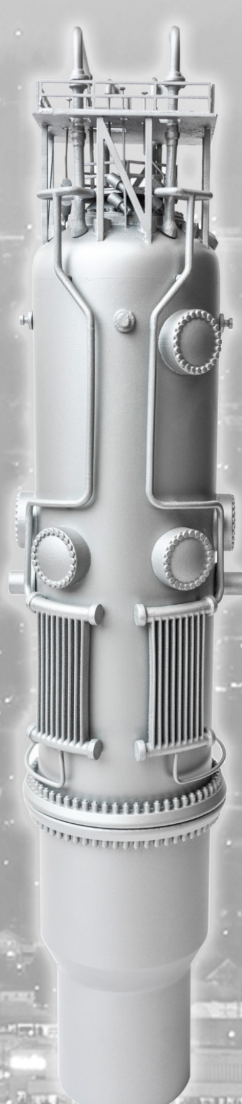
NuScale Standard Plant
Design Certification Application



Chapter Twelve

Radiation Protection

PART 2 - TIER 2

Revision 4
January 2020

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CHAPTER 12 RADIATION PROTECTION

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

The 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 radiation protection policies, programs and features are described in this chapter.

12.1.1 Policy Considerations

12.1.1.1 Design and Construction Policies

The commitment to ALARA is demonstrated by processes that implement ALARA radiation protection design criteria in the plant design. The application of ALARA principles occurs throughout the design process. As the design progresses, application of ALARA-related decisions and design features are reviewed and updated, as appropriate, to reflect operating experience and lessons learned. Design reviews are conducted to 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. This integrated approach to the design of structures, systems, and components ensures that exposures to offsite and onsite personnel are ALARA.

12.1.1.2 Operational Policies

Implementation of the operational ALARA policy is addressed in Section 12.1.3.

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 (RGs) 8.8, Revision 3 and 8.10, Revision 1-R. Plant operations policy considerations described in RG 8.8, RG 8.10, and RG 1.8, Revision 3 will be demonstrated by the COL applicant as discussed in Section 12.1.3 and Section 12.5. The design meets the guidance of RG 8.8, Sections C.2 and C.4 that address facility, equipment, and instrumentation design features. Plant features that are examples of compliance with RG 8.8 are addressed in Section 12.3.

12.1.2 Design Considerations

This section describes the methods and features that allow the policy considerations of Section 12.1.1 to be applied. Design features and attributes for maintaining personnel exposures ALARA are presented in Section 12.3.

12.1.2.1 General Design Considerations for Maintaining Exposures ALARA

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

The ALARA philosophy guiding the design is to ensure that 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

The design considerations implemented to minimize production, distribution, and retention of activated corrosion products includes appropriate material selection and proper water chemistry. These design considerations are described in more detail in Section 12.1.2.2.

Design considerations and features addressing 10 CFR 20.1406 to minimize contamination of the facility and the environment, minimize waste generation, and facilitate decommissioning are described in Section 12.3.

12.1.2.2 Equipment Design Considerations for Maintaining Exposures ALARA

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, with consideration given to potential radiation problems associated with component and materials design. The following paragraphs summarize examples of design considerations utilized in the design to implement ALARA.

12.1.2.2.2 Equipment Design Considerations to Limit Time Spent in Radiation Areas

Equipment is designed with instrumentation and controls in accessible areas during normal and abnormal operating conditions. Equipment is selected to minimize the potential dose to personnel during maintenance. The equipment is provided with drainage capabilities to facilitate maintenance and smooth surfaces to reduce contamination and ease decontamination efforts. Components are chosen for high reliability, ease of maintenance, ease of replacement, accessibility, and ease of decontamination. 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

Site-specific information describing how the plant implements the design consideration guidance provided in RG 8.8 is provided in Section 12.1.3.

12.1.2.2.3 Equipment Design Considerations to Limit Component Radiation Levels

The materials selected for components were chosen to meet service requirements and minimize cobalt-containing materials (e.g., Stellite) coming in contact with the primary coolant system. The cobalt and nickel content and corrosion resistance of a given material influence the production of corrosion products that can become activated. 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. This material selection results in significantly less production of activated corrosion products.

Nickel-based alloys are used in pressurized water reactors due to their excellent corrosion resistance in both reducing and oxidizing environments. Alloy 690 was chosen for the steam generator tubes based on its resistance to primary water stress corrosion cracking and its good heat transfer properties.

The cobalt content of alloy materials is controlled to minimize the radiation exposures resulting from the activation of wear or corrosion products.

Forged low-alloy steel ASME SA-508 is selected for the reactor pressure vessel (RPV) that surrounds the reactor core, pressurizer, and steam generators. For the lower RPV that is subjected to high levels of radiation, copper and phosphorous contents are limited because these elements are primarily responsible for the hardening and embrittlement caused by neutron irradiation.

Equipment is designed with provisions to limit leaks and the plant is designed to collect and control leaked fluid through the use of 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 such as tanks and piping. Piping design avoids the creation of stagnant legs, uses sloping pipe runs, and locates connections above the pipe centerline. Crud deposition within 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 of system components with demineralized water, as needed.

12.1.2.2.4 Water Chemistry

Reactor water chemistry is controlled during operation to minimize corrosion of surfaces in contact with the coolant and minimize the production of activated corrosion products. For reactivity control, boric acid is added as a soluble neutron poison. To maintain the reactor coolant at a slightly alkaline pH, enriched lithium-7 hydroxide is added to the coolant.

