bed 7A	Charcoal Decay Bed 8A
	Lower Bound	Upper Bound	photons/sec	photons/sec	photons/sec	photons/sec	photons/sec
31	2.35E+00	2.50E+00	5.1E+07	7.3E+07	1.0E+06	1.5E+04	2.4E+02
32	2.50E+00	2.75E+00	8.8E+06	1.1E+07	9.3E+04	1.3E+03	1.9E+01
33	2.75E+00	3.00E+00	5.8E+05	7.9E+05	9.4E+03	1.3E+02	2.0E+00
34	3.00E+00	3.50E+00	8.5E+05	1.2E+06	1.4E+04	2.0E+02	2.9E+00
35	3.50E+00	4.00E+00	1.3E+05	1.8E+05	2.4E+03	3.6E+01	6.5E-01
36	4.00E+00	4.50E+00	2.4E+04	3.4E+04	4.7E+02	6.8E+00	1.0E-01
37	4.50E+00	5.00E+00	2.8E+05	4.0E+05	5.7E+03	8.2E+01	1.2E+00
38	5.00E+00	5.50E+00	5.2E+02	7.5E+02	1.1E+01	1.5E-01	2.2E-03
39	5.50E+00	6.00E+00	-	-	-	-	-
40	6.00E+00	6.50E+00	-	-	-	-	-
41	6.50E+00	7.00E+00	-	-	-	-	-
42	7.00E+00	7.50E+00	-	-	-	-	-
43	7.50E+00	8.00E+00	-	-	-	-	-
44	8.00E+00	1.00E+01	-	-	-	-	-
45	1.00E+01	1.20E+01	-	-	-	-	-
46	1.20E+01	1.40E+01	-	-	-	-	-
47	1.40E+01	2.00E+01	-	-	-	-	-
Total			**5.8E+11**	**1.1E+12**	**1.3E+11**	**1.7E+10**	**4.1E+09**

Table 12.2-17: Solid Radioactive Waste System Source Term Inputs

Model Parameter	Value
Spent resin storage tank-	-
Contents	Spent Resin from the CVCS and PCWS
Geometry	Vertical Cylinder
Source dimensions of vessel	Diameter=12 ft, height=14 ft
Shield thickness of steel shell	0.25 in
Phase separator tank-	-
Inputs	Spent Resin from the LRWS
Geometry	Vertical Cylinder
Source dimensions of vessel	Diameter=12 ft, height=17 ft
Shield thickness of steel shell	0.25 in
High integrity container (HIC)-	-
Inputs	Spent Resins from SRST
Geometry	Vertical Cylinder
Source dimensions of container	diameter=62 in, height=75 in
Array of HICs	2x1 and 3x1

Table 12.2-18: Solid Radioactive Waste System Component Source Terms - Radionuclide Content

Isotope	Spent Resin Storage Tank (Ci)	Phase Separator Tank (Ci)	HIC from SRST (Ci/HIC)
Kr83m	2.4E-11	-	-
Kr85m	7.7E-12	-	
Kr85	9.1E-09	-	2.9E-09
Kr87	-	-	-
Kr88	-	-	-
Kr89	-	-	-
Xe131m	4.4E-02	-	6.8E-20
Xe133m	1.6E-03	-	-
Xe133	1.4E-01	-	-
Xe135m	1.8E-06	-	-
Xe135	1.5E-04	-	-
Xe137	-	-	-
Xe138	-	-	-
Br82	9.1E-05	1.7E-05	-
Br83	6.1E-12	6.6E-06	-
Br84	-	6.8E-07	-
Br85	-	7.4E-09	-
I129	4.5E-04	9.6E-08	1.6E-04
I130	1.6E-05	4.8E-05	-
I131	1.7E+00	3.0E-02	-
I132	1.6E-02	5.7E-04	-
I133	9.5E-03	3.2E-03	-
I134	6.4E-09	2.4E-05	-
I135	1.1E-05	6.3E-04	-
Rb86m	3.9E-15	2.0E-11	-
Rb86	4.1E-02	3.1E-03	9.0E-14
Rb88	1.2E-06	3.5E-04	-
Rb89	3.9E-08	1.4E-05	-
Cs132	9.1E-05	2.1E-05	-
Cs134	2.0E+03	8.7E+00	3.7E+02
Cs135m	7.3E-09	7.2E-07	-
Cs136	6.2E-01	6.8E-02	2.9E-17
Cs137	1.8E+03	6.0E+00	6.0E+02
Cs138	1.6E-06	2.4E-04	-
P32	7.8E-08	8.7E-10	-
Co57	1.7E-07	1.1E-10	1.0E-08
Sr89	4.8E-02	1.5E-04	1.2E-06
Sr90	7.2E-01	2.1E-04	2.4E-01
Sr91	7.2E-08	5.8E-07	-
Sr92	7.9E-10	8.6E-08	-
Y90	7.2E-01	9.4E-05	2.4E-01
Y91m	4.6E-08	1.8E-07	-
Y91	9.4E-03	2.4E-05	9.0E-07
Y92	3.1E-09	1.3E-07	-
Y93	2.0E-08	1.3E-07	-

Table 12.2-18: Solid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	Spent Resin Storage Tank (Ci)	Phase Separator Tank (Ci)	HIC from SRST (Ci/HIC)
Zr97	2.3E-07	3.2E-07	-
Nb95	1.6E+00	1.0E-03	4.8E-04
Mo99	2.6E-02	2.3E-03	-
Mo101	2.7E-10	3.2E-07	-
Tc99m	2.5E-02	1.1E-03	-
Tc99	2.7E-02	7.9E-06	9.7E-03
Ru103	8.1E-03	3.1E-05	1.4E-08
Ru105	7.8E-10	4.8E-08	-
Ru106	2.6E-01	1.4E-04	2.5E-02
Rh103m	8.0E-03	1.4E-05	1.4E-08
Rh105	3.5E-06	8.1E-07	-
Rh106	2.6E-01	6.1E-05	2.5E-02
Ag110	4.6E-01	1.2E-04	2.4E-02
Sb124	2.9E-05	7.1E-08	3.4E-09
Sb125	9.5E-03	3.5E-06	2.1E-03
Sb127	3.2E-06	1.7E-07	-
Sb129	1.6E-10	1.0E-08	-
Te125m	3.0E-02	7.4E-05	5.2E-04
Te127m	3.8E-01	5.2E-04	1.6E-03
Te127	3.7E-01	2.3E-04	1.6E-03
Te129m	1.0E-01	4.7E-04	2.2E-08
Te129	6.5E-02	1.3E-04	1.4E-08
Te131m	1.7E-04	5.7E-05	-
Te131	3.8E-05	5.9E-06	-
Te132	1.5E-02	1.1E-03	-
Te133m	3.5E-09	1.1E-06	-
Te134	2.8E-09	1.2E-06	-
Ba137m	1.7E+03	4.6E+00	5.7E+02
Ba139	2.1E-10	4.3E-08	-
Ba140	4.1E-03	5.2E-05	6.0E-20
La140	4.7E-03	2.5E-05	6.9E-20
La141	5.2E-10	3.8E-08	-
La142	3.7E-11	7.0E-09	-
Ce141	4.4E-03	2.0E-05	5.8E-10
Ce143	2.3E-06	6.5E-07	-
Ce144	2.0E-01	1.3E-04	1.3E-02
Pr143	6.9E-04	7.8E-06	9.9E-20
Pr144	2.0E-01	5.5E-05	1.3E-02
Np239	2.1E-04	2.3E-05	-
Na24	1.3E-03	2.5E-03	-
Cr51	2.3E+00	3.6E-02	2.3E-08
Mn54	5.5E+01	1.8E-01	4.2E+00
Fe55	1.0E+02	2.1E-01	2.2E+01
Fe59	4.7E-01	5.7E-03	3.4E-06
Co58	5.7E+01	1.2E+00	2.3E-02

Table 12.2-18: Solid Radioactive Waste System Component Source Terms - Radionuclide Content (Continued)

Isotope	Spent Resin Storage Tank (Ci)	Phase Separator Tank (Ci)	HIC from SRST (Ci/HIC)
Co60	5.5E+01	1.0E-01	1.5E+01
Ni63	3.5E+01	5.9E-02	1.2E+01
Zn65	1.3E+01	4.8E-02	6.4E-01
Zr95	1.1E+00	1.1E-02	2.2E-04
Ag110m	3.4E+01	1.2E-01	1.8E+00
W187	8.3E-03	4.1E-04	-
H3	-	-	-
C14	-	-	-
N16	-	-	-
Ar41	-	-	-
Total	**5.8E+03**	**2.1E+01**	**1.6E+03**

Table 12.2-19: Solid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Energy (MeV)		Spent Resin Storage Tank	Phase Separator Tank	High Integrity Container
	Lower Bound	Upper Bound	photons/sec	photons/sec	photons/sec
1	1.00E-02	2.00E-02	3.9E+11	1.6E+09	1.0E+11
2	2.00E-02	3.00E-02	2.3E+11	9.5E+08	5.3E+10
3	3.00E-02	4.50E-02	4.6E+12	1.4E+10	1.5E+12
4	4.50E-02	6.00E-02	1.0E+11	4.3E+08	2.7E+10
5	6.00E-02	7.00E-02	4.8E+10	3.2E+08	1.2E+10
6	7.00E-02	7.50E-02	1.9E+10	8.2E+07	5.0E+09
7	7.50E-02	1.00E-01	7.4E+10	4.5E+08	1.8E+10
8	1.00E-01	1.50E-01	6.4E+10	3.2E+08	1.5E+10
9	1.50E-01	2.00E-01	3.5E+10	5.4E+08	7.6E+09
10	2.00E-01	2.60E-01	3.6E+10	1.9E+08	7.2E+09
11	2.60E-01	3.00E-01	1.3E+10	3.7E+08	1.5E+09
12	3.00E-01	4.00E-01	9.0E+10	2.2E+09	4.0E+09
13	4.00E-01	4.50E-01	4.9E+10	1.9E+08	2.8E+09
14	4.50E-01	5.10E-01	1.1E+12	4.8E+09	2.0E+11
15	5.10E-01	5.12E-01	6.4E+11	1.3E+10	1.1E+09
16	5.12E-01	6.00E-01	1.8E+13	7.8E+10	3.3E+12
17	6.00E-01	7.00E-01	1.3E+14	4.5E+11	3.2E+13
18	7.00E-01	8.00E-01	7.1E+13	3.1E+11	1.3E+13
19	8.00E-01	9.00E-01	8.1E+12	6.8E+10	7.7E+11
20	9.00E-01	1.00E+00	4.3E+11	1.6E+09	2.2E+10
21	1.00E+00	1.20E+00	4.5E+12	1.6E+10	1.0E+12
22	1.20E+00	1.33E+00	1.1E+12	2.6E+09	3.0E+11
23	1.33E+00	1.44E+00	3.5E+12	1.3E+10	7.0E+11
24	1.44E+00	1.50E+00	5.3E+10	1.9E+08	2.7E+09
25	1.50E+00	1.57E+00	1.8E+11	6.7E+08	9.5E+09
26	1.57E+00	1.66E+00	4.7E+08	2.1E+06	1.6E+07
27	1.66E+00	1.80E+00	1.1E+10	2.3E+08	1.6E+07
28	1.80E+00	2.00E+00	2.3E+08	4.4E+06	1.2E+07
29	2.00E+00	2.15E+00	2.2E+07	5.8E+05	1.1E+06
30	2.15E+00	2.35E+00	7.8E+07	1.7E+06	1.0E+07
31	2.35E+00	2.50E+00	7.8E+06	3.1E+05	5.6E+05
32	2.50E+00	2.75E+00	3.3E+07	4.9E+07	1.2E+05
33	2.75E+00	3.00E+00	2.4E+07	4.4E+07	4.6E+04
34	3.00E+00	3.50E+00	2.2E+05	1.1E+05	1.6E+04
35	3.50E+00	4.00E+00	3.7E+04	9.0E+04	6.8E+00
36	4.00E+00	4.50E+00	4.3E+02	4.8E+03	-
37	4.50E+00	5.00E+00	8.3E+01	2.4E+04	-
38	5.00E+00	5.50E+00	1.6E-01	4.6E+01	-
39	5.50E+00	6.00E+00	-	-	-
40	6.00E+00	6.50E+00	-	-	-
41	6.50E+00	7.00E+00	-	-	-
42	7.00E+00	7.50E+00	-	-	-
43	7.50E+00	8.00E+00	-	-	-
44	8.00E+00	1.00E+01	-	-	-

Table 12.2-19: Solid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy (MeV)		Spent Resin Storage Tank	Phase Separator Tank	High Integrity Container
	Lower Bound	Upper Bound	photons/sec	photons/sec	photons/sec
45	1.00E+01	1.20E+01	-	-	-
46	1.20E+01	1.40E+01	-	-	-
47	1.40E+01	2.00E+01	-	-	-
Total			**2.4E+14**	**9.8E+11**	**5.3E+13**

Table 12.2-20: Spent Fuel Gamma Source Strength

Energy Group	Energy (MeV)		Single Spent Fuel Assembly (48 hr decay time)	Full Fuel Storage Rack (48 hour decay time for all 660 Assemblies)
	Lower Bound	Upper Bound	photons/sec	photons/sec
1	1.00E-02	2.00E-02	7.8E+16	5.1E+19
2	2.00E-02	3.00E-02	2.2E+16	1.4E+19
3	3.00E-02	4.50E-02	2.4E+16	1.6E+19
4	4.50E-02	6.00E-02	1.0E+16	6.9E+18
5	6.00E-02	7.00E-02	4.2E+15	2.8E+18
6	7.00E-02	7.50E-02	1.1E+15	7.0E+17
7	7.50E-02	1.00E-01	1.9E+16	1.3E+19
8	1.00E-01	1.50E-01	8.1E+16	5.4E+19
9	1.50E-01	2.00E-01	4.3E+15	2.8E+18
10	2.00E-01	2.60E-01	2.6E+16	1.7E+19
11	2.60E-01	3.00E-01	2.0E+16	1.3E+19
12	3.00E-01	4.00E-01	1.5E+16	1.0E+19
13	4.00E-01	4.50E-01	1.8E+15	1.2E+18
14	4.50E-01	5.10E-01	1.9E+16	1.3E+19
15	5.10E-01	5.12E-01	1.8E+15	1.2E+18
16	5.12E-01	6.00E-01	8.6E+15	5.7E+18
17	6.00E-01	7.00E-01	1.9E+16	1.2E+19
18	7.00E-01	8.00E-01	3.5E+16	2.3E+19
19	8.00E-01	9.00E-01	5.9E+15	3.9E+18
20	9.00E-01	1.00E+00	3.9E+15	2.6E+18
21	1.00E+00	1.20E+00	3.6E+15	2.4E+18
22	1.20E+00	1.33E+00	1.6E+15	1.0E+18
23	1.33E+00	1.44E+00	9.0E+14	5.9E+17
24	1.44E+00	1.50E+00	4.0E+14	2.6E+17
25	1.50E+00	1.57E+00	4.6E+13	3.0E+16
26	1.57E+00	1.66E+00	1.1E+16	7.2E+18
27	1.66E+00	1.80E+00	1.1E+14	7.5E+16
28	1.80E+00	2.00E+00	5.4E+14	3.5E+17
29	2.00E+00	2.15E+00	4.1E+14	2.7E+17
30	2.15E+00	2.35E+00	4.9E+14	3.2E+17
31	2.35E+00	2.50E+00	7.3E+13	4.8E+16
32	2.50E+00	2.75E+00	4.0E+14	2.6E+17
33	2.75E+00	3.00E+00	9.1E+12	6.0E+15
34	3.00E+00	3.50E+00	3.4E+12	2.2E+15
35	3.50E+00	4.00E+00	3.9E+08	2.6E+11
36	4.00E+00	4.50E+00	5.5E+07	3.6E+10
37	4.50E+00	5.00E+00	9.1E+07	6.0E+10
38	5.00E+00	5.50E+00	1.5E+07	1.0E+10
39	5.50E+00	6.00E+00	8.8E+06	5.8E+09
40	6.00E+00	6.50E+00	5.1E+06	3.3E+09
41	6.50E+00	7.00E+00	2.9E+06	1.9E+09
42	7.00E+00	7.50E+00	1.7E+06	1.1E+09
43	7.50E+00	8.00E+00	9.9E+05	6.5E+08

Table 12.2-20: Spent Fuel Gamma Source Strength (Continued)

Energy Group	Energy (MeV)		Single Spent Fuel Assembly (48 hr decay time)	Full Fuel Storage Rack (48 hour decay time for all 660 Assemblies)
	Lower Bound	Upper Bound	photons/sec	photons/sec
44	8.00E+00	1.00E+01	1.2E+06	7.7E+08
45	1.00E+01	1.20E+01	6.0E+04	4.0E+07
46	1.20E+01	1.40E+01	-	-
47	1.40E+01	2.00E+01	-	-
Total			**4.2E+17**	**2.8E+20**

Table 12.2-21: Spent Fuel Neutron Energy Spectrum

Energy Group	Energy (MeV)		Single Spent Fuel Assembly (48 hr decay time)	Full Fuel Storage Rack (48 hour decay time for all 660 Assemblies)
	Lower Bound	Upper Bound	neutrons/sec	neutrons/sec
1	1.0E-11	1.0E-08	1.2E-04	8.0E-02
2	1.0E-08	3.0E-08	1.4E-04	9.1E-02
3	3.0E-08	5.0E-08	1.0E-02	6.7E+00
4	5.0E-08	1.0E-07	7.8E-03	5.2E+00
5	1.0E-07	2.3E-07	6.6E-02	4.4E+01
6	2.3E-07	3.3E-07	6.0E-02	3.9E+01
7	3.3E-07	4.1E-07	4.4E-02	2.9E+01
8	4.1E-07	8.0E-07	3.6E-01	2.3E+02
9	8.0E-07	1.0E-06	1.8E-01	1.2E+02
10	1.0E-06	1.1E-06	1.8E-01	1.2E+02
11	1.1E-06	1.3E-06	2.2E-01	1.4E+02
12	1.3E-06	1.9E-06	7.8E-01	5.2E+02
13	1.9E-06	3.1E-06	2.1E+00	1.4E+03
14	3.1E-06	1.1E-05	2.2E+01	1.5E+04
15	1.1E-05	2.9E-05	9.1E+01	6.0E+04
16	2.9E-05	1.0E-04	6.5E+02	4.3E+05
17	1.0E-04	5.8E-04	9.8E+03	6.4E+06
18	5.8E-04	3.0E-03	1.1E+05	7.6E+07
19	3.0E-03	1.5E-02	1.2E+06	8.2E+08
20	1.5E-02	1.1E-01	2.5E+07	1.7E+10
21	1.1E-01	4.1E-01	1.5E+08	9.6E+10
22	4.1E-01	9.1E-01	3.2E+08	2.1E+11
23	9.1E-01	1.4E+00	3.2E+08	2.1E+11
24	1.4E+00	1.8E+00	2.3E+08	1.5E+11
25	1.8E+00	3.0E+00	4.8E+08	3.1E+11
26	3.0E+00	6.4E+00	4.2E+08	2.7E+11
27	6.4E+00	2.0E+01	4.3E+07	2.8E+10
Total			**2.0E+09**	**1.3E+12**

Table 12.2-22: In-Core Instrument Source Term Input Assumptions

Component (quantity)	Material
Emitter (4)	Rh-103
Signal wire (4)	316LSS
Insulation(1)	Al_2O_3
Outer sheath (1)	316LSS
Inner sheath (6)	316LSS
Thermocouple-type K chromel-alumel (2)	Chromel
	Alumel
Parameter	**Value**
Number of irradiation cycles	1-40
Neutron flux	3.1E+14n/cm^2-sec

Table 12.2-23: In-Core Instrumentation Gamma Spectra

Energy Group	Energy Lower Bound (MeV)	Energy Upper Bound (MeV)	Cycle 1	Cycle 40
			Discharge (photons/sec)	3-Day Decay (photons/sec)
1	2.00E-02	3.50E-02	2.5E+13	1.1E+11
2	3.50E-02	5.00E-02	1.2E+13	5.9E+10
3	5.00E-02	7.50E-02	1.4E+13	4.5E+10
4	7.50E-02	1.25E-01	1.4E+13	9.1E+10
5	1.25E-01	1.75E-01	6.2E+12	5.3E+11
6	1.75E-01	2.50E-01	4.7E+12	8.0E+10
7	2.50E-01	4.00E-01	8.2E+12	6.4E+11
8	4.00E-01	9.00E-01	5.6E+13	3.6E+12
9	9.00E-01	1.35E+00	3.6E+12	6.8E+12
10	1.35E+00	1.80E+00	8.0E+12	9.9E+09
11	1.80E+00	2.20E+00	9.8E+12	1.7E+08
12	2.20E+00	2.60E+00	3.8E+11	1.6E+06
13	2.60E+00	3.00E+00	3.6E+11	8.0E+08
14	3.00E+00	3.50E+00	6.2E+10	1.9E+06
15	3.50E+00	4.00E+00	4.3E+07	6.5E+05
16	4.00E+00	4.50E+00	7.2E+06	6.8E+03
17	4.50E+00	5.00E+00	5.5E+06	-
18	5.00E+00	1.00E+01	2.6E+09	-
Total			1.6E+14	1.2E+13

Table 12.2-24: Control Rod Assembly Tip Source Term Input Assumptions

Parameter	Value
Neutron flux at control rod tip	5.9E+13n/cm^2-sec
CRA tip irradiation duration	1-40 cycles
Control Rod Assembly component	**Material**
Absorber	Ag-In-Cd
Stack support	X-750
Cladding	304 SS
Lower end plug	308 SS

Table 12.2-25: Control Rod Assembly Tip Gamma Spectra (End of Cycle 1)

Energy Group	Energy Lower Bound (MeV)	Energy Upper Bound (MeV)	Cycle 1		
			0-Day Decay	3-Day Decay	30-Day Decay
			Photons/sec-kg		
1	2.00E-02	3.50E-02	1.3E+15	1.0E+13	6.8E+12
2	3.50E-02	5.00E-02	7.3E+14	2.3E+12	1.2E+12
3	5.00E-02	7.50E-02	6.8E+14	2.1E+12	1.1E+12
4	7.50E-02	1.25E-01	8.1E+14	2.2E+12	1.2E+12
5	1.25E-01	1.75E-01	3.6E+14	9.1E+11	5.1E+11
6	1.75E-01	2.50E-01	2.8E+14	2.5E+12	1.4E+12
7	2.50E-01	4.00E-01	4.3E+14	4.6E+12	1.5E+12
8	4.00E-01	9.00E-01	1.4E+15	4.6E+14	4.3E+14
9	9.00E-01	1.35E+00	1.9E+15	4.9E+13	4.5E+13
10	1.35E+00	1.80E+00	2.2E+14	6.6E+13	6.2E+13
11	1.80E+00	2.20E+00	1.9E+14	2.9E+10	2.7E+10
12	2.20E+00	2.60E+00	1.2E+12	1.2E+08	1.2E+08
13	2.60E+00	3.00E+00	2.5E+11	5.6E+06	2.8E+06
14	3.00E+00	3.50E+00	3.9E+10	3.4E+05	2.5E+05
15	3.50E+00	4.00E+00	1.3E+06	6.4E+03	3.6E+03
16	4.00E+00	4.50E+00	7.4E+04	2.0E+01	2.0E-12
17	4.50E+00	5.00E+00	9.3E+01	0.0E+00	0.0E+00
18	5.00E+00	1.00E+01	6.3E+01	0.0E+00	0.0E+00
Total			8.2E+15	6.0E+14	5.5E+14

Table 12.2-26: Secondary Neutron Source Gamma Spectra (End of Cycle 1)

Energy Group	Energy Lower Bound (MeV)	Energy Upper Bound (MeV)	Cycle 1		
			0-Day Decay	3-Day Decay	30-Day Decay
			Photons/sec-kg		
1	2.00E-02	3.50E-02	2.8E+15	1.2E+15	3.9E+14
2	3.50E-02	5.00E-02	9.9E+14	5.5E+14	1.7E+14
3	5.00E-02	7.50E-02	1.5E+15	5.0E+14	1.5E+14
4	7.50E-02	1.25E-01	1.1E+15	5.5E+14	1.7E+14
5	1.25E-01	1.75E-01	4.5E+14	2.8E+14	1.2E+14
6	1.75E-01	2.50E-01	3.2E+14	1.7E+14	5.3E+13
7	2.50E-01	4.00E-01	4.6E+14	2.6E+14	9.2E+13
8	4.00E-01	9.00E-01	1.9E+16	1.3E+16	6.2E+15
9	9.00E-01	1.35E+00	7.7E+14	6.1E+14	3.8E+14
10	1.35E+00	1.80E+00	4.3E+15	4.1E+15	3.0E+15
11	1.80E+00	2.20E+00	6.3E+14	4.4E+14	3.2E+14
12	2.20E+00	2.60E+00	9.9E+12	3.0E+12	2.2E+12
13	2.60E+00	3.00E+00	6.5E+12	4.3E+11	3.1E+11
14	3.00E+00	3.50E+00	1.1E+12	4.7E+07	3.5E+07
15	3.50E+00	4.00E+00	1.1E+10	7.9E+05	5.0E+05
16	4.00E+00	4.50E+00	4.0E+09	1.7E+03	1.6E-10
17	4.50E+00	5.00E+00	3.9E+09	0.0E+00	0.0E+00
18	5.00E+00	1.00E+01	8.2E+09	0.0E+00	0.0E+00
Total			3.2E+16	2.1E+16	1.1E+16

Table 12.2-27: Post-Accident Equipment Qualification Source Term Input Assumptions

Parameter	Value
Containment release delay	0
Containment release duration	1E-05 hr
Containment leak rate	0.2 %/day
Containment leak rate after 24 hours	0.1 %/day
Aerosol fraction of non-noble gases released	100%
Bioshield envelope volume	1.3E+04 ft^3
Primary coolant water density	44.5 lbm/ft^3
Air density	0.07 lbm/ft^3
Containment air volume	3.7E+03 ft^3
Combined water volume	2.3E+03 ft^3

Table 12.2-28: Post-Accident Integrated Energy Deposition and Integrated Dose

Volume	Medium	Time	Integrated Dose (Rad)
Reactor and containment	Water	1 hour	6.7E+02
		36 hours	6.0E+03
		3 days	7.7E+03
		30 days	1.4E+04
		100 days	1.5E+04
Containment vessel	Air	1 hour	2.6E+05
		36 hours	2.4E+06
		3 days	3.2E+06
		30 days	6.9E+06
		100 days	1.0E+07
Bioshield envelope	Air and water	1 hour	3.2E+01
		36 hours	7.8E+02
		3 days	1.4E+03
		30 days	1.3E+04
		100 days	5.6E+04

Table 12.2-29: Input Parameters for Determining Facility Airborne Concentrations

Parameter	Value
Primary coolant leak rate	160 lbm/day-NPM
Flash fraction of primary coolant leaks	40%
Gas release from primary coolant leaks	100%
Partition coefficients for evaporation and leaks: • Noble gases and tritium • Halogens • Particulates • Iodines (pool evaporation only)	Table 11.3-4
Primary coolant source term	Table 11.1-4
Pool water source term	Table 12.2-9
Pool evaporation rate: • Pool surface water temperature • Area of pool water surface • Air velocity over water surface • Room air temperature • Room air relative humidity	1300 lb/hr 100 degrees F 9300 ft^2 30 ft/min 73 degrees F 68%
CVCS pump valve room leak	3 lbm/day
Degasifier room leak	8 lbm/day
Normal ventilation air change rates in RXB: • Pool air space (100'elevation) • CVCS pump/valve rooms (35'-8"elevation) • Degasifier rooms (24' elevation)	 1 air/change-hr 2 air/change-hr 2 air/change-hr
Pool air space volume	2.66E+10 ml
CVCS pump valve room volume	1.20E+08 ml
Degasifier room volume	2.55E+08 ml

Table 12.2-30: Reactor Building Airborne Concentrations

Radionuclide	CVCS Pump Room Airborne Concentration	LWRS Degasifier Room Airborne Concentration	Air Space above Reactor Pool Airborne Concentration
	(µCi/ml)	(µCi/ml)	(µCi/ml)
Kr83m	2.0E-09	2.0E-09	5.1E-15
Kr85m	9.3E-09	9.3E-09	-
Kr85	1.8E-06	1.8E-06	-
Kr87	4.3E-09	4.3E-09	-
Kr88	1.4E-08	1.4E-08	-
Kr89	4.8E-11	4.8E-11	-
Xe131m	4.2E-08	4.2E-08	1.2E-08
Xe133m	3.6E-08	3.6E-08	1.4E-08
Xe133	2.7E-06	2.7E-06	2.1E-07
Xe135m	1.5E-09	1.5E-09	1.9E-09
Xe135	6.9E-08	6.9E-08	1.0E-09
Xe137	1.8E-10	1.8E-10	-
Xe138	1.6E-09	1.6E-09	-
Br82	2.6E-13	2.6E-13	2.2E-16
Br83	1.3E-12	1.3E-12	3.7E-22
Br84	4.2E-13	4.2E-13	-
Br85	1.0E-14	1.0E-14	-
I129	4.3E-18	4.3E-18	1.1E-20
I130	2.1E-12	2.1E-12	2.3E-16
I131	5.4E-11	5.4E-11	1.1E-11
I132	2.2E-11	2.2E-11	2.7E-14
I133	8.2E-11	8.2E-11	9.6E-13
I134	1.1E-11	1.1E-11	-
I135	5.0E-11	5.0E-11	3.6E-16
Rb86m	1.5E-18	1.5E-18	-
Rb86	1.9E-13	1.9E-13	2.9E-14
Rb88	7.6E-09	7.6E-09	-
Rb89	2.8E-11	2.8E-11	-
Cs132	3.7E-15	3.7E-15	4.8E-16
Cs134	2.7E-11	2.7E-11	4.5E-12
Cs135m	1.6E-14	1.6E-14	-
Cs136	5.8E-12	5.8E-12	8.6E-13
Cs137	1.4E-11	1.4E-11	2.3E-12
Cs138	6.4E-10	6.4E-10	-
P32	5.2E-19	5.2E-19	2.4E-20
Co57	3.9E-21	3.9E-21	2.0E-22
Sr89	3.1E-14	3.1E-14	1.2E-15
Sr90	3.7E-15	3.7E-15	1.9E-16
Sr91	1.2E-14	1.2E-14	1.1E-17
Sr92	5.8E-15	5.8E-15	1.6E-22
Y90	9.1E-16	9.1E-16	1.1E-16
Y91m	6.6E-15	6.6E-15	7.1E-18
Y91	3.4E-15	3.4E-15	1.7E-16
Y92	5.6E-15	5.6E-15	2.2E-20

Table 12.2-30: Reactor Building Airborne Concentrations (Continued)

Radionuclide	CVCS Pump Room Airborne Concentration (µCi/ml)	LWRS Degasifier Room Airborne Concentration (µCi/ml)	Air Space above Reactor Pool Airborne Concentration (µCi/ml)
Y93	2.5E-15	2.5E-15	3.0E-18
Zr97	3.8E-15	3.8E-15	2.0E-17
Nb95	5.6E-15	5.6E-15	1.4E-13
Mo99	7.0E-12	7.0E-12	2.0E-13
Mo101	1.1E-13	1.1E-13	-
Tc99m	6.5E-12	6.5E-12	1.9E-13
Tc99	1.3E-16	1.3E-16	6.8E-18
Ru103	6.7E-15	6.7E-15	3.3E-16
Ru105	2.1E-15	2.1E-15	1.8E-20
Ru106	4.2E-15	4.2E-15	2.1E-16
Rh103m	6.6E-15	6.6E-15	3.2E-16
Rh105	4.5E-15	4.5E-15	8.3E-17
Rh106	4.2E-15	4.2E-15	-
Ag110	2.7E-15	2.7E-15	-
Sb124	1.0E-17	1.0E-17	5.0E-19
Sb125	7.4E-17	7.4E-17	3.8E-18
Sb127	3.7E-16	3.7E-16	1.3E-17
Sb129	4.3E-16	4.3E-16	3.6E-21
Te125m	1.1E-14	1.1E-14	5.4E-16
Te127m	4.1E-14	4.1E-14	2.1E-15
Te127	1.6E-13	1.6E-13	2.1E-15
Te129m	1.2E-13	1.2E-13	5.7E-15
Te129	1.5E-13	1.5E-13	3.6E-15
Te131m	3.8E-13	3.8E-13	5.4E-15
Te131	1.4E-13	1.4E-13	1.2E-15
Te132	2.8E-12	2.8E-12	8.7E-14
Te133m	1.8E-13	1.8E-13	-
Te134	2.3E-13	2.3E-13	-
Ba137m	1.3E-11	1.3E-11	-
Ba139	5.1E-15	5.1E-15	2.4E-28
Ba140	3.4E-14	3.4E-14	1.6E-15
La140	1.0E-14	1.0E-14	1.2E-15
La141	1.8E-15	1.8E-15	5.1E-21
La142	7.7E-16	7.7E-16	4.1E-28
Ce141	5.3E-15	5.3E-15	2.6E-16
Ce143	4.0E-15	4.0E-15	6.3E-17
Ce144	4.5E-15	4.5E-15	2.3E-16
Pr143	4.7E-15	4.7E-15	2.3E-16
Pr144	4.4E-15	4.4E-15	2.3E-16
Np239	8.4E-14	8.4E-14	2.2E-15
Na24	8.3E-12	8.3E-12	3.2E-14
Cr51	4.8E-13	4.8E-13	2.3E-11
Mn54	2.4E-13	2.4E-13	1.2E-11
Fe55	1.8E-13	1.8E-13	9.4E-12

Table 12.2-30: Reactor Building Airborne Concentrations (Continued)

Radionuclide	CVCS Pump Room Airborne Concentration	LWRS Degasifier Room Airborne Concentration	Air Space above Reactor Pool Airborne Concentration
	(μCi/ml)	(μCi/ml)	(μCi/ml)
Fe59	4.6E-14	4.6E-14	2.3E-12
Co58	7.0E-13	7.0E-13	3.5E-10
Co60	8.1E-14	8.1E-14	4.1E-12
Ni63	4.0E-14	4.0E-14	2.1E-12
Zn65	7.8E-14	7.8E-14	4.0E-12
Zr95	6.0E-14	6.0E-14	3.0E-12
Ag110m	2.0E-13	2.0E-13	1.0E-11
W187	4.2E-13	4.2E-13	4.3E-12
H3	3.2E-07	3.2E-07	2.3E-06
C14	3.2E-11	3.2E-11	1.9E-11
N16	-	-	-
Ar41	5.5E-08	5.5E-08	-

Table 12.2-31: Maximum Post-Accident Radionuclide Concentrations

Radionuclide	RCS Peak Concentration (µCi/g)
Kr83m	4.2E-01
Kr85m	4.5E-02
Kr85	5.9E+00
Kr87	1.8E-02
Kr88	5.1E-02
Kr89	1.2E-03
Xe131m	1.4E-01
Xe133m	2.1E-01
Xe133	1.0E+01
Xe135m	2.7E+00
Xe135	9.2E+00
Xe137	3.9E-03
Xe138	1.3E-02
Br82	1.1E-01
Br83	1.9E+00
Br84	3.2E+00
Br85	4.0E+00
I129	1.5E-06
I130	1.1E+00
I131	2.0E+01
I132	3.4E+01
I133	3.8E+01
I134	4.4E+01
I135	3.6E+01
Rb86m	5.2E-08
Rb86	3.0E-04
Rb88	6.0E-02
Rb89	2.4E-03
Cs132	6.0E-06
Cs134	4.3E-02
Cs135m	3.6E-05
Cs136	9.4E-03
Cs137	2.2E-02
Cs138	1.9E-02
P32	8.4E-10
Co57	6.4E-12
Sr89	3.9E-05
Sr90	6.0E-06
Sr91	2.0E-05
Sr92	1.1E-05
Y90	1.9E-06
Y91m	1.8E-05
Y91	5.6E-06
Y92	1.2E-05
Y93	4.3E-06
Zr97	6.3E-06
Nb95	9.7E-06

Table 12.2-31: Maximum Post-Accident Radionuclide Concentrations (Continued)

Radionuclide	RCS Peak Concentration (µCi/g)
Mo99	1.1E-02
Mo101	4.3E-04
Tc99m	1.6E-02
Tc99	2.1E-07
Ru103	1.1E-05
Ru105	3.6E-06
Ru106	6.8E-06
Rh103m	2.1E-05
Rh105	7.7E-06
Rh106	1.4E-05
Ag110	1.2E-05
Sb124	1.6E-08
Sb125	1.2E-07
Sb127	6.1E-07
Sb129	7.6E-07
Te125m	1.8E-05
Te127m	6.7E-05
Te127	2.9E-04
Te129m	1.9E-04
Te129	3.9E-04
Te131m	6.2E-04
Te131	4.3E-04
Te132	4.6E-03
Te133m	3.9E-04
Te134	5.6E-04
Ba137m	4.2E-02
Ba139	1.0E-05
Ba140	5.6E-05
La140	2.3E-05
La141	3.2E-06
La142	1.5E-06
Ce141	8.6E-06
Ce143	6.5E-06
Ce144	7.3E-06
Pr143	7.7E-06
Pr144	1.4E-05
Np239	1.4E-04
Na24	1.4E-02
Cr51	7.7E-04
Mn54	4.0E-04
Fe55	3.0E-04
Fe59	7.5E-05
Co58	1.1E-03
Co60	1.3E-04
Ni63	6.6E-05
Zn65	1.3E-04
Zr95	9.7E-05

Table 12.2-31: Maximum Post-Accident Radionuclide Concentrations (Continued)

Radionuclide	RCS Peak Concentration (μCi/g)
Ag110m	3.2E-04
W187	7.0E-04
H3	2.6E+00
C14	2.6E-04
N16	-
Ar41	1.0E-02

12.3 Radiation Protection Design Features

This section describes the incorporation of radiation protection design features in the NuScale Power Plant US460 standard design facilities. These include features to reduce both onsite and offsite exposures and to protect the environment. The facility incorporates design features using the guidance of U.S. Nuclear Regulatory Commission and industry documents (e.g., Regulatory Guide (RG) 8.8, RG 4.21, and NEI 08-08A) to ensure compliance with applicable regulations such as 10 CFR 20.1101, 10 CFR 20.1406 and 10 CFR 50.34.

12.3.1 Facility Design Features

The following discussion contains specific system and facility design features that implement as low as reasonably achievable (ALARA) principles to the NuScale plant design. The design engineers incorporate these design features during the design process using applicable operating experience.

12.3.1.1 Equipment Design

This section provides specific design features for component types that aid in maintaining occupational exposures ALARA.

12.3.1.1.1 Tanks

The use of tanks with bottoms that slope toward outlets and, where practicable, providing built in spray features, spargers and eductors for mixing tank contents reduce radiation sources from sedimentation in tanks.

The tanks in the liquid radioactive waste (LRWS) and solid radioactive waste systems (SRWS) are stainless steel with sloped bottoms. The LRWS collection tanks and the solid radioactive waste storage tanks contain mixing eductors or spargers.

Tanks that are expected to contain radioactive contaminated fluids are of welded construction with a smooth interior finish that minimizes crevices and crud traps.

Tanks have overflow lines routed to receiving tanks, sumps or drains. Tank level alarms protect against overflow situations. The vents and drains associated with tanks containing contaminated fluids are processed by the building ventilation and the radioactive waste drain systems, respectively.

Tank materials are designed to be compatible with the service environment to reduce corrosion and leaks.

The pool cooling and cleanup system (PCWS) storage tank is located in a steel-lined catch basin with sufficient capacity to contain the tank volume along with its associated piping.

12.3.1.1.2 Valves

Remotely actuated valves are used to minimize personnel exposures, where practicable. Many valves are also located in valve galleries to provide additional shielding.

Double isolation valves are used at the interface between contaminated and non-contaminated systems to prevent cross-contamination.

Valves are designed to fail to the safe position upon a loss of power or air to the valve operator.

Full-port ball valves are used in the SRWS and the PCWS slurry lines to reduce potential crud traps.

Relief valves are provided to protect equipment and the relief discharge is directed toward the radioactive waste drain system (RWDS) to minimize the spread of contamination.

Reach-rods are used for valves in some applications.

Where practicable, valves are installed in the "stem-up" orientation.

Valves are designed to be repacked without removing the yoke or topworks.

12.3.1.1.3 Piping

Construction and fabrication techniques employed by the following systems are used to exclude crevices and crud traps: LRWS, SRWS, PCWS, chemical and volume control system (CVCS), and radioactive waste drain system (RWDS). The SRWS deploys sloping, bends, and long radius elbows to minimize crud traps.

The LRWS and SRWS piping is provided with clean-in-place and flushing capabilities to reduce the buildup of crud and other contaminates. The CVCS and the PCWS have flushing connections to aid in the removal of contamination and reduce potential exposures to plant personnel.

Piping is designed for the lifetime of the facility.

System piping that contains radioactive fluids uses welded construction and smooth internal surfaces, as practicable. Whenever practicable, horizontal resin sluice lines are sloped to facilitate draining and prevent potential hot spots.

Embedded or underground piping is limited to the extent practical. Underground pipes containing radioactive liquids, such as LRWS and PCWS piping, are enclosed within structured pipe chases or are double-walled. The CVCS, the process sampling system (PSS), and resin sluice pipes that are expected to contain highly contaminated fluids are routed through shielded pipe chases, as much as practicable.

12.3.1.1.4 Pumps

Pump leakage is reduced by using canned pumps whenever they are compatible with service needs.

The LRWS uses centrifugal pumps to minimize potential for crud traps.

Where appropriate, the design uses pumps with flanged connections for removal to low dose area for maintenance.

Drain connections on pump casings are provided where appropriate to reduce the radiation field during pump servicing.

12.3.1.1.5 Heat Exchangers

Heat exchangers are designed for complete drainage before maintenance activities.

Flushing connections are provided for heat exchangers, where appropriate.

Heat exchangers use corrosion resistant materials to minimize need for replacement and internal surfaces are smoothed to be free of crevices.

Where practicable, heat exchangers are designed such that the contaminated fluid is on the tube side.

12.3.1.1.6 Instrumentation

Whenever practicable, remote (in low dose areas) instruments and transmitters are used for systems with radioactive fluids.

Instruments in high radiation areas are designed to be easily removable.

Instrument sensing lines containing contaminated fluids are designed with back-flushing capability.

The locations and equipment specifications for seismic monitoring equipment (per RG 1.12) reduce the frequency and duration of testing, inspection or maintenance of seismic monitoring equipment.

12.3.1.1.7 Ventilation

The Reactor Building HVAC system (RBVS) and the Radioactive Waste Building HVAC system (RWBVS) are once-through systems changing air volume from 0.5 to two times per hour for potentially contaminated areas. The design permits convenient inspection, maintenance and decontamination, and facilitates the replacement of critical components such as filters, fans, and dampers. Condensate from heating, ventilation, and air conditioning (HVAC) equipment is routed to the RWDS. Exhaust duct air is exhausted from areas where low levels of airborne contamination may be present toward areas of higher potential contamination. The exhaust ducts consist of straight runs and long radius elbows to reduce the buildup of contaminated particulate. The duct air velocity is kept at sufficiently high velocities to keep particulates suspended. Construction materials have smooth internal and external surface finishes to aid in decontamination. Back draft dampers are provided at each tank and equipment connection where required to prevent blowback in the event of an exhaust system trip.

Section 12.3.3 discusses ventilation systems.

12.3.1.1.8 Floor Drains

Floor drains are provided for rooms and cubicles with components containing radioactive fluids that might leak or be spilled from process equipment or sampling stations.

Drain piping that is shielded by pipe chases, or otherwise shielded to reduce personnel exposure, include leak detection and confinement such that the fluid is contained.

12.3.1.1.9 Filters

Filters for reactor coolant and potentially high-activity applications incorporate shielding, remote handling equipment, and radiation monitoring instrumentation to aid in maintaining personnel exposures ALARA.

Cartridge filter housings are designed with isolation valves, vents, and drains to allow the spent filter to be drained before maintenance activities.

Cartridge filter housings have minimal internal crevices to minimize the buildup of crud.

The design and configuration of filter housings and cartridges are standardized such that the same equipment and procedures can be used to change out spent filters.

12.3.1.1.10 Demineralizers

Resin transfer operations are performed remotely through piping that is routed through a shielded pipe chase. Portions of piping that are not located in a shielded pipe chase are shielded to reduce the potential for worker exposure. If necessary, administrative controls are enacted when high-activity resin transfers are planned to ensure that areas where high radiation may occur during the resin transfer are evacuated of personnel.

Demineralizers are designed and configured to allow for full drainage.

Demineralizers are designed and constructed to minimize internal crevices and crud traps.

12.3.1.1.11 Charcoal Beds

The redundant charcoal bed skids of the gaseous radioactive waste system (GRWS) are located in individual shielded cubicles.

Radiation detectors monitor the charcoal bed cubicles for gas leaks.

12.3.1.1.12 Sample Stations

Radiation protection design features are incorporated into the design of sample stations for radioactive fluid. Primary coolant samples are routed to sample stations to minimize radiation dose from local sample points. Sample stations are shielded and are located in low-dose areas to minimize occupational exposures. Shielding of sample stations for primary coolant is achieved by routing sample lines in shielded pipe chases to the extent practicable. Sampling components that contain potentially radioactive fluids are located in shielded compartments or away from the sample panel to the extent practicable. Reactor coolant grab sample stations are equipped with vent hoods to reduce personnel exposure.

Sample stations contain flushing provisions with drains routed to the LRWS.

The laboratory and counting room are designed to provide low background radiation.

12.3.1.1.13 Material Selection

Proper material selection is an important factor to balance component performance while reducing the amount of corrosion and activation products generated. The use of materials containing cobalt and nickel is minimized to reduce the quantity of activation products. Ni-Cr-Fe alloys, such as Inconel, have a high nickel content that can become Co-58 when activated. Production of Co-58 and Co-60 are reduced by utilizing low nickel and low cobalt bearing materials, to the extent practicable.

In limited situations, the selection of materials containing higher percentages of cobalt (e.g., hard face materials) or nickel (e.g., Alloy 690TT steam generator (SG) tubing) is preferable from a design standpoint. In these cases, the additional generation of activation products is balanced against component reliability to achieve the lowest overall personnel exposure. These types of materials are used only where operating experience suggests that it is the preferred option. Alloy 690TT is the material used for SG tubing because of its high resistance to corrosion.

Systems use stainless steel, or stainless steel clad, for components and piping where corrosion resistance and water quality is an important consideration (e.g., components that can come into contact with primary coolant or the reactor pool water). Radioactive waste system components are largely designed using stainless steel also, following the recommendations of RG 1.143.

The reactor pressure vessel (RPV), control rod drive mechanisms (CRDMs), and containment vessel (CNV) materials are predominately low alloy steels, clad with stainless steel, and austenitic stainless steels. The use of cobalt containing materials in contact with the primary coolant, such as Stellite, is limited to a small number of wear components such as CRDM latches, hard faces, springs and control rod assembly hub connection couplings (Haynes Alloy-25). Table 12.3-3 summarizes the typical cobalt content for materials and components.

12.3.1.2 Plant and Layout Design Features for as Low as Reasonably Achievable

This section provides descriptions and examples of facility design features to reduce personnel exposures in accordance with the guidance of RG 8.8 and the ALARA principle.

12.3.1.2.1 Pipe Routing

Whenever possible, pipes with radioactive fluids are routed through pipe chases or shielded areas, and away from pipes for "clean" services.

If pipes with radioactive fluids are routed near clean service pipes, provisions for isolation and draining of the radioactive pipes are provided.

Piping is designed to minimize "dead legs" and low points.

12.3.1.2.2 Valve Galleries

Valve galleries are provided in several locations to protect plant operators from radiation exposures from process equipment. Floors are sloped towards local drain hubs to collect leakage. Concrete surfaces within the valve galleries are coated to facilitate decontamination.

12.3.1.2.3 Penetrations

Penetrations through shield walls are minimized as much as practicable.

If penetrations through shield walls are necessary, the penetrations are designed to minimize streaming (e.g., with an offset) from a radiation source to accessible areas. If penetration offsets are not practical, then penetrations are either shielded or elevated above floor level. Shield wall penetrations are compensated to comply with the associated radiation zone map dose rates for normally accessible areas.

12.3.1.2.4 Equipment Layout

Radioactive system components are located separately from "clean" components as much as practicable. Individual components of a radioactive system are typically located in separate shielded compartments with short piping runs between components. Where appropriate, shielded valve galleries are employed to allow system operation while shielding operators from high radiation components.

12.3.1.2.5 Lighting

Adequate lighting is provided in radiation areas requiring access to facilitate surveillance and maintenance activities. Light fixtures are located in accessible areas to reduce replacement time. Multiple light fixtures are provided to reduce the need for immediate light bulb replacement. Emergency lighting fixtures reduce personnel exposures by permitting prompt egress from radiation areas if normal lighting fails.

12.3.1.2.6 Cubicles

Shielded cubicles are provided for components containing significant radioactive sources. Cubicles are lined with stainless steel to a height necessary to contain the contents of the residing component plus piping drainage. In the event of a leak or spill, cubicle floors slope toward floor drains that are connected to sump tanks.

12.3.1.3 Radiation Zoning and Access Control

12.3.1.3.1 Normal Conditions

The plant is analyzed for expected radiation levels resulting from normal operation. Because potential airborne exposures are possible in portions of the Reactor Building (RXB), principally due to off-gassing from the reactor pool and possible leaks or spills, airborne radiation zones are also developed. Radiation levels are categorized along with anticipated personnel occupancy in Table 12.3-1, which tabulates the radiation zone categories and their access descriptions. Table 12.3-2 tabulates the airborne zone categories and their access descriptions.

Figures 12.3-1a through 12.3-1h show the normal operation radiation zones for the RXB. Table 12.3-4a and Table 12.3-4b list the areas that have the potential for airborne radiation in the RXB and the Radioactive Waste Building (RWB), respectively. Figure 12.3-2a through Figure 12.3-2c show the normal operation radiation zones for the RWB. The radiation zones are based on conservative assumptions related to source terms and are not intended to reflect the anticipated dose rates over the entire area.

Access to radiologically controlled areas (RCAs) is controlled by the facility's radiation protection staff. Access control facilities are provided to control the entrance and exit of personnel and materials into and out of the RCA. Access is controlled through a portal located in the RWB. Radiological areas are posted with signage in compliance with 10 CFR 20.1901 and 20.1902.

High radiation areas either are locked or have alarmed barriers. For areas that are not within lockable enclosures or other barriers, the area is barricaded and posted, and be provided with a visible warning light. Positive control is exercised over each individual entry when access to the area is required, and egress from the area is not impeded.

COL Item 12.3-1: An applicant that references the NuScale Power Plant US460 standard design will develop the administrative controls regarding access to high radiation areas per the guidance of Regulatory Guide 8.38.

Very high-radiation areas are locked. Positive control is exercised over each individual entry when access to the area is required, and egress from the area is not impeded. Access to very high-radiation areas complies with guidance in RG 8.38. Based on the design and calculations, there are no very high-radiation areas.

COL Item 12.3-2: An applicant that references the NuScale Power Plant US460 standard design will develop the administrative controls regarding access to very high radiation areas per the guidance of Regulatory Guide 8.38.

COL Item 12.3-3: An applicant that references the NuScale Power Plant US460 standard design will specify personnel exposure monitoring hardware, specify contamination identification and removal hardware, and establish administrative controls and procedures to control access into and exiting the radiologically controlled area.

12.3.1.3.2 Accident Conditions

Post-accident access is discussed in Section 12.4.1 and equipment qualification is addressed in Section 12.2.1 and Section 3.11. A radiation and shielding design review is performed of spaces around systems that may contain core damage source term materials, consistent with 10 CFR 50.34(f)(2)(vii). The resultant equipment protection from a core damage source term is addressed in Section 19.2. Area radiation monitors are provided to indicate the post-accident radiation levels, to monitor plant areas during the progression of a postulated accident, and provide local indication to plant personnel before area entry.

Section 7.1 contains additional information on post-accident monitoring (PAM) instrumentation.

12.3.2 Shielding

12.3.2.1 Design Bases

The design function of shielding is to limit dose from plant radiation sources under normal operations and postulated accident conditions in accordance with General Design Criterion (GDC) 61, 10 CFR 50.34(f)(2)(vii), and 10 CFR 50.49.

Dose is limited to protect plant personnel, members of the public, and susceptible equipment subject to environmental qualification requirements.

Shielding performance is in accordance with the following criteria:

- ALARA radiation protection principles of 10 CFR 20
- exposure limits of 10 CFR 20
- dose limits of principal design criteria (PDC) 19

In addition, plant layout and shielding are used to limit equipment radiation doses to levels that are consistent with the assumptions used to demonstrate environmental qualification.

12.3.2.2 Design Considerations

Shielding is provided for radioactive systems and components to reduce radiation levels commensurate with area personnel access requirements and ALARA principles. Section 12.3.1 describes the radiation zones and indicates the radiation levels for those plant areas.

Section 12.3.1 describes shielding design features including permanent shielding and separation of components that constitute substantial radiation sources, the use of shielded cubicles, labyrinths, and shielded entrances to minimize radiation exposures. The selection of shielding materials considers the ambient environment and potential degradation mechanisms. Temporary shielding is considered where it is impractical to provide permanent shielding for substantial radiation sources.

Consistent with RG 8.8, streaming of radiation into accessible areas through penetrations for pipes, ducts, and other shield discontinuities is reduced by using layouts that prevent alignment with the radiation source, placing penetrations above head height to reduce personnel exposures, and using shadow shields to attenuate radiation streaming.

Consistent with RG 8.8, shielding analysis employs accurate modeling techniques and conservative approaches in the determination of shielding thickness. Source terms, geometries, and field intensities are analyzed conservatively. In addition to normal conditions, source terms include transient conditions such as resin transfers.

The material used for a significant portion of plant shielding is concrete. For most applications, concrete shielding is designed in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 6.4-2006 (Reference 12.3-1). Table 12.3-5 and Table 12.3-6 show the shielding thicknesses assumed in the shielding analyses in plant buildings. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. The use of lead is minimized.

A listing of radiation shield barrier equivalent doors is provided in Table 12.3-7 for the RWB. There are no credited shield doors for the RXB. They are modeled as open doorways.

Shield floor plugs provide an equivalent radiation attenuation as the shield floor that contains the plug.

12.3.2.3 Calculation Methods

The primary computer program to evaluate shielding is Monte Carlo N-Particle Transport Code (MCNP6) (Reference 12.3-2), which was developed by Los Alamos National Laboratory. The MCNP6 code is used for shielding calculations and for dose rate determinations.

Radioactive components in the RXB and RWB are modeled using MCNP6. Section 12.2 describes the codes used to prepare source strength input data. A three-dimensional shielding model is constructed for radioactive components using structure, location, and equipment data. Source geometries and source term distributions and intensities are conservatively determined. In general, the component source geometries are modeled as cylindrical volumes that incorporate the full volume of the component.

Shielding credit and material selections for MCNP6 cells are conservatively applied. Credit is not taken for reinforcing steel bars in the concrete. Table 12.3-5 and Table 12.3-6 describe the credited shield barriers for the RWB and RXB in terms of nominal concrete equivalent thicknesses. The design provides equivalent density thicknesses for the barrier described using a variety of structural design solutions.

The reactor shielding calculations consider dose rates from fission neutrons, fission photons, and gamma output from buildup of radioisotopes in the reactor coolant. The NuScale Power Module (NPM) model is conservatively developed using methods similar to the building evaluations.

The fission neutron and fission photon output is based on a total power output of 250 MWt and energy spectra are described in Section 12.2. The gamma output from the reactor coolant is based on the reactor coolant isotopic inventory described in Section 12.2. In order to reduce complexity, some region densities (e.g., water and piping in the SGs) are homogenized in the MCNP model. This simplification does not result in significant differences in dose rates.

The shielding thicknesses are selected to reduce the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone depicted in the zone maps (Figure 12.3-1a though Figure 12.3-1h). Radiation zones are selected to facilitate personnel access for operation and maintenance.

12.3.2.4 Major Component Shielding Design Description

12.3.2.4.1 NuScale Power Module

Biological shielding is provided above each NPM to allow personnel access. The bioshield provides shielding using concrete, steel, and high-density polyethelene (HDPE).The bioshield design and the venting of radiolytically-generated gases from the HDPE shielding are described in Section 3.7.3.

Degradation of the polyethylene radiation shielding material could potentially occur if the exhaust ventilation provided for the reactor module bays does not maintain air temperatures under the bioshield less than 180°F (e.g., due to damper failure). Therefore, conditions in which the air temperature under the bioshield exceeds 180°F require an evaluation of the continued efficacy of the bioshield polyethylene material's radiation shielding properties.

The CNV, pool water, and pool wall provide shielding and attenuation. The pool wall thickness is used for attenuating radiation from the radiation sources associated with the NPM.

Pool wall penetrations into the reactor module bay are modeled using MCNP6, with the shield voids occupied by the piping, HVAC ducting, cabling, and insulation, as designed for the penetrations. The model includes filler materials between the void spaces following design standard specifications for shielding penetrations. Calculations performed to determine the doses in areas near the module bay shielding penetrations show that the US460 design provides adequate protection from radiation streaming. These penetration compensatory measures provide protection of equipment, and reduce exposures to workers and the public.

12.3.2.4.2 Main Control Room

The dose rate in the main control room (MCR) during normal operations is negligible. The Control Building (CRB) room locations and elevations are shown in figures provided in Section 1.2. The CRB walls are designed to attenuate radiation from the RXB. As indicated by Table 15.0-10, the PDC 19 dose acceptance criteria for the control room are met for postulated accidents.

12.3.2.4.3 Reactor Building

In general, the calculated dose rates in open areas and corridors of the RXB are less than five mrem/hr during normal operation as shown in the radiation zone maps (Figure 12.3-1a through Figure 12.3-1h).

The RXB includes systems that contain radioactive components. The major radiation sources in the RXB are associated with the NPM (Section 12.3.2.4.1), chemical volume and control system, the pool cooling and cleanup system (PCWS), liquid radwaste degasifiers, and spent fuel storage. The shielding designs for these systems are described below.

Chemical and Volume Control System

The CVCS contains radioactive ion exchangers, filters, and heat exchangers. The CVCS components and piping are located below grade in the RXB as shown in the radiation zone maps.

The filters, ion exchangers, and heat exchangers are located in shielded cubicles. The knockout panels for accessing the CVCS ion exchangers provide equivalent shielding as the wall in which they are located. The CVCS filters and resin traps are accessible via removable floor shield plugs for maintenance purposes. The labyrinths in the cubicles provide shielding that significantly lowers the dose rates from areas adjacent to the radioactive component.

The CVCS is equipped with a resin transfer line used to transport resin slurry to the SRWS. The line is generically modeled in the RXB shielding model using the CVCS mixed-bed demineralizer transfer spectra, decayed for 48 hours. Resin transfers are planned evolutions to minimize operator exposure in accordance with ALARA principles.

Primary coolant piping in CVCS equipment rooms is shielded to minimize surveillance and maintenance dose rates. The reactor coolant system (RCS) discharge lines, which travel from the modules to the CVCS heat exchangers and purification equipment through a shielded pipe chase, are a radioactive source in the CVCS.

The CVCS design features that reduce radiation exposures are described in Section 9.3.4, Chemical and Volume Control System.

Pool Cooling and Cleanup Systems

During normal conditions of operation, the pool cooling and cleanup system (PCWS) provides for water level control and temperature maintenance of the reactor pool, the refueling pool (RFP) and the spent fuel pool (SFP). It also removes impurities to reduce radiation exposures and to maintain water chemistry and clarity.

For purposes of radiation shielding, the pool cooling heat exchangers are a minimal source of external radiation, and do not require shielding.

The design features of the pool cooling and cleanup systems that reduce radiation exposures are described in Section 9.1.3.

Degasifier Room

The LRWS degasifiers receive primary letdown and pressurizer vent flow from the CVCS. The degasifier contributes minor dose rates to the adjacent mechanical rooms and surrounding corridors and is acceptable for operations and maintenance activities. Mechanical skids for each degasifier are located in the adjacent mechanical room.

Fuel Storage and Handling Systems

The ultimate heat sink (UHS) is a safety-related pool of borated water that consists of the combined water volume of the reactor pool, RFP, and spent fuel pool. The UHS pool is located below grade in the RXB. The UHS provides shielding for the spent fuel assemblies in the SFP and the RFP as described in Section 9.1 and Section 9.2.5.

Shielding for radiation protection is maintained by the fuel handling equipment as described in Section 9.1.4.

Spent fuel is stored in the spent fuel pool, as described in Section 9.1. The radiation shielding provided by the pool water and the pool walls surrounding the fuel keeps radiation dose rates in the lower levels of the RXB within acceptance criteria.

12.3.2.4.4 Radioactive Waste Building

The RWB houses significant radiation sources that belong to the radioactive waste processing systems, and pool cooling and cleanup system components. The specifics of these systems are discussed below. The radiation zone maps are located in Figures 12.3-2a through 12.3-2b.

Liquid Radioactive Waste System

The LRWS is primarily located below grade in the RWB. The low-conductivity waste (LCW) and high-conductivity waste (HCW) sample tanks (two of each) contain LRWS water that is processed to comply with discharge or recycle requirements.

The LCW and HCW collection tanks are located in separate shielded compartments in the RWB. The respective transfer pumps for these tanks are located in shared compartments in the RWB. Each pair of transfer pumps is separated by sufficient space to allow room for temporary shielding, as well as space for tools, spare parts, and personnel.

Liquid radioactive waste demineralizers are located in a shared shielding labyrinth. Additional filtration systems are located on modular skids with integrated process shielding.

Additional shielding is modeled for the processing skids containing the LCW demineralizers as noted in Table 12.3-6.

Gaseous Radioactive Waste System

The gaseous radioactive waste system (GRWS) is located in the RWB. The GRWS components are generally located in separate, shielded compartments.

The redundant charcoal decay beds share a shielded compartment, separated by a shield wall. The decay beds are protected by a single guard bed that is separated by a shield wall.

The remaining GRWS components, consisting of gas heat exchangers (vapor condensers) and moisture separators, occupy a shared shielded compartment, separate from the decay beds and the guard bed, located in the RWB.

Solid Radioactive Waste System

The SRWS is located in the RWB. The SRWS components are generally located in separate, shielded compartments.

The two phase separator tanks and the two spent resin tanks are located in individual shielded compartments. The respective transfer pumps for these tanks are located in shared compartments in the RWB. Each pair of transfer pumps is separated by sufficient space to allow room for temporary shielding, as well as space for tools, spare parts, and personnel.

The SRWS consists of both Class A and Class B/C waste storage areas. A Class A waste package storage area is located at grade level. The Class A high integrity container (HIC) and Class B/C HIC storage area is located on the lower level. Access to the HIC storage area is through floor shield plugs.

Additional shielding is modeled for the HIC process shield, and an external shield is modeled outside a penetration through the HIC storage room shield wall, as noted in Table 12.3-6.

Pool Cooling and Cleanup

The PCWS demineralizers and filters are located in shielded cubicles in the RWB. The dose rates in surrounding areas are acceptable for operations or maintenance activities.

12.3.3 Ventilation

The plant HVAC systems are designed to provide a controlled environment for personnel and equipment during normal operation. In areas subject to airborne radioactivity, the ventilation systems are designed to collect, process, and direct airflow to processed exhausts (Section 9.4, HVAC Systems). This section discusses the radiation control considerations of the HVAC systems design.

12.3.3.1 Design Objectives

Design objectives for the plant HVAC systems include the following:

- During normal plant operations, the airborne radioactivity levels to which plant personnel are exposed in radiation controlled areas are maintained ALARA and within the limits specified in 10 CFR 20. The airborne radioactivity released during normal plant operations are also maintained ALARA and within the limits of 10 CFR 20, Appendix B, Table II.

- During normal plant operations, the dose from airborne radioactive material exposure in unrestricted areas is maintained ALARA and within the limits specified in 10 CFR 20.1301 and 10 CFR 50, Appendix I.
- The dose to the control room personnel does not exceed the limits specified in PDC 19 following the design basis accidents described in Chapter 15.

12.3.3.2 Design Features to Minimize Personnel Exposure from Heating, Ventilation, and Air Conditioning Equipment

The design of the building ventilation systems maintain a negative pressure where required with respect to the outside environs and create air flow inside the building from areas of low airborne potential contamination to areas of higher airborne potential contamination.

Other design features that minimize radiation exposures to personnel are listed below.

- The design of the plant ventilation systems incorporates the guidance of RG 8.8.
- Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The HVAC system is designed to allow rapid replacement of components. Filter-adsorber unit performance complies with the recommendations of RG 1.140.
- Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.
- Access to ventilation systems in potentially radioactive areas can result in personnel exposure during maintenance, inspection, and testing. Equipment is located in low-dose areas as much as practicable, with most equipment being located outside of rooms that contain significant radiation sources. The outside air supply units and building exhaust system components have adequate work space provided around each unit for anticipated maintenance, testing, and inspection.

12.3.3.3 Reactor Building Heating, Ventilation, and Air Conditioning System

During normal operation, the RBVS services the areas inside the RXB by providing conditioned and filtered outside air. The exhaust from the spent fuel pool area is filtered by a high-efficiency particulate air (HEPA) filter. If the spent fuel pool exhaust radiation monitors detect radioactivity above their setpoints, the exhaust flow from the spent fuel pool area is diverted to go through an additional HEPA filter and charcoal adsorbers. Section 9.4.2, Reactor Building HVAC, contains additional details.

The dry dock area has dedicated exhaust vents to entrain airborne contamination that may result from air being exposed to NPM components during maintenance activities.

Heating, ventilation, and air conditioning equipment drains are routed to the RWDS.

The design provides adequate space for temporary shielding to minimize personnel exposures during maintenance of ventilation equipment, including filters, inspection, and testing. In addition, the filter units have design features that minimize the time required for filter changes.

12.3.3.4 Radioactive Waste Building Heating, Ventilation, and Air Conditioning System

The RWBVS serves the RWB as a once-through system. Outside air is introduced by the main supply air handling unit (AHU) and is exhausted through the RBVS exhaust system. The main supply AHU contains both low and medium efficiency outside air filters. Supply air from the main RWBVS is distributed throughout the RWB. Exhaust air is collected and conveyed to the RBVS general area exhaust subsystem and exhausted through the main stack. The RWBVS maintains airflow from areas of lesser potential contamination to areas of greater potential contamination. The RWBVS also maintains the RWB atmosphere at a slight negative pressure with respect to the outside. Section 9.4.3, Radioactive Waste Building HVAC, contains additional details.

12.3.3.5 Normal Control Room Heating, Ventilation, and Air Conditioning System

During normal operations, the normal control room HVAC system (CRVS) supplies conditioned air to the CRB, including the control room envelope (CRE), the technical support center, and the other areas of the CRB with outside air that is filtered (low and medium efficiency) to maintain a suitable environment for personnel and equipment. The CRVS is designed to maintain a positive pressure inside the CRB with respect to adjacent spaces. Section 9.4.1, Control Room Area Ventilation System, contains additional details.

If a high radiation indication is received from an outside air intake radiation monitor, the supply air is routed through the CRVS filter unit, which provides HEPA and charcoal filtration. The CRVS is designed to maintain operator doses in the MCR and technical support center within PDC 19 limits.

If power is not available, or if a high radiation indication is received from the radiation monitors in the CRE supply duct, the CRE isolation dampers close and the control room habitability system (CRHS) is initiated.

12.3.3.6 Control Room Habitability System

The CRHS uses a set of compressed air storage bottles to supply the CRE with breathing air in the event of an emergency. Upon receiving an initiation signal, the CRHS supplies the CRE with clean air and maintains the CRE at a positive pressure with respect to adjacent areas. Section 6.4, Control Room Habitability, contains additional details.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The fixed area radiation monitors (ARMs) are placed in selected general plant locations. They provide local and MCR indication of gamma radiation at each location and provide an alarm function both locally and in the MCR when predetermined thresholds are exceeded.

The ARMs located under the bioshield for each NPM provide normal and post-accident indication of containment gamma radiation within the bioshield envelope during normal and accident conditions. The monitors detect fuel cladding breach under potential core damaging accident conditions. Section 9.3.2, Process Sampling System, discusses the use of ARMs.

The ARMs located adjacent to the CVCS reactor coolant filters provide indication of radioactive material buildup to allow the operator to remove them from service for maintenance before reaching predetermined thresholds for radiation. Area radiation monitoring design features, quality group, seismic requirements, and functions are described below. The CVCS design features, functions, and flow paths are described in Section 9.3.4.

The ARMs located in the spent fuel pool area provide the same functions as the general plant location monitors, and in addition monitor the fuel storage and handling areas. Additionally, a local ARM is mounted on the refueling bridge with local and MCR alarm function that monitors refueling activities.

The fixed continuous air monitors (CAMs) are placed in selected general plant locations. They provide local and MCR indication of airborne radioactivity within the plant environs at each location and provide an alarm function both locally and in the MCR when predetermined thresholds are exceeded.

The CAM for the GRWS provides an additional function: detection of GRWS process gas leakage. The GRWS continuous airborne radiation monitor function is described in Section 11.3, Gaseous Waste Management System.

12.3.4.1 Design Bases

The area and airborne radiological monitoring equipment is designed to meet the following design basis requirements:

- provide monitoring of area and airborne radiation levels in fuel storage and handling areas, and radioactive waste management systems to detect excessive radiation levels (GDC 63)
- provide monitoring of plant area and airborne radiation levels such that worker dose can be maintained ALARA conforming to 10 CFR 20.1101(b)
- monitor plant areas during normal operations and anticipated operational occurrences such that the worker limits do not exceed the limits specified in 10 CFR 20.1201 and 10 CFR 20.1202
- provide monitoring of airborne radiation levels such that area dose rate can be monitored, in part conforming to 10 CFR 20.1203

- provide monitoring of airborne radioactive materials in work areas conforming to 10 CFR 20.1204
- provide monitoring of plant area and airborne radiation levels such that contaminated system leaks can be detected and addressed in a timely manner, in part conforming to 10 CFR 20.1406
- provide monitoring of plant area and airborne radiation levels such that effective surveys of these parameters can be maintained, conforming to 10 CFR 20.1501
- provide monitoring of plant area and airborne radiation levels for use in the emergency response data system (ERDS), conforming to 10 CFR 50, Appendix E, VI.2(a)
- provide monitoring of containment radiation levels, conforming to 10 CFR 50.34(f)(2)(xvii)
- provide monitoring of plant area and airborne radiation levels for a broad range of routine and accident conditions, conforming to 10 CFR 50.34(f)(2)(xxvii)
- provide radiation monitoring in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions, conforming to 10 CFR 50.68(b)(6).

12.3.4.2 Fixed Area Radiation Monitoring Instrumentation

The fixed ARMs and associated instrument and controls platforms provide indication and archiving function to the MCR, furnishing information that can supplement radiological surveys, meet reporting requirements, and inform workers of radiological conditions before accessing monitored areas, thus providing the capability for plant staff to meet the requirements of 10 CFR 20.1501.

The ARMs provide both indication and alarm functions to the local plant area, the MCR, and, for selected areas, the waste management control room. The indication ensures operator and worker awareness of changing radiological conditions that could indicate system leakage or component malfunction, and provides a warning to plant personnel before entry into the affected areas. Where appropriate, local visual alarms are provided outside of the monitored area to ensure worker awareness before entry into the affected area.

The above design features conform to the requirements of 10 CFR 20.1101(b), 10 CFR 20.1201, 10 CFR 20.1406, and 10 CFR 50.34(f)(2)(xxvii).

For the ARMs in general plant locations, alarm setpoints are established to alert plant personnel when radioactivity in a specific location reaches levels that have been determined to be abnormal. The alarm setpoints are adjusted to values that are low enough for the minimum detectable activity anticipated and high enough not to give false alarms. Alarms are designed such that they do not reset without operator action. The radiation monitor remains operable when the alarm setpoint is exceeded.

Meters, alarm indicators, and audible devices are designed so plant personnel can quickly determine the status of each radiation channel. These functions ensure personnel working in the vicinity are able to determine easily the status of an ARM channel when in the vicinity of the local indication devices.

Fixed ARM placement conforms to the criteria for selection and placement of the area radiation monitoring instrumentation contained within ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors" (Reference 12.3-4). Area radiation detectors are located in those areas that are normally accessible and require entry, exit, or both to monitor for purpose of occupational radiation protection. To the extent practicable, detectors are located to best measure the representative exposure rates within a given area or specified location.

The following criteria are considered for detector placement.

- Areas that are normally accessible and where changes in normal plant operating conditions can cause significant increases in exposure rates above those expected for the areas.
- Areas that are normally or occasionally accessible and where significant increases in exposure rates might occur because of operational transients or maintenance activities.
- Areas where shielding of the detector by equipment or structural materials are avoided to ensure correct monitor response to increases in exposure rates within a specific area
- Environmental conditions under which the monitor operates consider the range of temperature, pressure and humidity of areas where the detector and electronics are located.
- The electronic controls for the monitors are placed in the lowest dose area practicable to provide maintenance access in an unobstructed area.

The fixed ARM indicating ranges consider the design maximum dose rate of the radiation zone in which they are located and the maximum dose rate for anticipated operational occurrence and accident conditions. Multiple-range devices are used for applications where a single monitoring range is not sufficient to envelope the entire anticipated indication requirement. The range of the radiation monitor is chosen so that the upper end of the scale is high enough to ensure on-scale reading for exposure rates far greater (approximately two decades) than the expected peak exposure rate, and the low end is at the lower end of the expected exposure rate range but provides an on-scale value for the range of the instrumentation selected.

The ARMs employ a simple and robust design to minimize routine maintenance. The radiation monitors allow for periodic and corrective maintenance during normal operation, accident conditions, and outages. The fixed ARM calibration methods are performed on an annual basis or in alignment with the manufacturer's recommendations. Each area radiation monitor may have a check source that is used to provide operability checks.

Consumable filter media and cartridges for the collection of airborne radioactive particulates and iodine are replaced on a periodic basis. The periodicity of filter and cartridge replacements is dependent upon vendor recommendations and plant operating experience. Replacement of consumable filter media and cartridges shall be part of routine maintenance with proper procedures in place.

Selected ARMs support accident condition response and are PAM system variables, as described in Table 12.3-8 and Table 7.1-7. The fixed ARMs used for PAM have ranges that consider the maximum calculated accident levels and are designed to operate effectively under the environmental conditions caused by an accident. These monitors conform to the guidance of RG 1.97.

The ARMs located under the bioshield for each NPM detect fuel damage under accident conditions, and are considered PAM system B, C, and F variables. Monitors are located at the top of each NPM beneath the bioshield. The radiation monitors under the bioshield are environmentally qualified to survive an accident and perform their design functions. The instruments are designed to respond to gamma radiation over the energy range of at least 60 keV to 3 MeV, with a dose rate response accuracy within a factor of two over the entire range. These monitors also meet the applicable requirements of Institute of Electrical and Electronics Engineers Standard 497-2016 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" (Reference 12.3-7). By the use of independent vendors using software that is independent, the design configuration meets the requirements of Institute of Electrical and Electronics Engineers Standard 497-2016, Section 6.2, Common Cause Failure.

Fixed area radiation monitoring data are capable of being supplied to the NRC Operations Center through the connection to emergency response system via a secure direct electronic data link in the event of an emergency. The Connection to emergency response systems is discussed in Section 7.2.13.

Electrical power to the ARMs is provided by the following systems.

- Fixed ARMs that are classified as a Type B PAM variable receive power from the augmented DC power system (EDAS).
- Fixed ARMs that are classified as a Type C PAM variable receive power from the (EDAS) DC power system.
- Fixed ARMs that are classified as a Type E PAM variable receive power from the normal DC power system (EDNS).
- Fixed ARMs that are not used for PAM variables receive power from the EDNS.

The methodology for inspections, tests, analyses, and acceptance criteria associated with the fixed area radiation monitors are described in Section 14.3.

Table 12.3-10 provides information about the ARMs used including location and design features such as the type of radiation monitored and the associated principle isotope(s), instrument ranges, and the identification of monitors that serve a PAM function.

COL Item 12.3-4: An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs necessary for the implementation of 10 CFR 20.1501 related to conducting radiological surveys, maintaining proper records, calibration of equipment, and personnel dosimetry.

12.3.4.3 Airborne Radioactivity Monitoring Instrumentation

The fixed continuous airborne radiation monitors and their associated instrument and controls platforms provide indication and archiving function to the MCR, furnishing information that can supplement radiological surveys, meet reporting requirements, and inform workers of radiological conditions before accessing monitored areas, thus providing the capability for plant staff to meet the requirements of 10 CFR 20.1501.

The CAMs provide both indication and alarm functions to the local plant area, the MCR, and, for selected areas, the waste management control room. These functions ensures operator and worker awareness of changing radiological conditions that could indicate system leakage or component malfunction, and provides a warning to plant personnel before entry into the affected areas. Where appropriate, local visual alarms are provided outside of the monitored area to ensure worker awareness before entry into the affected area.

The above design features conform to the requirements of 10 CFR 20.1101(b), 10 CFR 20.1201, 10 CFR 20.1406, and 10 CFR 50.34(f)(2)(xxvii).

Selected fixed CAMs that support accident condition response and are PAM system variables, as described in Table 12.3-8 and Table 7.1-7. The fixed CAMs for PAM system have ranges that consider the maximum calculated accident levels and are designed to operate effectively under the environmental conditions caused by an accident. The CAMs used to fulfill PAM system functions conform to the guidance of RG 1.97.

Fixed CAM data are capable of being supplied to the NRC Operations Center through the connection to emergency response system via a secure direct electronic data link in the event of an emergency. Section 7.2 discusses the connection to emergency response system.

Alarm setpoints are established to alert plant personnel when airborne radioactivity in a specific location reaches levels that have been determined to be abnormal. The alarm setpoints are adjusted to values that are low enough for the minimum detectable activity anticipated and high enough not to give false alarms. Alarms are designed such that they do not reset without operator action. The radiation monitor remains operable when the alarm setpoint is exceeded.

Meters, alarm indicators, and audible devices are designed so plant personnel can quickly determine the status of each radiation channel. These functions ensure personnel working in the vicinity are able to determine easily the status of fixed CAMs when in the vicinity of the local indication devices.

Fixed CAM placement and selection conforms to the criteria contained within ANSI/HPS N13.1-2011, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities" (Reference 12.3-8) and RG 8.25. The following criteria are considered for monitor placement:

- Areas that are normally accessible and where changes in normal plant operating conditions can cause significant increases in airborne radioactivity above those expected for the areas.
- Areas that are normally or occasionally accessible and where significant increases in airborne radioactivity might occur because of operational transients or maintenance activities.
- Optimal location to measure the increase in airborne radioactivity within a specific area, including leak detection functions for systems that convey process gases that contain radionuclides and hydrogen.
- Provide maintenance access in an unobstructed area. The electronic controls for the monitors are placed in the lowest dose area practicable.
- Proximity of the airflow path that is as close as practical to potential release points.

The range of the fixed CAMs is chosen so that the upper end of the scale is high enough to ensure on-scale reading for exposure rates far greater (approximately two decades) than the expected peak exposure rate, and the low end is at the lower end of the expected exposure rate range but provide an on-scale value for the range of the instrumentation selected.

The fixed CAM calibration methods and frequency are in accordance with manufacturer recommendations and consider the rate at which instrument components age or become damaged. The calibrations are performed in a manner consistent with ALARA principles and follow the guidance of EPRI report TR-102644 Revision 1, "Calibration of Radiation Monitors at Nuclear Power Plants" (Reference 12.3-5). Recalibrations are performed on the detectors after maintenance or replacement of components that affect calibration. Radiation detectors used to satisfy PAM requirements are provided a means of calibration and testing the operability of each instrument channel during plant operation. Functional testing of the fixed CAMs is performed to verify the operability of the channel, including alarm functions in accordance with manufacturer's requirements and using the guidance of EPRI report TR-104862, Revision 2, "Area and Process Radiation Monitoring System Guide" (Reference 12.3-6). Check sources integral to the monitor are designed to ensure that the source is returned to the non-test mode upon deactivation or loss of power to the monitor.

The following systems provide electrical power to the fixed CAMs.

- Fixed continuous airborne radiation monitors that are classified as a Type E PAM variable receive power from the EDNS.
- Fixed continuous airborne radiation monitors that are not used for PAM variables receive power from the EDNS.

The methodology for inspections, tests, analyses, and acceptance criteria associated with the fixed area radiation monitors are described in Section 14.3.

Table 12.3-9 provides information about the fixed airborne monitors used including location and design features such as the type of radiation monitored and the associated principle isotope(s), instrument ranges, and the identification of which monitors serve a PAM function.

12.3.4.4 Portable Airborne Monitoring Instrumentation

COL Item 12.3-5: An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs necessary for the use of portable airborne monitoring instrumentation, including accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

12.3.5 Dose Assessment

Section 12.4, Dose Assessment.

12.3.6 Minimization of Contamination and Radioactive Waste Generation

Design features incorporated into the plant, combined with operational programs, are provided to comply with 10 CFR 20.1406, and the guidance of RG 4.21, to minimize contamination of the facility and the environment, minimize the generation of radioactive waste, and facilitate decommissioning. The design of the facility is evaluated in a systematic and risk-informed fashion against the design objectives below to determine the design features necessary for plant structures, systems, and components (SSC).

The SSC that have potential to contain contaminated fluids have design features to reduce the likelihood of leaks, provisions to detect leaks that do occur and reduce the spread of contamination, thus reducing the need for decontamination and the generation of waste. Table 12.3-11 provides a list of structures and systems that have been evaluated for design features consistent with RG 4.21. Table 12.3-11 through Table 12.3-40 provide the results of the system evaluations.

12.3.6.1 Facility Design Objectives for 10 CFR 20.1406

Regulatory Guide 4.21 addresses each phase of the NuScale Power Plant lifecycle, from the early design phases through decommissioning. The discussion of the lifecycle phases is separated into four design phase objectives and two operational objectives. Application of these six objectives demonstrates compliance with 10 CFR 20.1406 requirements.

The design features in the NuScale facility include measures to minimize and detect leakage, reduce radioactive waste generation, and facilitate decommissioning. These 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.

In combination with the four design related objectives listed above, two operational objectives are included to fully demonstrate compliance:

- Objective 5 - Operational and programmatic considerations
- Objective 6 - Site Radiological Environmental Monitoring

12.3.6.1.1 Design Considerations to Minimize Leaks and Contamination - Objective 1

Facility contamination can be spread by leaks and spills of fluids containing radioactive material. To reduce the potential for these leaks and spills, the following design features include:

- proper selection of materials that are commensurate with the SSC service conditions to reduce the effects of corrosion, temperature, pressure, and more.
- providing walls, dikes, drains, sloped floors, and other leak collection features to contain leaks and spills from SSC containing contaminated, or potentially contaminated, liquids
- minimize the use of buried or embedded piping and drains without the concurrent use of double-walled pipe with leak detection capability
- proper design of components regarding properly sized overflow lines and catch basins or drip pans that are routed to drains
- use proven technologies with the proper quality controls and compliance to applicable codes and standards

12.3.6.1.2 Design Considerations for Leak Detection - Objective 2

Prompt leak detection from SSC provides the opportunity for an appropriate response to prevent unintended spread of radioactive contamination. The following design features are incorporated, where appropriate, in the facility design:

- leak detection instrumentation for specific plant SSC
- floors designed with drains and leak detection equipment

- drainage collection provisions with leak detection equipment
- liquid- and moisture-detection instruments with appropriate sensitivity
- provisions for periodic calibration and maintenance of leak detection instrumentation
- sufficient space for access to assess detected leaks and allow operator response
- area and airborne radiation monitors
- trenches or guard pipes with leak detection capabilities

12.3.6.1.3 Design Considerations for Reduction of Cross-Contamination, Decontamination and Waste Generation - Objective 3

Design features are incorporated to reduce the potential for cross-contamination, the need for decontamination, and radioactive waste generation. These kinds of design considerations include

- separation of components according to their contamination level and characteristics.
- the ability to sufficiently contain, isolate, and hold contamination until operator responses can be initiated.
- smooth and cleanable surfaces on SSC to ease decontamination.
- flushing capabilities for appropriate systems to be able to clean in place.
- an on-site decontamination facility.
- design for ventilation flow to be from lower contaminated areas to higher contaminated areas.
- double isolation valves between clean and contaminated systems.
- the use of butt welds, full ported valves and diaphragm seals, where appropriate, to minimize crud traps.

12.3.6.1.4 Design Considerations for Decommissioning - Objective 4

The following facility design features are included, as appropriate:

- use of modular construction
- minimizing the use of buried or embedded piping and components
- use of removable walls to ease component removal
- component designs to include, as appropriate, lifting lugs, easily removable insulation, and sufficient means for removal

12.3.6.1.5 Operational and Programmatic Considerations - Objective 5

The following procedural measures are employed:

- periodic review of site procedures and programs to ensure adherence by, and training of, plant personnel and to verify proper updates to reflect plant modifications (Reviews are also conducted after leak or spill events to verify the adequacy of the associated programs and procedures).
- site procedures and programs include measures to control contamination from potential leaks and spills, including monitoring, surveillances, and preventative maintenance.
- proper documentation of the facility design, construction, modifications, and operations, including site contamination events.

12.3.6.1.6 Site Radiological Environmental Monitoring - Objective 6

A conceptual site model is developed that

- characterizes the site's geology and hydrology and evaluates the predominant ground water flow characteristics and gradients.
- identifies potential pathways for ground water migration to offsite locations.
- evaluates the impact of construction upon the site's hydrogeological characteristics.
- forms part of the basis for a site radiological monitoring program for ground water migration of potential releases.

COL Item 12.3-6: An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary to demonstrate compliance with 10 CFR 20.1406, and the guidance of Regulatory Guide 4.21.

12.3.7 References

12.3-1 American National Standards Institute/American Nuclear Society, "Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," ANSI/ANS 6.4-2006, La Grange Park, IL.

12.3-2 Monte Carlo N-Particle Transport Code System Including MCNP6. 1, MCNP5-1.60, MCNPX-2.7.0 and Data Libraries [Computer Program]. Oak Ridge National Laboratory Radiation Safety Information Computational Center (RSICC) Computer Code Collection (CCC-740), Oak Ridge, TN.

12.3-3 American National Standards Institute/American Nuclear Society, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," ANSI/ANS 6.1.1-1977, La Grange Park, IL.

12.3-4 American National Standards Institute/American Nuclear Society, "Location and Design Criteria for Area Radiation Monitoring Systems for

Light Water Nuclear Reactors," ANSI/ANS/HPSSC-6.8.1-1981, LaGrange Park, Il.

12.3-5 Electric Power Research Institute, "Calibration of Radiation Monitors at Nuclear Power Plants," EPRI #102644, EPRI, Palo Alto, CA, 2005.

12.3-6 Electric Power Research Institute, "Area and Process Radiation Monitoring System Guide," EPRI #104862, EPRI, Palo Alto, CA, 2003.

12.3-7 Institute of Electrical and Electronics Engineers, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," IEEE Standard 497-2016, New York, NY.

12.3-8 American National Standards Institute/Health Physics Society, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," ANSI/HPS N13.1-2011, Washington, DC.

12.3-9 Oak Ridge National Laboratory, "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6, Vols. I-III, ORNL/TM-2005/39, Version 6.1, June 2011.

Table 12.3-1: Normal Operation Radiation Zone Designations

Zone	Designation	Restriction	Radiation Limit		Description	Regulatory Requirement
Unrestricted Zones						
0	Unrestricted area	None	≤ 0.05 mrem/hr	≤ 0.5 µSv/hr	Areas of the plant that can be occupationally occupied without exceeding annual dose limits for members of the public.	10 CFR 20.1301(a)(2)
Controlled Zones						
I	Controlled area	Limited occupancy	≥ 0.05 mrem/hr ≤ 0.25 mrem/hr	≥ 0.5 µSv/hr ≤ 2.5 µSv/hr	Areas of the plant that can be occupationally occupied without exceeding personnel radiation monitoring requirements (10% of 10 CFR 20.1201(a) limit).	10 CFR 20.1502(a)(1)
Restricted Zones						
II	Controlled area	Personnel radiation monitoring	≥ 0.25 mrem/hr ≤ 2.5 mrem/hr	≥ 2.5 µSv/hr ≤ 0.025 mSv/hr	Areas of the plant that can be occupationally occupied without exceeding the annual occupational dose limit of 5 rem (0.05 Sv).	10 CFR 20.1201(a)
III			≥ 2.5 mrem/hr ≤ 5 mrem/hr	≥ 0.025 mSv/hr ≤ 0.05 mSv/hr	Areas of the plant that require limited access to ensure compliance with the annual occupational dose limit of 5 rem (0.05 Sv).	10 CFR 20.1201(a)
IV	Radiation area	Posting required	≥ 5 mrem/hr ≤ 100 mrem/hr	≥ 0.05 mSv/hr ≤ 1 mSv/hr	Areas of the plant that require posting as radiation areas.	10 CFR 20.1902(a)
V	High Radiation area[1]	Access restriction	≥ 100 mrem/hr ≤ 1 Rad/hr	≥ 1 mSv/hr ≤ 10 mGy/hr	Areas of the plant that require controlled access and posting as high radiation areas.	10 CFR 20.1601 10 CFR 20.1902(b)
VI	Locked high-radiation area[1]		≥ 1 Rad/hr ≤ 500 Rad/hr	≥ 10 mGy/hr ≤ 5 Gy/hr		
VII	Very high-radiation area		≥ 500 Rad/hr	≥ 5 Gy/hr	Areas of the plant that require controlled access and posting as very high radiation areas.	10 CFR 20.1602 10 CFR 20.1902(c)

[1] The high-radiation area designation is split for administrative purposes into two zones. This is to provide the opportunity for additional controls to be used for zones with radiation dose rates above 1 Rad/hr.

Table 12.3-2: Airborne Radiation Zone Designations

Zone	Designation	Restriction	Radiation Limit	Description	Regulatory Requirement
Unrestricted Zones					
0	Unrestricted area	None	≤ 0.01 DAC[1]	Areas that can be maintained as uncontrolled because radionuclide concentrations can be inhaled continuously without exceeding public dose limit.	10 CFR 20 App. B, Table 2 notes
Controlled Zones					
I	Controlled area	Limited occupancy	≥ 0.01 DAC ≤ 0.1 DAC	Areas of the plant that can be continuously occupied without personnel radiation monitoring required (10% of applicable ALIs[2]).	10 CFR 20.1502(b)(1)
Restricted Zones					
II	Controlled area	Personnel radiation monitoring	≥ 0.1 DAC ≤ 0.3 DAC	Areas of the plant that can be occupationally occupied without exceeding the airborne radioactivity area limit requiring respiratory protective equipment.	10 CFR 20.1003 10 CFR 20.1502(b)(1)
III			≥ 0.3 DAC ≤ 1 DAC	Areas of the plant that require respiratory protective equipment.	
IV	Airborne radioactivity area	Posting required	≥ 1 DAC	Areas of the plant that require posting as airborne radioactivity areas.	10 CFR 20.1003 10 CFR 20.1902(d)

Notes:

1. DAC = derived air concentration
2. ALI = annual limit on intake

Table 12.3-3: Typical Cobalt Content of Materials

Material or Application	Maximum Weight Percent of Cobalt
Austenitic stainless steel weld filler metals (including cladding)	0.05
Austenitic stainless steel base materials for large forgings >10,000 lbm	0.05
Reactor vessel internal core reflector blocks	0.05
Ni-Cr-Fe base metal and weld filler metals (except Alloy 690 SG tubing below)	0.05
Alloy 690 SG tubing	0.014 max average per NuScale power module (NPM), with no single heat exceeding 0.020
SB-637, N07718 (Aloy 718) bolting	0.05
Other small components in contact with primary coolant	0.2 target
CRDM internals springs in contact with primary coolant (Inconel X-750)	1

Table 12.3-4a: Reactor Building Areas of Potential Airborne Radioactive Material

Location & Room # (Note 1)	Description	Source of Airborne Radioactive Material
Elevation 25’-0 (035 & 036)	LRWS degasifier pump rooms	Primary coolant, CVCS letdown
Elevation 25’-0 (006, -009, -012, -024, -027, & 030)	CVCS demineralizer valve gallery/labyrinth	Primary coolant
Elevation 40’-0 (108, through -116,& 122 through -130)	CVCS recirculation pump rooms and valve galleries	Primary coolant
Elevation 40’-0 (034 & 037)	LRWS degasifier tank cubical	Primary coolant, CVCS letdown
Elevation 55’-0 (206 & 215)	CVCS heat exchanger utility areas	Leaked airborne from CVCS valve galleries
Elevation 55’-0 (213)	Module heatup system (MHS) equipment room	Primary coolant
Elevation 70’-0 (330)	PSS primary sampling panels room	Primary coolant
Elevation 70’-0 (304)	Pool cooling and cleanup system equipment aisleway	Pool water
Elevation 70’-0 (317)	CFDS pump utilities area	Pool water
Elevation 85’-0 (432)	Hot lab	Primary coolant samples
Elevation 100’-0 (507 & 512)	MS galleries	Vented RCS leaks into CNV
Elevation 100’-0 (042)	Reactor pool area	Pool evaporation

Note 1: Figure 1.2-8 through Figure 1.2-18 depict room locations.

Table 12.3-4b: Radioactive Waste Building Areas of Potential Airborne Radioactive Material

Location & Room # (Note 1)	Description	Source of Airborne Radioactive Material
Elevation 70'-0 (029 & 030)	PCWS filter rooms	Pool water
Elevation 70'-0 (031 & 032)	PCWS demineralizers and adjacent valve galleries	Pool water, PCWS demineralizer resin
Elevation 70'-0 (010, -015, -016, -022, -023, & 026)	SRWS tank and transfer pump rooms	Spent resin and decant liquid
Elevation 70'-0 (008, -009, -011 through -014, -018 through -021, -024, & 025)	LRWS tank and transfer pump rooms	LRWS
Elevation 70'-0 (037 & 038)	GRWS decay bed and condenser rooms	Gaseous radioactive waste
Elevation 70'-0 (035)	SRWS HIC filling room	Spent resin and decant liquid
Elevation 70'-0 (034)	SRWS HIC storage room	Spent resin and decant liquid
Elevation 70'-0 (033)	SRWS drum storage room	Packaged solid radioactive waste
Elevation 70'-0 (005)	LRWS processing equipment room	LRWS
Elevation 70'-0 (005)	LRWS drum dryer area	LRWS
Elevation 100'-0 (225)	Class A and low level solid radwaste storage rooms	Packaged solid radioactive waste
Elevation 145'-0 (402)	RWBVS equipment area	RWBVS exhaust

Note 1: Figure 1.2-24 to Figure 1.2-26 depict room locations.

Table 12.3-5: Reactor Building Shield Wall Geometry

Elevation (Note 1)	Room No. (Note 1)	Room Name (Note 1)	Radioactive Source	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling
25'-0"	006	Module 01 CVC ion exchanger valve room	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	27.5" concrete
25'-0"	007	Module 01 CVC ion exchanger enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	008	Module 01 CVC filter enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	48" concrete	Basemat	24" concrete
25'-0"	009	Module 02 CVC ion exchanger valve room	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	27.5" concrete
25'-0"	010	Module 02 CVC ion exchanger enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	011	Module 02 CVC filter enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	012	Module 03 CVC ion exchanger valve room	CVC IXs, CVC FLTs	20" concrete	48" concrete	20" concrete	20" concrete	Basemat	27.5" concrete
25'-0"	013	Module 03 CVC ion exchanger enclosure	CVC IXs, CVC FLTs	20" concrete	48" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	014	Module 03 CVC filter enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	024	Module 04 CVC ion exchanger valve room	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	27.5" concrete
25'-0"	025	Module 04 CVC ion exchanger enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	026	Module 04 CVC filter enclosure	CVC IXs, CVC FLTs	20" concrete	48" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	027	Module 05 CVC ion exchanger valve room	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	27.5" concrete
25'-0"	028	Module 05 CVC ion exchanger enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	029	Module 05 CVC filter enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	030	Module 06 CVC ion exchanger valve room	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	48" concrete	Basemat	27.5" concrete
25'-0"	031	Module 06 CVC ion exchanger enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	48" concrete	Basemat	24" concrete

Table 12.3-5: Reactor Building Shield Wall Geometry (Continued)

Elevation (Note 1)	Room No. (Note 1)	Room Name (Note 1)	Radioactive Source	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling
25'-0"	032	Module 06 CVC filter enclosure	CVC IXs, CVC FLTs	20" concrete	20" concrete	20" concrete	20" concrete	Basemat	24" concrete
25'-0"	034	LRW degasifier tank B room	LRW DGS B	48" concrete	32" concrete	32" concrete	32" concrete	Basemat	32" concrete
25'-0"	037	LRW degasifier tank A room	LRW DGS A	48" concrete	32" concrete	32" concrete	48" concrete	Basemat	32" concrete
26'-0"	040	Dry dock	Pool water	48" concrete	-	48" concrete	48" concrete	Basemat	-
26'-0" to 100'-0"	041	Spent fuel pool	Spent fuel assemblies, Pool water	48" concrete	-	48" concrete	48" concrete	Basemat	36" concrete
26'-0" to 100'-0"	042	Reactor pool	Pool water	48" concrete	48" concrete	48" concrete	48" concrete	Basemat	30" concrete
26'-0" to 126'-0"	-	Module bays - Modules 01-03	RXMs, Pool water	51.75" concrete (TYP), 72" concrete (below EL 43'-0")	51.75" concrete (TYP), 72" concrete (East pool wall below EL 43'-0")	3.5" HDPE panels, 5% boron content (vertical bioshield)	51.75" concrete	Basemat	29" concrete
26'-0" to 126'-0"	-	Module bays - Modules 04-06	RXMs, Pool water	3.5" HDPE panels, 5% boron content (vertical bioshield)	51.75" concrete (TYP), 72" concrete (East pool wall below EL 43'-0")	51.75" concrete (TYP), 72" concrete (below EL 43'-0")	51.75" concrete	Basemat	29" concrete
25'-0"	-	Pipe chase - SRW resin transfer line	SRW resin transfer line	24" concrete	24" concrete	24" concrete	48" concrete	Basemat	-
40'-0"	109	Module 01 CVC recirculation pump valve room	CVC discharge line	20" concrete	10" concrete	20" concrete	10" concrete	24" concrete	24" concrete
40'-0"	112	Module 02 CVC recirculation pump valve room	CVC discharge line	20" concrete	10" concrete	20" concrete	10" concrete	24" concrete	24" concrete

Table 12.3-5: Reactor Building Shield Wall Geometry (Continued)

Elevation (Note 1)	Room No. (Note 1)	Room Name (Note 1)	Radioactive Source	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling
40’-0”	115	Module 03 CVC recirculation pump valve room	CVC discharge line	20” concrete	10” concrete	20” concrete	10” concrete	24” concrete	24” concrete
40’-0”	123	Module 04 CVC recirculation pump valve room	CVC discharge line	20” concrete	10” concrete	20” concrete	10” concrete	24” concrete	24” concrete
40’-0”	126	Module 05 CVC recirculation pump valve room	CVC discharge line	20” concrete	10” concrete	20” concrete	10” concrete	24” concrete	24” concrete
40’-0”	129	Module 06 CVC recirculation pump valve room	CVC discharge line	20” concrete	10” concrete	20” concrete	10” concrete	24” concrete	24” concrete
40’-0”	-	Pipe chase - SRW resin transfer line	SRW resin transfer line	24” concrete	24” concrete	24” concrete	48” concrete	-	-
55’-0”	207	Modules 01-03 CVC HX valve gallery	CVC discharge line, CVC HXs	20” concrete	48” concrete	20” concrete	48” concrete	24” concrete	29” concrete
55’-0”	208	Module 01 CVC HX room	CVC discharge line, CVC HXs	20” concrete	20” concrete	48” concrete	48” concrete	24” concrete	24” concrete
55’-0”	209	Module 02 CVC HX room	CVC discharge line, CVC HXs	20” concrete	20” concrete	48” concrete	20” concrete	24” concrete	24” concrete
55’-0”	210	Module 03 CVC HX room	CVC discharge line, CVC HXs	20” concrete	48” concrete	48” concrete	20” concrete	24” concrete	24” concrete
55’-0”	213	MHS equipment room	MHS HT	18” concrete	18” concrete	48” concrete	48” concrete	24” concrete	24” concrete
55’-0”	216	Modules 04-06 CVC HX valve gallery	CVC discharge line, CVC HXs	20” concrete	48” concrete	20” concrete	48” concrete	24” concrete	29” concrete
55’-0”	217	Module 04 CVC HX room	CVC discharge line, CVC HXs	48” concrete	48” concrete	20” concrete	20” concrete	24” concrete	24” concrete
55’-0”	218	Module 05 CVC HX room	CVC discharge line, CVC HXs	48” concrete	20” concrete	20” concrete	20” concrete	24” concrete	24” concrete
55’-0”	219	Module 06 CVC HX room	CVC discharge line, CVC HXs	48” concrete	20” concrete	20” concrete	48” concrete	24” concrete	24” concrete
55’-0”	-	Pipe chase - SRW resin transfer line	SRW resin transfer line	24” concrete	24” concrete	24” concrete	48” concrete	-	-
70’-0”	304	PCW equipment aisleway	PCW HXs	48” concrete	48” concrete	48” concrete	48” concrete	24” concrete	24” concrete

Table 12.3-5: Reactor Building Shield Wall Geometry (Continued)

Elevation (Note 1)	Room No. (Note 1)	Room Name (Note 1)	Radioactive Source	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling
70’-0”	-	Pipe chase - SRW resin transfer line	SRW resin transfer line	24” concrete	24” concrete	24” concrete	48” concrete	-	24” concrete
70’-0”	-	Pipe chases - Modules 01-03 CVC discharge lines	CVC discharge lines	16” concrete	16” concrete	48” concrete	16” concrete	-	-
70’-0”	-	Pipe chases - Modules 04-06 CVC discharge lines	CVC discharge lines	48” concrete	16” concrete	16” concrete	16” concrete	-	-
85’-0”	-	Pipe chases - Modules 01-03 CVC discharge lines	CVC discharge lines	16” concrete	16” concrete	48” concrete	16” concrete	-	-
85’-0”	-	Pipe chases - Modules 04-06 CVC discharge lines	CVC discharge lines	48” concrete	16” concrete	16” concrete	16” concrete	-	-
100’-0”	-	Pipe chases - Modules 01-03 CVC discharge lines	CVC discharge lines	16” concrete	16” concrete	48” concrete	16” concrete	-	16” concrete
100’-0”	-	Pipe chases - Modules 04-06 CVC discharge lines	CVC discharge lines	48” concrete	16” concrete	16” concrete	16” concrete	-	16” concrete

Note 1: Figure 1.2-8 through Figure 1.2-18 depict room locations.
Note 2: A 1" steel plate is placed above the entrance to the CVCS demineralizer valve gallery.
Note 3: The vertical pipe chase enclosure containing the CVCS resin transfer line the to the SRWS extends from EL 25'-0" to EL 81'-0".
Note 4: The reactor pool walls are typically 48" concrete, with the exception of areas immediately north, south, and east of the operating RXMs, which are described as "Module bays" with thier own entry in the table.
Note 5: The vertical pipe chase enclosures containing the CVCS discharge lines from the RXMs to the CVC heat exchangers extend from EL 70'-0" to EL 111'- 7".

Table 12.3-6: Radioactive Waste Building Shield Wall Geometry

Room Number (Note 1)	Source Term	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling	Labryinth Walls
005	LRW Processing Equipment	Exterior subgrade wall	Exterior subgrade wall	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Entrance) - 24” Concrete (Drum dryer Area) - 24” Concrete (Drum dryer Accumulator Tank Area) - 24” Concrete (Ion Exchanger area) - 24” Concrete (Charcoal Filter Area)
008	LRW LCW Sample Tank Pumps	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Doorway) - 12” Concrete (Pump Divider)
009	LRW HCW Sample Tank Pumps	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Doorway) - 12” Concrete (Pump Divider)
010	SRW PST Pumps	24” Concrete	36” Concrete	36” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Doorway) - 12” Concrete (Pump Divider)
024	LRW LCW Collection Tank Pumps	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Doorway) - 12” Concrete (Pump Divider)
025	LRW HCW Collection Tank Pumps	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Doorway) - 12” Concrete (Pump Divider)
026	SRW SRST Pumps	36” Concrete	36” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	- 24” Concrete (Doorway) - 12” Concrete (Pump Divider)
011	LRW LCW Sample Tank	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
012	LRW LCW Sample Tank	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
018	LRW LCW Collection Tank	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
019	LRW LCW Collection Tank	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
013	LRW HCW Sample Tank	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A

Table 12.3-6: Radioactive Waste Building Shield Wall Geometry (Continued)

Room Number (Note 1)	Source Term	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling	Labryinth Walls
014	LRW HCW Sample Tank	24” Concrete	36” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
020	LRW HCW Collection Tank	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
021	LRW HCW Collection Tank	24” Concrete	36” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
015	SRW PST	36” Concrete	36” Concrete	36” Concrete	36” Concrete	Facility Basemat	36” Concrete	N/A
016	SRW PST	36” Concrete	36” Concrete	36” Concrete	36” Concrete	Facility Basemat	36” Concrete	N/A
022	SRW SRST	36” Concrete	36” Concrete	36” Concrete	36” Concrete	Facility Basemat	36” Concrete	N/A
023	SRW SRST	36” Concrete	36” Concrete	36” Concrete	36” Concrete	Facility Basemat	36” Concrete	N/A
017	Pipe Chase	24” Concrete	24” Concrete	24” Concrete	24” Concrete	Facility Basemat	24” Concrete	N/A
029	PCU FLT	30” Concrete	Exterior subgrade wall	Exterior subgrade wall	30” Concrete	Facility Basemat	30” Concrete	N/A
030	PCU FLT	30” Concrete	30” Concrete	Exterior subgrade wall	30” Concrete	Facility Basemat	30” Concrete	N/A
032	PCU IX	24” Concrete	30” Concrete	Exterior subgrade wall	24” Concrete	Facility Basemat	24” Concrete	N/A
033	SRW Drum Storage	36” Concrete	36” Concrete	24” Concrete	Exterior subgrade wall	Facility Basemat	24” Concrete	20” Concrete
034	SRW HIC Storage	36” Concrete	36” Concrete	36” Concrete	Exterior subgrade wall	Facility Basemat	36” Concrete	N/A
035	SRW HIC Filling	36” Concrete	36” Concrete	36” Concrete	Exterior subgrade wall	Facility Basemat	36” Concrete	N/A
037	GRW Vapor Condensers / Gas Coolers	24” Concrete	36” Concrete	24” Concrete	Exterior subgrade wall	Facility Basemat	24” Concrete	N/A
038	GRW Charcoal Beds	24” Concrete	24” Concrete	24” Concrete	Exterior subgrade wall	Facility Basemat	24” Concrete	24” Concrete

Note 1: Refer to Figure 1.2-22 through Figure 1.2-24 for room locations.
Note 2. The equivalent attenuation to an additional 4.5 inches of lead is provided for a HIC process shield.
Note 3: The equivalent attenuation to an additional one inch of steel in addition to the LRWS process skid.
Note 4: A penetration to the HIC storage room is modeled with a 1ft concrete shadow shield.

Table 12.3-7: Radioactive Waste Building Radiation Shield Doors

Radioactive Waste Building Door #	Radioactive Waste Building Room # (Note 1)	Radioactive Waste Building Room Description	Door Shielding (Note 2)
034A	034	SRWS HIC storage room	75% equivalent wall
035A	035	SRWS HIC filling room	20% equivalent wall
015A	015	SRWS PST A room	45% equivalent wall
016A	016	SRWS PST B room	45% equivalent wall
022A	022	SRWS SRST A room	100% equivalent wall
023A	023	SRWS SRST B room	100% equivalent wall
030A	030	PCWS filter A room	60% equivalent wall
029A	029	PCWS filter B room	60% equivalent wall

Note 1: Figure 1.2-22, Figure 1.2-23 and Figure 1.2-24 depict room locations.

Note 2: Door shielding is given as a percentage of the effective shielding provided by the barrier in which the door is located.

Table 12.3-8: Fixed Area and Airborne Radiation Monitors Post-Accident Monitoring Variables

Area	Monitor	Estimated Dynamic Detection Range	Principal Parameter measured	Basis for Dynamic Range	PAM Variable Type
Under bioshield monitors	Fixed area	1E+0 to 1E+7 rem/hr	gamma	RG 1.97, Rev. 5 Equipment Qualification Post-Accident Radiological Source Term	Type B, Type C and Type F
Hot lab	Fixed area	1E-1 to 1E+4 mrem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Hot lab	Fixed airborne	3E-10 to 1E-6 µCi /cc	Cs-137: γ	RG 1.97, Rev. 5 Radiological Source Term	Type E
Safety instrument rooms	Fixed area	1E-1 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
EDAS switchgear rooms	Fixed area	1E-1 to 1E+4 mrem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
MCR envelope - main control room ARM	Fixed area	1E-1 to 1E+4 mrem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
MCR envelope - MCR area airborne radiation monitor	Fixed airborne	1E-7 to 1E-1 µCi/cc 1E-10 to 1E-6 µCi/cc 1E-10 to 1E-5 µCi/cc	Kr-85, Xe-133: β Cs-137: γ, β I-131: γ	RG 1.97, Rev. 5 ANSI/HPS 13.1-2011	Type E
Technical support center	Fixed area	1E-1 to 1E+4 mrem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Primary Sampling Equipment	Fixed Area	1E-1 to 1E+4 mrem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Containment Sampling System Equipment	Fixed Area	1E-1 to 1E+4 mrem/hr	gamma	RG 1.97, Rev. 5 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Refuel Pool Area Continuous Airborne Monitor	Fixed airborne	3E-7 to 1E+4 µCi/cc 5E-12 to 1E+2 µCi/cc 4E-12 to 1E+2 µCi/cc	Kr-85, Xe-133: β Cs-137: γ, β I-131: γ	Regulatory Guide 1.97, Rev. 5 Radiological Source Term ANSI 42.18-2004	Type E
Refuel Pool Area	Fixed Area	1E-4 to 1E+4 mrem/hr	gamma	ANSI/ANS-HPSSC-6.8.1-1981	Type E
North Steam Gallery Continuous Airborne Monitor	Fixed Airborne	3E-7 to 1E+4 µCi/cc 5E-12 to 1E+2 µCi/cc 4E-12 to 1E+2 µCi/cc	Kr-85, Xe-133: β Cs-137: γ,β I-131: γ	Regulatory Guide 1.97, Rev. 5 Radiological Source Term ANSI 42.18-2004	Type E
South Steam Gallery Continuous Airborne Monitor	Fixed Airborne	3E-7 to 1E+4 µCi/cc 5E-12 to 1E+2 µCi/cc 4E-12 to 1E+2 µCi/cc	Kr-85, Xe-133: β Cs-137: γ,β I-131: γ	Regulatory Guide 1.97, Rev. 5 Radiological Source Term ANSI 42.18-2004	Type E

Table 12.3-9: Fixed Airborne Radiation Monitors

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 25’	Degasifier room A	1 / Noble gas	Kr-85, Xe-133: ß	3E-4 to 1E+3 µCi / cc	No
	Degasifier room B	1 / Noble gas	Kr-85, Xe-133: ß	3E-4 to 1E+3 µCi / cc	No
Reactor Building elevation 85’	Hot lab	1 / Particulate	Cs-137: γ	3E-10 to 1E-6 µCi /cc	Yes / E
Reactor Building Elevation 100'-0"	Refuel Pool North Steam Gallery South Steam Gallery	1 / Noble gas 1 / Particulate 1 / Iodine	Kr-85, Xe-133: ß Cs-137: γ, ß I-131: γ	3E-7 to 1E+4 µCi / cc 5E-12 to 1E+2 µCi / cc 4E-12 to 1E+2 µCi / cc	Yes / E
Radioactive Waste elevation 100’	Hot shop	1 / Particulate	Cs-137: γ	3E-10 to 1E-6 µCi /cc	No
Radioactive Waste Building elevation 70’	Gaseous radioactive waste process tank area	1 / Noble gas 1 / Particulate 1 / Iodine	Kr-85, Xe-133: ß Cs-137: γ I-131: γ	3E-7 to 1E-2 µCi /cc 3E-10 to 1E-6 µCi /cc 3E-10 to 5E-8 µCi /cc	No
Control Building elevation 125’	Main Control Room	1/Noble Gas 1/Particulate 1/Iodine	Kr-85, Xe-133: β Cs-137: γ, β I-131: γ	1E-7 to 1E-1 µCi/cc 1E-10 to 1E-6 µCi/cc 1E-10 to 1E-5 µCi/cc	Yes / E

Table 12.3-10: Fixed Area Radiation Monitors

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 25’	CVCS filters	6 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	RWDS sump pump areas	4 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Reactor Building elevation 40’	CVCS recirculation rooms	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Reactor Building elevation 55’	PCWS equipment	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	CVCS recirculation pumps	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	MHS heater room	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Reactor Building elevation 70’	PSS sample panel	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
	PCWS heat exchanger	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Safety I&C and EDAS equipment rooms	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
Reactor Building elevation 85’	Refuel Bridge	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Refuel Pool	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
	Safety I&C and EDAS equipment rooms	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
	Hot Lab	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
Reactor Building elevation 100’	North PSS sample panel	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
	South PSS sample panel	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	Yes / E
	Inside bioshield monitor	12 / gamma-sensitive	gamma γ	1E0 to 1E+7 rem/hr	Yes / B & C & F
Reactor Building elevation 126’	Module Maintenance Center	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	North Equipment	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	South Equipment	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Reactor Building elevation 146'	North RBVS AHU	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	South RBVS AHU	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Control Building elevation 125'	Main CRE	1 / gamma-sensitive	γαμμα γ	1E-1 to 1E+4 mrem/hr	Yes / E
	Technical support center	1 / gamma-sensitive	γαμμα γ	1E-1 to 1E+4 mrem/hr	Yes / E

Table 12.3-10: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Radioactive Waste Building elevation 70’	LRWS equipment area	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Reactor Building Access	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	RWDS sump pumps	4 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	LRWS storage tank pump rooms	4 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	SRWS storage tank pump rooms	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	SRWS HIC storage room	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 rem/hr	No
	PCWS filter room	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Waste drum storage room	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Radioactive Waste Building elevation 100’	Module Import Trolley Bay	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Class A Waste Storage Room	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Waste management control room	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
	Hot Shop area	1 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Turbine Building elevation 100’	General area	2 / gamma-sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No
Radioactive Waste Building Elevation 119'	RBV Filter Area	1 / gamma sensitive	gamma γ	1E-1 to 1E+4 mrem/hr	No

Table 12.3-11: NuScale Power Plant Systems with Nuclear Regulatory Commission Regulatory Guide 4.21 Evaluation

System Code	System Name	System Code	System Name
ABS	auxiliary boiler system	PCWS	pool cooling and cleanup system
ACCS	air cooled condenser system	PLDS	pool leakage detection system
BPDS	balance-of-plant drain system	PSS	process sampling system
CES	containment evacuation system	RBVS	Reactor Building HVAC system
CFDS	containment flooding and drain system	RCCWS	reactor component cooling water system
CHWS	chilled water system	RCS	reactor coolant system
CPS	condensate polisher resin regeneration system	RWBVS	Radioactive Waste Building HVAC system
CRVS	normal control room HVAC system	RWB	Radioactive Waste Building
CVCS	chemical and volume control system	RWDS	radioactive waste drain system
DHRS	decay heat removal system	RXB	Reactor Building
DWS	demineralized water system	SCWS	site cooling water system
FWS	condensate and feedwater system	SRWS	solid radioactive waste system
GRWS	gaseous radioactive waste system	UHS	ultimate heat sink
LRWS	liquid radioactive waste system	UWS	utility water system
MSS	main steam system		

Table 12.3-12: Regulatory Guide 4.21 Design Features for Auxiliary Boiler System

Objective	Design Features
Objective 1- Minimize the potential for leaks or spills and provide containment areas	The ABS uses proven, corrosion resistant material for the system components in accordance with industry codes and standards. This design feature also applies to Objective 3.
	The ABS distribution piping uses welded pipe construction as much as practicable. This design feature minimizes the potential for contamination to spread. This design feature also applies to Objectives 3 and 4.
	The valves are rated specifically for the required ABS pressure and temperature with hard face seats used for sealing to minimize leakage.
Objective 2- Provide leak detection capability	The ABS is provided with several pressure, level, and flow instruments throughout the system to monitor system performance.
	The ABS steam header is designed such that only one module can supply steam to the header at any given moment.
	The ABS has radiation monitoring on vents and drains. If radiation is detected, steam supply valves are shut and an alarm alerts operations staff.
Objective 3- Reduce contamination to minimize releases, cross-contamination and waste generation	The ABS is designed to preclude contamination of connecting systems.
	Isolation valves are used to minimize the possibility of cross contamination among multiple NPMs in the event of a SG tube leak.
Objective 4- Facilitate decommissioning	Parts of the ABS are designed using modular vendor skids, which facilitates decommissioning.
Objective 5- Operating programs and documentation	COL item
Objective 6- Site radiological environmental monitoring	COL item

Table 12.3-13: Regulatory Guide 4.21 Design Features for Air Cooled Condenser System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The air cooled condenser system (ACCS) is designed to minimize the effects of corrosion using a combination of corrosion resistant materials.
	Provisions for representative grab sampling are provided in the gaseous discharge line and allow for removal and return of the sample as part of an integrated skid system. The sample connections include a supply and return connection to facilitate purging after a sample is taken. This feature also prevents operator contact with radioactive material and minimizes contamination as the sample is returned to the process line. The sample is returned directly to the process line to minimize contamination.
Objective 2 - Provide leak detection capability	The ACCS condenser air removal system (CARS) has radiation monitors that meet the requirements of a continuous radiation monitor per the EPRI primary to secondary leakage guidelines. Manual sampling points are available in the event the radiation monitors are inoperable.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	Steam generated in the SGs is not radioactive during regular operating conditions. In the unlikely event of a primary to secondary system leak due to steam generator tube failure, the MSS and the FWS provide for closure of isolation valves to minimize the propagation of contamination into the CARS.
	Radiation monitors have two setpoints to minimize release of contamination to the environment.
Objective 4 - Facilitate decommissioning	The ACCS has removable/maintainable/replaceable outside of radiation field and not expected to have contamination for regular operation
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-14: Regulatory Guide 4.21 Design Features for Balance-of-Plant Drain System

Objective	Design Features
Objective 1- Minimize the potential for leaks or spills and provide containment areas	The BPDS uses proven materials in accordance with ASME B31.1. This design feature reduces the potential for leaks.
	The wastewater and chemical waste collection tanks are enclosed in stainless-steel-lined concrete containments to serve as two boundaries against leakage to the environment. This design feature also applies to Objective 3.
	The BPDS contains above ground and below ground SSC. Outdoor SSC located above ground are provided with freeze protection insulation and drains to prevent rupture from freezing.
	The BPDS has high-level alarms on the wastewater and chemical waste collection tanks that indicate overflow.
Objective 2 - Provide leak detection capability	The wastewater and chemical waste collection tanks are provided with leak detection capabilities. Radiation monitors are located on drain lines with the potential to be contaminated.
	The BPDS has alarms associated with level, pressure, radiation, and leak detection.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The BPDS piping and component materials are designed for the 60-year design life of the plant.
	Drain lines in the BPDS are sloped, which reduces accumulation of solid forms of radioactivity within the piping.
	Floor drains and pump suctions are provided with maintainable strainers to promote system cleanliness and longevity.
Objective 4 - Facilitate decommissioning	The BPDS collects and segregates normally non-radioactive liquid waste from areas associated with power-related or process-related outside the RCA. Being outside the RCA allows access to facilitate decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-15: Regulatory Guide 4.21 Design Features for Containment Evacuation System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The CES uses welded components and welded piping to the extent practicable to minimize potential for leaks and spills.
	Stainless steel is the material for wetted components, including the sample vessel, condenser, and valves, to provide corrosion resistances and to minimize degradation and particulate generation. This design feature also applies to Objective 3.
	The CES equipment vents and drains are routed in closed lines to the RWDS. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The gaseous discharge path has radiation monitors for monitoring leaks.
	The CES control system uses input from pressure, temperature, sample vessel level, valve position, and radiation instruments to monitor system performances and control vacuum pump speed. The instrumentation on the CES sample vessel provides indication that the system may have developed a leak.
Objective 3 - Reduces contamination to minimize releases, cross-contamination and waste generation	The CES sample vessel has provisions for flushing with demineralized water and for purging with nitrogen.
Objective 4 - Facilitate decommissioning	Most CES components are specified for the 60-year design life of the plant. The filters are specified for a 10-year life with planned replacement. This design feature also applies to Objective 3.
	The CES components outside of the CNV are in shielded equipment galleries separate from SSC that are expected to generate high radiation levels. This feature reduces equipment contamination and reduces the effort for decommissioning the CES components.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-16: Regulatory Guide 4.21 Design Features for Containment Flooding and Drain System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	CFDS vents and drains are sent to the RWDS via closed piping. Pump leakage and leakage from the containment drain separator tank is collected by the RWDS. This design feature also applies to Objective 3.
	The CFDS piping is designed using proven materials in accordance with ASME B31.1. This design feature reduces the potential for leaks.
	The CFDS piping and wetted components are designed using corrosion-resistant materials that reduce the potential for leaks and contamination.
Objective 2 - Provide leak detection capability	Pressure, flow, temperature, and level transmitters provided throughout the system monitor system performance and possible system pressure boundary failure.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	A HEPA filter is installed in the containment drain separator tank vent line to capture radioactive particles before discharging the air to the RBVS.
	High radioactivity measured by the CFDS process radiation monitor results in automatic isolation of the line on a high-radioactivity indication. This filtration and automatic function minimizes transfer of radioactive effluent to the RXB exhaust stack.
Objective 4 - Facilitate decommissioning	Embedded piping is minimized. This design prevents ground contamination and reduces decontamination effort during the decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-17: Regulatory Guide 4.21 Design Features for Chilled Water System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The CHWS uses proven materials that are corrosion-resistant materials for piping and components, like stainless steel, for piping (designed according to ASME B31.1).
	The CHWS water is a glycol solution to prevent freezing in the HVAC coils during the winter months.
Objective 2 - Provide leak detection capability	The CHWS has localized pressure monitoring at the interface with the GRWS and LRWS. Changes in pressure can indicate a leak. The chillers include factory-installed microprocessor control systems to monitor and control chiller functions.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The CHWS is maintained at a higher pressure than the GRWS and the LRWS at the heat exchanger. The CHWS is isolated to and from the LRWS skid if the pressure difference is less than the setpoint to avoid the possibility of the CHWS being at a lower pressure than LRWS.
Objective 4 - Facilitate decommissioning	The CHWS components are designed with elements that can be easily removed during decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-18: Regulatory Guide 4.21 Design Features for Condensate Polisher Resin Regeneration System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The CPS materials are selected based on compatibility to temperature, pressure, and chemistry of the various resins, demineralized water, sulfuric acid, and sodium hydroxide. Piping and non-replaceable component materials are consistent with the 60-year design life. CPS piping is designed to ASME B31.1. This material selection and design process reduces the likelihood of equipment failures and leaks. This design feature also applies to Objectives 3 and 4.
	Adequate workspace and laydown areas are provided to perform maintenance on equipment and other CPS components. Adequate space allows for preventative maintenance to reduce leaks. This also applies to Objectives 3 and 4.
Objective 2 - Provide leak detection capability	The CPS is has radiation monitors to alert personnel of changing radiological conditions, and to alert the MCR of previously undetected primary to secondary leakage manifesting during resin transfer.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The BPDS supports the CPS by accepting drains and neutralized water from the neutralization water pump. This feature minimizes the spread of contamination if the CPS becomes contaminated.
	The resin in the cation regeneration vessel is washed and recycled back to the CPS. This feature minimizes waste resin generation.
Objective 4 - Facilitate decommissioning	The CPS is located above ground in the Turbine Generator Building (TGB), in the yard near the TGB and the offsite chemical storage area. The above ground placement of many components facilitates decommissioning.
	The CPS structures, systems, and components are not located in buildings or areas considered a radiation hazard, which facilitates decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-19: Regulatory Guide 4.21 Design Features for Normal Control Room Ventilation System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The CRVS air handling units provide conditioned air to maintain the control room's environment within an acceptable range. If the outside air becomes radioactively contaminated, the outside air is filtered. This feature minimizes the potential for contaminated air from entering the system. This design feature also applies to Objective 3.
	The CRVS components are designed in accordance with applicable industry codes and standards.
	The CRVS ductwork is designed to various leakage classes per ASME AG-1. This feature limits the allowable leakage from the system.
	Drains from the potentially contaminated CRVS filter unit are routed to the BPDS. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The CRVS has radiation monitors both upstream and downstream of the filter unit.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The CRVS maintains the CRB at a positive ambient pressure relative to the RXB and the outside atmosphere to control the ingress of potentially airborne radioactivity.
	The CRVS filter unit filters the outside make-up air when radiation is detected in the outside air intake ductwork.
	Upon detection of a high radiation level in the outside air intake, the system is realigned so that 100 percent of the outside air passes through the CRVS filter unit, containing HEPA and charcoal filters, to filter outside air and minimize radiation exposure to personnel within the CRB.
Objective 4 - Facilitate decommissioning	Most of the CRVS is above ground. This design feature facilitates decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-20: Regulatory Guide 4.21 Design Features for Chemical and Volume Control System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The CVCS tanks, piping, and valves are designed to the applicable ASME code with corrosion resistant materials to minimize degradation and particulate generation over the 60-year design life of the plant. Welded construction minimizes the potential for leaks as practicable.
	A chemical mixing allows the addition of chemicals into the primary coolant system to minimize the potential for primary water stress-corrosion cracking and resultant equipment failures.
	The discharge lines from pressure relief valves are routed to the RWDS.
	The floor and equipment drains are provided for CVCS and MHS equipment to direct spills or leaks to the nearest drain hub or sump in the RWDS by gravity.
	The floor in each equipment room is sloped to facilitate the collection of leaks and spills.
Objective 2 - Provide leak detection capability	The CVCS ion exchange vessels, filters, and resin traps are in cubicles equipped with moisture sensors for early leak detection. The CVCS heat exchangers and pumps drain to a RWDS sump.
	The CVCS expansion tank has a level transmitter that monitors the water level in the tank to provide indication of CVCS leakage.
	The CVCS provides the capability to inject argon into the RCS, increasing the sensitivity of SG tube leak detection.
	The CVCS includes flow, temperature, and pressure indications throughout the system to monitor system performance, including mass flow rate in the CVCS makeup line, the letdown line to the LRWS, and the NPM injection and discharge lines. The mass flow mismatch warns operators of abnormal conditions.
	The ARMs near the components assist in detecting potential airborne leaks.
	Most indicators have remote capabilities to ensure prompt alerting of operators.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	A continuous sample connection is provided on the CVCS module discharge line to the PSS, which is also returned to the CVCS to reduce waste generation.
	Utility connections are designed with a minimum of two barriers to prevent cross contamination of non-radioactive systems from potentially radioactive systems. For example, DWS and the boron addition system are aligned to independent CVCS pumps for boration and dilution with dual containment isolation valves for each CVCS containment penetration. There are double isolation valves upstream and downstream of each CVCS recirculation pump, ion exchanger, resin trap and filter.
	An automatic feature diverts flow around the ion exchangers if the fluid temperature exceeds the high temperature set point for ion exchanger resin protection, preventing resin damage and minimizing waste generation.
	The ion exchange and reactor coolant filters remove particulate and impurities from the CVCS water. This feature reduces the contamination of downstream equipment.
Objective 4 - Facilitate decommissioning	The CVCS components are designed for the 60-year design life of the plant.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-21: Regulatory Guide 4.21 Design Features for Decay Heat Removal System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The decay heat removal system (DHRS) piping and components are designed to ASME standards, and use welded construction and corrosion resistant materials. This feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The DHRS is provided with temperature indications, calculated water level in the SG, and steam and feed water flow mass differences to indicate system performance and identify system leaks.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	A portion of the DHRS is located above the reactor pool water level under the bioshield, with the remaining portion of the system located under water in the reactor pool. This arrangement reduces the potential for spreading contaminated steam or fluid into the RXB atmosphere.
Objective 4 - Facilitate decommissioning	The DHRS is designed for the 60-year design life of the plant and is designed, to the extent practicable, for removal.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-22: Regulatory Guide 4.21 Design Features for Demineralized Water System

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	The DWS tanks and piping system are designed and fabricated according to industry codes and standards. This feature minimizes the potential for leaks. This feature also applies to Objective 3.
	The DWS uses stainless steel piping, valves, and tanks to reduce the potential that corrosion results in leaks. This feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The DWS incorporates radiation monitors to detect the backflow of contamination into the DWS.
Objective 3 - Reduce contamination to minimize releases, cross-contamination & waste generation	The DWS contains no radioactive substances but does interface with radioactive systems. User distribution lines are equipped with backflow preventers to prevent contamination of the DWS. Isolation valves allow for system isolation in the event of a line break or other abnormal condition .
Objective 4 - Facilitate decommissioning	The DWS components are designed for 60-year design life with elements like aboveground piping that can be easily removed during decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-23: Regulatory Guide 4.21 Design Features for Condensate and Feedwater System

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	The FWS piping and components are designed to appropriate industry codes and standards, and are, where appropriate, made of corrosion resistant material that is compatible with the operating environment of FWS. This design feature reduces the potential for leaks.
	The FWS includes nonsafety-related valves downstream of some safety-related valves. These redundant valves reduce the potential for leaks.
Objective 2 - Provide leak detection capability	Grab sampling locations are provided in several locations to monitor the chemical composition and radiation contamination level of the FWS.
Objective 3 - Reduce contamination to minimize releases, cross-contamination & waste generation	The FWS includes a condensate polisher skid that treat and clean the feedwater to remove corrosion products and ionic impurities. A condensate strainer is located upstream of each condensate pump to filter out foreign materials. These design features can reduce cross-contamination.
	The FWS drains to the BPDS. This also applies to Objective 2.
Objective 4 - Facilitate decommissioning	The FWS piping is routed above the ground, as much as practicable. This routing facilitates the decommissioning effort.
	Many FWS components are located outside the RCA, making them easier to decontaminate and decommission.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-24: Regulatory Guide 4.21 Design Features for Gaseous Radioactive Waste System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The GRWS uses low leakage valves (e.g., packless metal diaphragm valves) wherever possible.
	Embedded piping is minimized, and liquid containing lines are sloped to facilitate drainage (also meets Objective 4 of RG 4.21).
	The GRWS components are designed using welded construction to minimize leaks.
	In-line radiation monitors minimize contamination by returning samples to the process stream and do not require interfaces with non-contaminated systems for purging (also meets Objective 3 of RG 4.21).
Objective 2 - Provide leak detection capability	Radiation monitors are provided near system components and cubicles to detect leakage from the GRWS.
	Gas analyzers detect potentially explosive gas mixtures to enable operators to prevent an event that challenges system boundary integrity.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The GRWS component materials are selected for reliable service for the 60-year design life, reducing waste generation from replacing components.
	Components are designed for reliability using industry codes and standards, per RG 1.143, minimizing the potential for cross-contamination and releases.
	Welding techniques and material finishes are designed to provide smooth internal surfaces for decontamination. This design also meets Objective 1 and 4.
	The GRWS gas sampler return samples to the process stream while also containing provisions to take a grab sample.
	The GRWS piping arrangement is routed to reduce the total length of piping.
Objective 4 - Facilitate decommissioning	The GRWS components are designed for 60-year design life and easily removable elements during decommissioning
	Nitrogen is provided to purge and decontaminate the system internals before disassembly (also meets Objective 3 of RG 4.21).
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-25: Regulatory Guide 4.21 Design Features for Liquid Radioactive Waste System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containments areas	The LRWS components, which include collection and sample tanks, are made of stainless steel material and are of welded construction to minimize the potential for leakage and unintended contamination of the facility and the environment. This design feature also applies to Objectives 3 and 4.
	The LRWS tanks are designed with sufficient capacity to provide liquid waste storage during normal operation, including anticipated operational occurrences. Redundant collection and sample tanks are provided to minimize the interruption of normal processing operation.
	The LRWS is designed without embedded piping and a minimum of buried piping. Buried LRWS piping or influent streams to the LRWS that are from outside of buildings is double walled with leak detection. Pipe leakage is either directed back to RWDS sumps or prevented by a pressure differential, that provides two barriers and thus prevents the spread of contaminations. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	LRWS tanks are designed with level indicating transmitters, protected by isolated diaphragms, to provide safe operation of the SSC. The instruments provide alarms to the MCR and the waste management control room (WMCR) to alert operators to take appropriate action upon high liquid tank level.
	The LRWS tank cubicles are provided with leak detection instrumentation to alarm for operator actions if leakage occurs. The LRWS leak detection is designed to detect small leakage quantities, which facilitates early detection.
	Radiation monitors are provided in the LRWS processing area to detect airborne activity from leaks.
	Pump bed plates that drain to the nearest floor drain are provided for LRWS transfer pumps to collect pump leakages.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The SSC are designed with life-cycle planning through the use of materials compatible with the chemical, physical, and radiological environment in the nuclear industry, thus minimizing waste generation.
	The floor drains, equipment drains, and chemical drains are collected and segregated in the HCW and LCW tanks accordingly. The LRWS is a batch operation process; therefore, treatment of HCW and LCW type wastes are determined through sampling and analyses, then the wastes are routed to appropriate treatment processes that minimize cross-contamination and potential mixed waste generation.
	The LRWS process piping containing contaminated solids are sized with sufficient velocities and sloped as necessary to facilitate flow and prevent the settling of solids.
	Where interfaces exist with utilities, such as compressed air and gas and service water, double isolation is provide to prevent cross contamination of clean systems by LRW.
	Connections to the demineralized water break tank are provided to flush and clean LRWS tanks to facilitate maintenance, inspection and removal of radioactive components.
	The clean-in-place skid is provided to flush and clean the reverse osmosis skid to facilitate maintenance
Objective 4 - Facilitate Decommissioning	The LRWS is comprised of skid packages to facilitate eventual removal and decommissioning. The SSC are designed for a 60-year dessign life of plant.
	The LRWS is designed to aid decontamination and decommissioning activities. Design features, such as material type, welding techniques, surface finishes, and equipment placement, are incorporated into the design to facilitate ALARA and minimize decontamination and minimize waste generation

Table 12.3-25: Regulatory Guide 4.21 Design Features for Liquid Radioactive Waste System (Continued)

Objective	Design Features
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-26: Regulatory Guide 4.21 Design Features for Main Steam System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The MSS piping is made of corrosion-resistant materials and designed in accordance with ASME code. The fluid chemistry is maintained to reduce corrosion and minimize the potential for system leaks. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The steam line radiation monitors are designed to detect SG tube leaks.
	Sampling capability is provided for the MSS to analyze the chemical composition.
	Radiation monitors in the RBVS and ARMs can assist in detecting steam leaks into the RXB.
	The air-cooled condenser system radiation monitors are designed to monitor effluents coming from the CARS and are designed to detect SG tube leaks.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	Fluid leaks from the MSS in the RXB and the TGB are collected by the RWDS and BPDS, respectively.
	The CPS is provided to cleanup the secondary coolant to reduce the level of radionuclide contamination and minimize releases, cross-contamination, and waste generation.
Objective 4 - Facilitate decommissioning	The MSS components are designed for the 60-year design life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning. The piping and equipment are above ground, as much as practicable.
	The MSS piping has smooth surfaces to facilitate decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-27: Regulatory Guide 4.21 Design Features for Pool Cooling and Cleanup System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	Leakage is captured to prevent system leaks, and drainage of the system is collected and processed for radioactive waste cleanup. This design feature also applies to Objective 3.
	The PCWS uses proven, corrosion-resistant materials for piping and components, like stainless steel, for piping (designed according to ASME B31.1), valves, pumps, filter housings, demineralizers, and wetted parts of the heat exchangers.
	The PCWS filters and demineralizers, with their resin traps, are segregated and individually shielded, and resin transfer lines are routed through shielded chases. Drainage from the PCWS filters is sent to the RWDS for collection before being sent to the LRWS for further processing.
	Major system equipment is supplied with drains that connect to the RWDS that enables the equipment to be drained to sumps while keeping contaminated water within system piping to help maintain contamination control during maintenance.
	The PCWS surge control tank is designed with a catch basin to contain the total volume the tank plus freeboard. The catch basin contains the leakage and transfers the fluid to the LRWS collection tanks. Using stainless steel components and piping reduces corrosion and the potential for leaks.
	The pool water from a surge event is processed through the PCWS and stored in the PCWS surge control storage tank instead of discharging it to the LRWS. This water is recycled back when pool level is low, reducing the amount of waste water to be processed by the LRWS. This design feature also applies to Objective 3.
	The use of underground or buried piping is minimized to the extent practicable. The portions of PCWS pool surge control subsystem piping that are embedded underground or in a yard area pipe chase include guard pipe, reducing the potential for contamination of the environment. This design feature also applies to Objective 4.
Objective 2 - Provide leak detection capability	The RWDS sumps located in the RXB are equipped with level transmitters to detect leakage.
	Although the SCWS is normally at a higher pressure than the PCWS, radiation detectors are provided in the site cooling water system to detect PCWS heat exchanger tube leaks into the SCWS.
	The PCWS demineralizer cubicle is equipped with a level switch to provide early leakage detection.
	The secondary containment tank placed around the PCWS surge control tank is equipped with leak detection capabilities. The secondary tank is covered and leak-proof. The secondary containment tank drain sump is also equipped with a level sensor that provides indication of potential leaks from the PCWS surge control storage tank or the piping located in the secondary tank.
	The PCWS is configured with pressure, flow, temperature, and conductivity instrumentation to monitor process conditions and to assist in identifying system leaks.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	Temperature transmitters upstream of the PCWS demineralizers protect the resin from being exposed to high temperature, preventing the damage of resins and reducing waste generation.
	The PCWS demineralizers and resin traps are provided with flushing capability to reduce the contamination levels within the system. This design feature also applies to Objective 4.
	The PCWS filters and demineralizers remove impurities and minimize pool contamination. The PCWS has the capability to recycle the pool water after being filtered and demineralized. This feature reduces waste generation.

Table 12.3-27: Regulatory Guide 4.21 Design Features for Pool Cooling and Cleanup System (Continued)

Objective	Design Features
Objective 4 - Facilitate decommissioning	The PCWS components are above ground and usage of long or embedded pipe is minimized. This design prevents ground contamination and facilitates decommissioning.
	The PCWS components are designed for the 60-year design life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-28: Regulatory Guide 4.21 Design Features for Pool Leakage Detection System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The PLDS has leak channels under the pool floor liner weld seam and attached to the pool walls to guide pool leaks to a drain header that leads to RWDS sumps. The PLDS uses welded channels and piping to minimize potential for leaks. Pool leaks are designed to flow through the leak channels to the RWDS sumps.
	The PLDS uses stainless steel material or equivalent corrosion resistant materials to prevent corrosion. This feature reduces the potential for leaks. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The RWDS sumps, located in the RXB, are equipped with level transmitters to detect leakage from the pool.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The PLDS is designed with corrosion resistant materials for the wetted parts (leak channels, piping, and valves). The selected material is compatible with the operating environment of the pool water.
Objective 4 - Facilitate decommissioning	The PLDS is designed for the 60-year design life of the plant. The PLDS components are above ground to the extent practicable. This design feature facilitates decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-29: Regulatory Guide 4.21 Design Features for Process Sampling System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The PSS uses proven materials to minimize leaks and contamination of the facility and the environment. Welded construction is used as appropriate to minimize the potential for leakage from contaminated systems.
	Reactor coolant samples are routed to the sample stations located in the chemistry hot lab for grab sample collection during normal operations. This design feature minimizes the potential for spills of samples in the RXB. In addition, the PSS is designed with minimum embedded or buried piping. The contaminated lines are routed through the pipe chases as much as practicable.
	The chemistry hot lab houses sample panels used for collecting reactor coolant grab samples. The purged sample is returned to the CVCS, and excess grab sample is drained into a sink that is hard piped to the RWDS. This design feature minimizes potential for leaks in the RXB.
	The floors are sloped to direct leakages or spills to the drain hubs leading to a sump in the RWDS. The slopped floors include the chemistry hot lab and other locations where sample panels are located in the RXB. This design approach prevents spread of contamination and contains the leakages.
	The PSS grab sample panels located in the chemistry hot lab of the RXB are equipped with a hood to minimize the airborne contamination and radioactive gases from grab samples. The hood discharges the gases into the RXB ventilation system.
Objective 2 - Provide leak detection capability	Area radiation monitors are provided in the chemistry hot lab and in areas where sample panels are located in the RXB.
	The PSS is designed with minimum embedded or buried piping to ensure potential leaks can be identified.
	Process radiation monitors provided in the non-radioactive interfacing systems provide leak detection capability.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The PSS components are designed with materials that are compatible with the operating environment. The wetted parts of piping, valves, pump, heat exchangers, analyzers, grab sample containers, instrumentation, and controls re made of stainless steel material to prevent corrosion.
	The automatic samples are returned back to the originating system as much as practicable to minimize waste generation.
	The grab sample panel located in the chemistry hot lab of the RXB is equipped with a hood to remove contaminated gases from the grab samples and discharge these gases to the RBVS.
	The PSS is designed with sample sinks to collect the excess of radioactive grab samples and transfer the drainage by gravity to the RWDS. These drains are hard piped to minimize leakage and cross contamination.
	Pipe connections to the clean systems are provided with adequate isolation capability and backflow prevention to prevent cross-contamination.
	The use of proven materials and welded construction for the piping and components minimizes leakage and unintended contamination of the facility and the environment.
	The PSS design includes the capability to isolate sample lines to mitigate the potential spread of contamination.
	The PSS sample sink is provided with flush water to clean the sink and flush the drain lines to the RWDS.
Objective 4 - Facilitate decommissioning	The PSS components are designed for the 60-year design life of the plant to the extent practicable,with easily removable elements during decommissioning.
	The PSS is designed with minimal embedded or buried piping. The contaminated lines are routed through the pipe chases as much as practicable.

Table 12.3-29: Regulatory Guide 4.21 Design Features for Process Sampling System (Continued)

Objective	Design Features
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-30: Regulatory Guide 4.21 Design Features for Reactor Building Heating, Ventilation, and Air Conditioning System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	Periodic in-place testing of the atmospheric portions of the RBVS is performed in accordance with RG 1.140 and ASME N510.
	The RBVS components are designed, fabricated and tested according to industry codes and standards, and constructed using corrosion resistant materials.
	The condensations drains from RBVS components are directed to RWDS drain sumps. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The RWDS sumps located in the RXB have level transmitters to detect leakage.
	The RBVS radiation monitors detect potential airborne contamination.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The RBVS maintains the RXB atmosphere at a negative pressure relative to the outside environment to prevent potentially contaminated air from leaking to the environment.
	The AHUs and fan coil units have filters to reduce the potential contamination levels of the air.
	The RBVS maintains air flow from areas of lesser potential contamination to areas of greater potential contamination to minimize the spread of contamination.
Objective 4 - Facilitate decommissioning	Smooth finished material is used as much as practicable for the equipment to minimize contamination of equipment and facilitates decommissioning.
	The RBVS exhaust ducts consist of straight runs and long radius elbows wherever possible to reduce the buildup of contaminated particulate.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-31: Regulatory Guide 4.21 Design Features for Reactor Component Cooling Water System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The reactor component cooling water system (RCCWS) is expected to be free of gamma-emitting radionuclides during normal operation but may contain tritiated water. The closed-loop system consists of piping and equipment that are made of stainless steel that is compatible with the RCCWS operating environment. This feature minimizes corrosion and potential leaks. This design feature also applies to Objective 3.
	The RCCWS is located in the RXB with sloped floors that direct drainage to the nearest drain hub. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The RCCWS is provided with radiation monitors downstream of loads that have a potential for a radioactive leak to detect cross contamination.
	The RCCWS drain tank has radiation monitoring and level instrumentation to provide indications of a leak.
	The SCWS has radiation monitors to detect cross-contamination from RCCWS heat exchanger leaks into the system.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The RCCWS components are selected for reliable service for the 60-year design life of the plant.
	The RCCWS is a closed-loop system that provides an intermediate cooling system between potentially radioactive heat loads and rejection of heat to the environment via the SCWS. The use of RCCWS as an intermediate cooling loop reduces the potential for releasing contamination.
	The RCCWS acts as an intermediate, closed-loop cooling system between radioactive systems and the non-radioactive SCWS. This design provides an additional barrier to ensure that potential contamination is contained within the RXB.
Objective 4 - Facilitate decommissioning	The RCCWS piping and equipment are made of stainless steel material to minimize degradation and particulate generation to facilitate decontamination and decommissioning. The DWS can be used for cleaning and flushing purposes to remove contamination.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-32: Regulatory Guide 4.21 Design Features for Reactor Coolant System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The RCS is entirely contained within the RPV, which is an ASME Boiler and Pressure Vessel Code Section III, Class 1 vessel, and is periodically inspected according to the applicable portions of ASME Section XI (Section 5.2.4).
	The RCS components are designed for the 60-year design life of the plant using nuclear industry-proven materials compatible with the operating environment. This also applies to Objective 4.
Objective 2 - Provide leak detection capability	Reactor coolant system leaks into the SG are detected by radiation monitors associated with the MSS or the CARS. Leakage monitoring for the SGs is in accordance with EPRI 97-06 (Section 5.4.1).
	The CES interfaces with the CNV and is used to detect leakage from the reactor coolant pressure boundary to satisfy GDC 30 (Section 5.2.5).
	The CVCS provides the capability to inject argon into the RCS, which increases the sensitivity to detect SG tube leaks using argon-41.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The CVCS filters and demineralizers are used to filter and clean the primary coolant, reducing the contamination levels in the RCS.
	Materials used in the RCS are low in nickel and cobalt content, to the maximum extent practical, to reduce contamination due to crud.
	The CVCS provides the capability to inject zinc into the RCS to help reduce corrosion product generation and deposition.
Objective 4 - Facilitate decommissioning	Sharp geometric discontinuities and recesses have been avoided to the extent practical in the RCS design in order to minimize flow dependent pressure loss and to minimize regions where activated corrosion products can accumulate. This design feature facilitates decommissioning.
	The RCS is designed as part of a module that facilitates decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL Item

Table 12.3-33: Regulatory Guide 4.21 Design Features for Radioactive Waste Building Heating, Ventilation, and Air Conditioning System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The RWBVS components are designed, fabricated, and tested according to industry codes and standards, and constructed using corrosion-resistant materials.
	The condensation drains from RWBVS components are directed to RWDS drain sumps. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The RWBVS radiation monitors detect potential airborne contamination.
	The RWDS sumps located in the RWB have level transmitters to detect leakage.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The RWBVS maintains the RWB atmosphere at a negative pressure relative to the outside environment to prevent potentially contaminated air from leaking to the environment.
	The RWBVS maintains air flow from areas of lesser potential contamination to areas of greater potential contamination to minimize the spread of contamination.
Objective 4 - Facilitate decommissioning	Smooth finished material is used as much as practicable for the equipment to minimize contamination of equipment and facilitate decommissioning.
	The RWBVS piping and ductwork are above ground. This design feature facilitates decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site Radiological Environmental Monitoring	COL item

Table 12.3-34: Regulatory Guide 4.21 Design Features for Radioactive Waste Building

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	The RWB equipment rooms are designed with curbs, sump pits, and stainless steel liners in tank cubicles. These features minimize the spread of contamination. This design feature also applies to Object 3.
Objective 2 - Provide leak detection capability	Objective 6
Objective 3 - Reduce contamination to minimize releases, cross-contamination & waste generation	The RXB is connected to the RWB via various subgrade and above grade aisle ways, chases, duct banks, or vestibules for utility and personnel access.
	SRW and LRW tank rooms containing radioactive liquids are lined with stainless steel in sufficient height to contain the failure of any single vessel or piece of equipment to support contamination minimization.
	The RWB design incorporates the separation of non-radiological activities/functions from the radiologically controlled activities and functions as most of the radiological activities are contained below-grade in a concrete structure which also controls airborne radioactivity, while the non-radiological functions are above-grade.
Objective 4 - Facilitate decommissioning	Liners or durable, deep penetrating, smooth-finish coatings are used for below-grade structures. This feature applies to the interior surface up to the grade level for floors and walls that form a boundary to the exterior.
	Structural surfaces with the potential for radioactive contamination are protected with a smooth coating of infused epoxy or equivalent and waterproofed or stainless steel lining to minimize absorption of contaminants
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site Radiological Environmental Monitoring	COL item

Table 12.3-35: Regulatory Guide 4.21 Design Features for Radioactive Waste Drain System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The RWDS uses proven materials and welded construction for the system components. This design feature reduces the potential for leaks, including the sump liners.
	Each sump has two pumps and a level instrument to facilitate automated pump starting and stopping. This design approach reduces the potential of sump overflow and the spread of contamination.
Objective 2 - Provide leak detection capability	For rooms that are expected to have high radiation, or are normally sealed, there is an isolation valve and level switch downstream of the floor drain.
	The RWDS sump tanks have leak detection capability in the interstitial space between the sump tank and the stainless steel lined concrete sump.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The RWDS components are selected for reliable service for the 60-year design life of the plant, reducing waste generation from replacing components.
	Loop seals have been provided at various elevations of the RXB and RWB to prevent air cross-contamination between the floors and the rooms with a common drain header.
	The chemical drain tank has a conical bottom design to reduce crud buildup.
	The process piping containing contaminated fluid is sloped to facilitate flow and reduce fluid traps. The transfer pumps in the RWDS have check valves in their discharge lines to prevent backflow from another sump or tank contaminating other sumps or systems.
	The RWDS equipment is made of stainless steel that is corrosion resistant with smooth surfaces to facilitate decontamination at the end of decommissioning in compliance with 10 CFR 20.1406 This design feature also applies to Objective 4.
	To the maximum extent possible, drain lines carrying radioactive waste are welded construction in order to minimize leaks, crud traps and hot spots.
	The RWDS sump tanks are strategically located to collect floor and equipment drains from various sources. The drains are segregated (floor, equipment, chemical, RCCWS, and detergent drains) for different handling and processing requirements to minimize cross-contamination.
	Equipment and floor drain sump tanks and other RWDS drain tanks are vented to the buildings’ HVAC exhaust systems to preclude airborne contamination.
	Upon detection of high radiation in the RCCWS drain tank, the plant control system alarms and interlock closes the valve back to the RCCWS expansion tank. High radiation in the RCCWS drain tank is the result of contamination of the RCCWS fluid and must be investigated.
	The RWDS drain lines and leak detection piping are sloped, atmospheric, gravity flow drain lines.
	With exception of RCCWS, the RWDS is not connected to nonradioactive systems in order to prevent inadvertent transfer of contaminated fluids to non-contaminated systems, in compliance with 10 CFR 20.1406.
Objective 4 - Facilitate decommissioning	The RWDS is designed with minimum embedded piping, as much as practicable. The use of embedded piping is limited to the RWDS drain lines in the RXB and RWB basemats and the RXB elevation 100’ floor slab, as well as the PLD channel drainage lines.
	Headers with multiple connected drain lines can be cleaned out via cleanout connections or the floor drains themselves, in accordance with generally recognized plumbing practice. Demineralized water can be temporarily connected to the cleanout connection and used to flush and unplug the lines if needed.

Table 12.3-35: Regulatory Guide 4.21 Design Features for Radioactive Waste Drain System (Continued)

Objective	Design Features
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site Radiological Environmental Monitoring	COL item

Table 12.3-36: Regulatory Guide 4.21 Design Features for Reactor Building

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	The RXB equipment rooms are designed with curbs and shielded cubicles. This feature minimizes the spread of contamination. (Also applies to Objective 3)
Objective 2 - Provide leak detection capability	Objective 6
Objective 3 - Reduce contamination to minimize releases, cross-contamination & waste generation	The RXB is connected to the RWB via various subgrade and above grade aisle ways, chases, duct banks or vestibules for utility and personnel access.
Objective 4 - Facilitate decommissioning	Liners or durable, deep penetrating, smooth-finish coatings are used for below-grade structures. This feature applies to the interior surface up to the grade level for floors and walls that form a boundary to the exterior.
	Structural surfaces with the potential for radioactive contamination are protected with a smooth coating of infused epoxy or equivalent and waterproofed or stainless steel lining to minimize absorption of contaminants
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-37: Regulatory Guide 4.21 Design Features for Site Cooling Water System

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	The SCWS is normally a clean (non-radioactive) system that supplies cooling water to heat loads in the RXB, Central Utility Building, and the TGB. The SCWS piping that is underground is concrete pipe or stainless steel A316/316L designed to American Water Works Association standards. The SCWS piping that is above ground is made of carbon steel designed to ASME B31.1. This material minimizes corrosion and reduces the potential for leaks. This design feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	The possibility of radioactive contamination occurs when a heat exchanger leaks. Radiation monitors are located directly downstream of RXB heat exchangers, and the cooling tower blowdown line before release to the environment.
	Grab sampling capability is provided to measure radiation contamination levels in the system.
Objective 3 - Reduce contamination to minimize releases, cross-contamination & waste generation	Radiation monitors that detect leaking heat exchangers have operator actuated isolation valves, except in cooling tower blowdown where isolation is automated in the event of the detection of radiation.
Objective 4 - Facilitate decommissioning	Site cooling water system components that interface with systems that have the potential to contain radioactive materials have provisions for isolation from SCWS and drains within the isolation boundary that are directed to the liquid radwaste system.
	The SCWS components are selected for reliable service for the 60-year design life of the plant.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL item

Table 12.3-38: Regulatory Guide 4.21 Design Features for Solid Radioactive Waste System

Objective	Design Features
Objective 1 - Minimize the potential for leaks or spills and provide containment areas	The low activity phase separator tanks and the high activity SRSTs are designed with stainless steel material and welded construction to minimize degradation over the life of the plant due to corrosion.
	The HEPA filter, downstream of the dewatering fill head vent and the Compactor vent, captures radioactive particles before discharging to the RWBVS.
	The high and low activity waste storage areas (Class A and B) are designed with epoxy-coated floors and drainpipes to direct drainage to a floor drain sump for collection and subsequent pumping to the LRWS for treatment and release to the environment.
	Tank cubicles, in which contaminated materials are handled and stored, have lined stainless steel walls and floors to contain the whole tank content if a leak develops. The floors are sloped to direct leakage to a low point floor drain in the tank room for ease of transfer and cleaning. Drain lines direct floor drains to the local sump tanks, which are equipped with level switches to detect liquid accumulation and pumps are provided to transfer the fluid to the LRWS for proper treatment.
	The SRWS is designed with above-ground piping to the extent practical. Buried or embedded piping is minimized. In the event that buried or embedded piping cannot be avoided, double-wall piping is used.
Objective 2 - Provide leak detection capability	Four resin storage tanks are designed with vibrating fork level switches and radar level transmitters to provide reasonable assurance of the integrity of the SSC. Leak detection (level switches) is located in each tank cubicle and provides alarm to warn the operators of leaks.
	Video monitoring is provided in the high and low activity waste storage area for waste handling operation and to monitor for container leakages.
	The fill head is designed with a local control panel with closed-circuit television and level indication to monitor HIC level during the resin transfer and dewatering operations. The closed-circuit television in the dewatering room monitors external leakages associated with HIC overflow or hose or joint failures in the dewatering room.
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	The SSC are designed with life-cycle planning using nuclear, industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation.
	The process piping containing contaminated slurry is sized properly to facilitate easier flow and with sufficient velocity to prevent settling. The piping is designed to reduce fluid traps, thus reducing the decontamination needs and waste generation. Decontamination fluid is collected and routed to the LRWS for processing and release.
	Utility connections are designed with a minimum of two barriers (double isolation valves) to prevent contamination of non-radioactive systems from potentially radioactive systems.
Objective 4 - Facilitate decommissioning	The SSC are designed for the 60-year design life and are fabricated, to the maximum extent practicable, as individual assemblies for removal.
	The SSC are designed with decontamination capabilities. Design features, such as welding techniques and surface finishes, are included to minimize the need for decontamination and minimize waste generation.
	Instruments that interface with contaminated fluid or slurry are designed with diaphragm seals to reduce decontamination requirements during decommissioning.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site Radiological Environmental Monitoring	COL item

Table 12.3-39: Regulatory Guide 4.21 Design Features for Ultimate Heat Sink

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	The UHS pool liner plates and the piping system are designed and fabricated according to industry codes and standards. This feature minimizes the potential for leaks. This design feature also applies to objective 3 for cross contamination.
	The UHS uses stainless steel makeup line material to reduce the potential that corrosion results in leaks. This feature also applies to Objective 3.
Objective 2 - Provide leak detection capability	Pool level instrumentation is provided to monitor the pool level.
	The PLDS is designed to collect and detect pool liner leakage. The PLDS leak channels drain potential pool liner leakage to the RWDS sumps located in the RXB, which are equipped with level transmitters to detect leaks. The PLDS also provides for a manual leak measurement capability for each leak channel to quantify small leaks (i.e., few gallons per week).
Objective 3 - Reduce contamination to minimize releases, cross-contamination and waste generation	Stainless steel is selected as the material for wetted components to provide corrosion resistance and minimize degradation and particulate generation over the 60-year design life of the plant. (also applies to objective 4)
	The water inventory within the UHS is provided with cleanup capability through the PCUS filters and demineralizers to reduce the contamination level of the water.
Objective 4 - Facilitate decommissioning	The UHS components are designed to facilitate decommissioning with stainless steel surfaces for easier decontamination.
Objective 5 - Operating programs and documentation	COL item
Objective 6 - Site radiological environmental monitoring	COL Item

Table 12.3-40: Regulatory Guide 4.21 Design Features for Utility Water System

Objective	Design Features
Objective 1 - Minimize the potential for leaks/ spills and provide containment areas	Piping and component materials in the UWS are consistent with a 60-year design life. Valve material(s) are consistent with system service(s) and design conditions. UWS piping is designed to ASME B31.1. The majority of the UWS is 304 stainless steel for above ground piping and HDPE for underground piping. The SCWS blowdown piping is carbon steel for above ground piping and 316 stainless steel for underground piping. The 316 stainless steel material is used for the underground piping penetrating through building slabs.
Objective 2 - Provide leak detection capability	The UWS is the single-point liquid effluent release path to the environment, and it is sampled and monitored for radiation. An off-line radiation monitor provides continuous indication of effluent parameters.
Objective 3 - Reduce contamination to minimize releases, cross-contamination & waste generation	The above and below ground UWS piping is designed to last for 60 years. Adequate space is provided for maintenance of equipment and other system components in the system layout and design. Additionally, the raw water pumps and supply pumps are provided with redundant pumps to allow for one pump to be taken out of service for maintenance. Each pump has an upstream and downstream isolation valve to enable separation from the system for maintenance, replacement, or both.
Objective 4 - Facilitate decommissioning	The UWS discharge basin may become contaminated during operation and can be decontaminated during decommissioning.
Objective 5 - Operating programs and documentation	COL Item
Objective 6 - Site radiological environmental monitoring	COL Item

Figure 12.3-1a: Reactor Building Radiation Zone Map - 25' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1b: Reactor Building Radiation Zone Map - 40’ Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1c: Reactor Building Radiation Zone Map - 55' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1d: Reactor Building Radiation Zone Map - 70' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1e: Reactor Building Radiation Zone Map - 85' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1f: Reactor Building Radiation Zone Map - 100' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1g: Reactor Building Radiation Zone Map - 126' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1h: Reactor Building Radiation Zone Map - 145' 6" Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-2a: Radioactive Waste Building Radiation Zone Map - 70' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-2b: Radioactive Waste Building Radiation Zone Map - 82' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-2c: Radioactive Waste Building Radiation Zone Map - 100' Elevation

{{ Withheld - See Part 9 }}

12.4 Dose Assessment

The dose assessment information in this section includes the estimated radiation exposures to plant personnel performing work activities involving normal operations, maintenance, inspections, refueling activities, and waste handling, using the methodology presented in Regulatory Guide 8.19 to demonstrate the facility design complies with 10 CFR 20.

The dose assessment process during the design of the plant and the program for keeping doses as low as reasonably achievable (ALARA) help maintain occupational radiation exposures ALARA. To estimate the occupational radiation exposures (OREs) for the facility, various work activities and durations are compiled along with the expected significant (>0.1 mrem/hr) radiation fields that are encountered.

12.4.1 Occupational Radiation Exposure

Regulatory Guide 8.19 methodology is used to determine the OREs to plant personnel. Section 12.3 describes radiation protection design features and Section 12.5 explains elements of the operational radiation protection program. Both the program and design maintain operator occupational radiation exposure ALARA. Section 12.2 provides the airborne concentration in various parts of the facility.

The calculated occupational exposure estimates are developed to reflect the expected dose rates in a facility during its operation. In the absence of operational dose information, the sources in Section 12.2 are used to inform some of the assumed dose rates used for occupational exposures.

The following activity categories are used to calculate the estimated annual ORE estimates:

- reactor operations and surveillances
- routine maintenance
- inservice inspection
- special maintenance
- waste processing
- refueling activities

Table 12.4-1 summarizes the total estimated annual occupational personnel doses.

Major elements contributing to lower occupational doses include low plant radiation fields due to crud reduction efforts and leak minimization, favorable plant arrangement and equipment layout, and operational practices and procedures that minimize time spent in radiation fields. The paragraphs below detail each work activity associated with annual ORE.

12.4.1.1 Reactor Operations and Surveillance

During plant operations, systems and components are monitored for performance and operating condition. Some examples of the specific activities performed by operators include:

- inspection and performance tests of plant components and systems
- checks of unidentified leaks
- operation of manual valves
- reading of instruments
- health physics patrols and surveys
- security sweeps or patrols
- calibration of electrical and mechanical equipment
- chemistry sampling and analysis

Table 12.4-2 provides estimated values of the collective doses for reactor operations and surveillances.

12.4.1.2 Routine Inspection and Maintenance

During plant operations, routine inspection and maintenance activities include various inspections, repairs, and replacements of pumps, valves, heat exchangers, and instrumentation within the Reactor Building and the hot machine shop. Table 12.4-3 provides occupational dose estimates from routine inspection and maintenance.

12.4.1.3 Inservice Inspection

Periodic inservice inspections are required for safety-related equipment by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI. These components include the reactor pressure vessel, containment vessel, core support, and other internal components. These inspections are typically performed during refueling outages.

Table 12.4-4 provides occupational dose estimates from inservice inspection (ISI) activities.

12.4.1.4 Special Maintenance

Special maintenance consists of activities that go beyond routine scheduled maintenance, including the modification of equipment to upgrade the NuScale Power Module (NPM) and repairs to failed components. The maintenance activities included in this evaluation are associated with the upper section of the NPM while staged in the dry dock during refueling outages. Maintenance activities that do not require a shutdown are considered to be routine maintenance activities. Some special maintenance activities for each NPM are assumed to occur once every 1.5 years, based on a 1.5 year refueling cycle. Table 12.4-5 provides the estimated occupational doses from special maintenance operations.

12.4.1.5 Waste Processing

The waste processing occupational dose estimate includes activities involving the processing of liquid, solid, and gaseous radioactive wastes and other activities in the Radioactive Waste Building, including routine inspections and maintenance and operations and surveillance activities. The doses are estimated based on identified activities that are grouped into three categories: waste management, waste maintenance, and other activities.

Waste management activities include the collection, processing, storing, and releasing of radioactive waste. Waste maintenance activities include system equipment inspections and repair, component flushing, and component replacement. Other activities are support activities within the Radioactive Waste Building, including chemistry sampling, instrument calibrations, health physics surveillances, and security patrols.

Table 12.4-6 lists the estimated occupational doses from waste processing.

12.4.1.6 Refueling Activities, Including Dry Dock Outage Activities

When an NPM is shut down for refueling, other NPMs continue operation. Therefore, dose contributions from operating NPMs are included with the dose received from outage activities, as appropriate. In addition, station personnel could be tasked with working on multiple NPM refueling outages within the same year.

The dose assessment for refueling activities includes the following major activities:

- preparing the NPM for movement
- disconnecting and moving the NPM to the containment flange tool
- disassembling the NPM and dry dock activities
- completing the lower containment vessel work
- refueling the reactor
- reassembling and moving the NPM to the operating bay
- reconnecting the NPM
- transitioning the NPM to power operations

Table 12.4-7 provides estimated occupational doses from refueling activities.

12.4.1.7 Overall Plant Doses

Occupational personnel dose estimates are calculated assuming a 6-NPM site and an 18-month fuel cycle for NPM operation, which equates to four refueling outages per year. Table 12.4-1 summarizes the estimated total annual occupational radiation exposures associated with the major activities shown above.

12.4.1.8 Post-Accident Actions

There are no vital areas, as defined by NUREG-0737, Item II.B.2, other than the main control room and the technical support center, which are in compliance with 10 CFR 50.34(f)(2)(vii). Chapter 15 defines post-accident operator actions, and there are no credited post-accident actions outside the main control room. Section 15.0.3 provides operator dose assessments for the main control room and the technical support center.

12.4.1.9 Construction Activities

For the construction of an additional NuScale plant adjacent to an existing NuScale plant, the estimated annual radiation exposure to a construction worker is estimated based upon a construction staffing plan over the estimated construction period. The annual dose for a construction worker is estimated to be 1.64 mrem/year.

COL Item 12.4-1: An applicant that references the NuScale Power Plant US460 standard design will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.

12.4.2 Radiation Exposure at the Restricted Area Boundary

The direct radiation to the restricted area boundary from on-site sources, such as buildings, is negligible.

Table 12.4-1: Estimated Total Annual Occupational Radiation Exposures

Activity Category	Estimated Annual Dose (man-rem)
Reactor operations & surveillance	2.3
Routine maintenance & inspections	3.7
Inservice inspection	3.4
Special maintenance	2.7
Waste processing	1.3
Refueling	1.7
Total	**15.1**

Note: Estimates assume a plant with 6 NPMs on an 18-month refueling cycle.

Table 12.4-2: Occupational Dose Estimates from Reactor Operations and Surveillance

Discipline	Exposure Times (man-hr/year)	Annual ORE (man-rem)
Operations	1.5E+03	6.3E-01
Health physics	1.7E+03	7.8E-01
Security	2.9E+03	2.9E-01
Instrument and controls (I&C)	1.5E+03	6.3E-01
Total	**7.5E+03**	**2.3E+00**

Table 12.4-3: Occupational Dose Estimates from Routine Inspection and Maintenance

Total ORE by Floor			
Floor Elevation (ft)	**Man Hours (hr)**	**Weighted Average Dose Rate (mrem/hr)**	**Total ORE (man-rem)**
25	4.0E-01	3.9E-01	1.6E-04
40	1.2E02	2.0E-01	2.5E-02
55	1.9E02	4.0E00	7.5E-01
70	4.2E01	1.9E-01	8.1E-03
85	2.3E02	2.1E-01	4.9E-02
100	1.1E03	2.6E+00	2.9E+00
Total	**1.7E03**	**2.2E+00**	**3.7E+00**

Note: Estimates assume a plant with 6 NPMs on an 18-month refueling cycle.

Table 12.4-4: Occupational Dose Estimates from Inservice Inspection

Activity	Exposure Time (man-hours/year)	Average Dose Rate (rem/hr)	ORE (man-rem/year)
ISI inside containment	3.2E+02	2.1E-03	6.8E-01
ISI outside containment	2.6E+02	1.5E-03	3.8E-01
Steam generator ISI	1.1E+02	2.1E-02	2.4E+00
Total	**6.9E+02**	**5.0E-03**	**3.5E+00**

Note: Estimates assume a plant with 6 NPMs on an 18-month refueling cycle.

Table 12.4-5: Occupational Dose Estimates from Special Maintenance

ORE by Type	Exposure Time (man-hours/year)	Average Dose Rate (rem/hr)	ORE (man-rem)
NPM prep and clean-up	3.0E+02	3.8E-03	1.1E+00
Leak testing	2.2E+02	2.1E-03	4.6E-01
Control rod drive mechanism replacement	3.4E+01	6.2E-03	2.1E-01
Pressurizer heater inspection and replacement	3.1E+01	4.0E-03	1.2E-01
ICI insertion, retraction, and replacement	4.8E+01	5.4E-03	2.6E-01
Sensor maintenance	4.0E+01	1.1E-02	4.5E-01
SG tube plugging and secondary-side chemical cleaning	2.1E+00	1.9E-02	3.9E-02
Total	**6.7E+02**	**3.9E-03**	**2.7E+00**

Note: Estimates assume a plant with 6 NPMs on an 18-month refueling cycle.

Table 12.4-6: Occupational Dose Estimates from Waste Processing

Activity	Average Dose Rate (mrem/hr)	Exposure Time (man-hr/year)	Estimated Annual Dose (man-rem)
Waste management	4.5E-01	4.5E+02	2.0E-01
Waste maintenance	8.8E-01	2.6E+02	2.3E-01
Other waste activities	4.0E-01	2.2E+03	8.8E-01
Total	**4.5E-01**	**2.9E+03**	**1.3E+00**

Note: Estimates assume a plant with 6 NPMs on an 18-month refueling cycle.

Table 12.4-7: Occupational Dose Estimates from Refueling Activities

Refueling Activity	Exposure Time per year (man-hrs)	Average Dose Rate (rem/hr)	Annual Total ORE from Refueling (man-rem/year)
Cooldown and cleanup	3.2E+01	8.3E-04	2.6E-02
Module disconnect/movement	3.2E+02	2.5E-03	7.9E-01
Disassembly	0.0E+00	0.0E+00	0.0E+00
Lower containment work window	0.0E+00	0.0E+00	0.0E+00
Refuel work window	0.0E+00	0.0E+00	0.0E+00
Reassembly	0.0E+00	0.0E+00	0.0E+00
Reconnection	3.4E+02	2.6E-03	8.6E-01
Heat-up to hot standby	6.4E+01	9.7E-04	6.2E-02
Refueling Total	**7.5E+02**	**2.3E-03**	**1.7E+00**

Note: Estimates assume a plant with 6 NPMs on an 18-month refueling cycle, assuming 4 refueling cycles per year.

12.5 Operational Radiation Protection Program

COL Item 12.5-1: An applicant that references the NuScale Power Plant US460 standard design will describe elements of the operational radiation protection program to ensure that occupational and public radiation exposures are as low as reasonably achievable in accordance with 10 CFR 20.1101.

Section B

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 12, “Radiation Protection” of the Standard Design Approval Application.

Section	Observation	Response
12.4	Table 12.4-7. The dose estimates for refueling activities were already low in the DCA and the doses got even lower in the SDAA, and it is not clear what caused the significant decrease in dose. It states that the dose for disassembly, lower containment work, refuel work, and reassembly are all 0.0 man-rem and that there is no exposure time for these activities. Staff understands that some activities can be performed remotely, but all of these items resulted in some dose in the DCA. It is unclear if there are any design changes resulting in lower expected doses during refueling activities. In addition, it is unclear how there can be no dose for some of these activities. For example, how can work be performed on the lower containment without receiving any expected dose?	Table 12.4-7 states that the dose for disassembly, lower containment work, refuel work, and reassembly are all 0.0 man-rem and that there is no exposure time for these activities. The module disassembly and reassembly steps are now planned to be entirely remote. It is assumed that actual refueling of the core is performed entirely remotely via cranes, cameras, and the fuel handling machine. The design for disassembly of the NPM, lower containment work, refueling and reassembly all support remote operations from the Module Maintenance Center. For the Design Certification Application, the designated location for the remote work was local to the refueling area. The US460 standard plant design now has an enclosed observation area from which the activities in question will be remotely performed in a controlled environment. For this occupational radiation exposure estimate, it is assumed the activities performed from this location will not result in measurable dose to the operating crew. The lower containment vessel remains submerged in the ultimate heat sink and is therefore, not a source of radiation exposure to workers. Inspections and maintenance of this component will be performed remotely with the assistance of cameras.