To maintain a reducing environment in the reactor coolant, dissolved hydrogen is added, thus minimizing oxidation and suppressing radiolytic oxygen generation during operation. During startup, oxygen removal is accomplished by a combination of mechanical degasification and chemical degasification using hydrazine. Zinc may also be added to reduce corrosion product transport. Impurities and suspended solids are removed from the reactor coolant by the chemical and volume control system.

12.1.2.3 Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA

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 Annex Building, and include change rooms, offices, calibration facilities, counting room, hot machine shop, and equipment and personnel decontamination facilities. The Annex 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 utilized in minimizing personnel time spent in radiation areas include:

- space is provided within cubicles for a laydown area for special tools and ease of servicing activities
- instrumentation readouts, monitors, and control points are 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 is provided between components
- labyrinth entrances are provided to reduce radiation streaming out of cubicle entrances
- shield wall penetrations are configured to prevent "line-of-sight" streaming
- pipe chases are used for pipes containing significant radioactive material
- radiation areas where station personnel spend substantial time are designed to the lowest practical dose rates
- curbing and sloped floors direct leakage to local drains or sumps to limit the spread of contamination from liquid systems
- tanks containing radioactive liquids are designed with sloped bottoms toward outlets and flushing or cleaning features
- spare connections on tanks and other components located in high radiation areas are provided to allow for greater operational flexibility
- pumps are selected to minimize leakage and provide housing drains
- radiation sources are separated from occupied areas where practicable (e.g., pipes or ducts containing high radioactive fluids not passing through occupied areas)
- when permanent shielding is impractical, provisions for temporary shielding are provided
- instrumentation is designed, selected and located with consideration for long service life, ease and low frequency of calibration, and low crud accumulation
- provisions to permit the rapid manipulation of shielding and insulation for equipment that requires periodic inspection or service are included
- adequate space for moveable or temporary shielding for sources is provided
- means to control contamination and to facilitate decontamination of potentially contaminated areas are provided
- piping for "clean services" (e.g., station air, potable water, nitrogen, etc.) is located separate from piping for contaminated systems to avoid cross-contamination are provided
- features that permit remote removal, installation, inspection, or servicing of radioactive components are provided
- design features such as ventilation isolation or filtration and heating ventilation and air conditioning design such that air flows from areas of lower radioactivity to areas of higher radioactivity to protect against airborne contamination are provided

12.1.3 Operational Considerations

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

12.2 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 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 and are based on the design basis primary coolant activity concentrations from Section 11.1. 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 code is used to bin the particle emissions into default energy bins based on the activity of each individual isotope. The radiation sources described in this section provide part of the basis for the design of radiation shielding features. Plan scale drawings showing locations of contained sources are included in the radiation zone maps (Section 12.3).

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. The fission neutron and fission gamma source strength and neutron energy spectrum information are provided in Table 12.2-1. The n-gamma source strength is internally generated by MCNP6 using the neutron source strength as an input. The fission gamma energy probability density function is provided in Table 12.2-2. 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 design basis source terms are described in Section 11.1.

The contribution of gamma radiation from the primary coolant is comprised of two components: the hot leg (lower riser, and upper riser) and a cold leg (pressurizer, steam generator, and reactor coolant downcomer region). The hot leg is modeled with the peak N-16 concentration in the lower riser, while the cold leg is modeled with an N-16 concentration equal to the entrance of the steam generator. Because of the low flow velocity of the primary coolant, the short half life of N-16 causes it to decay by about one order of magnitude by the time it reaches the steam generator entrance. By uniformly treating the hot leg with the peak N-16 concentration in the RCS loop, and the cold leg uniformly with the steam generator entrance concentration, the gamma contribution from N-16 is conservatively modeled. The fission isotopes and corrosion isotopes (CRUD) are uniformly modeled on a primary coolant mass basis. The primary coolant gamma spectra are provided in Table 12.2-3 and Table 12.2-4.

Nitrogen-16 is present throughout the primary coolant loop, and the modeling simplification described above is conservative from a bioshield radiation shielding perspective. Table 12.2-5 tabulates the nitrogen-16 concentration at several locations in the primary coolant system.

12.2.1.3 Chemical and Volume Control System

The chemical and volume control system (CVCS) takes a portion of the RCS and processes the water through heat exchangers, demineralizers, and filters. The treated primary coolant water 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 assumed to be 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 assumed to be in operation for one-half of the fuel cycle because they are operated intermittently during the operating cycle for lithium removal. The decontamination factors assumed are listed in Table 12.2-6.

The CVCS demineralizer beds are located in the Reactor Building (RXB) on the 24' elevation inside the CVCS cubicles. The mixed-bed source terms and source strengths are listed in Table 12.2-7 and Table 12.2-8, respectively. These source terms and the associated analyses do not include short-term transients such as CRUD bursts associated with refueling outages. Based on an assumed Co-58 peaking factor of 10,000 and an assumed peaking factor of 1,000 for other CRUD isotopes, it is estimated that a CRUD burst could add up to 450 curies of CRUD isotopes to the CVCS mixed bed demineralizer. This results in the estimates of activity within some plant SSCs to not reflect the CRUD burst related activity, including the CVCS mixed bed demineralizer values (both columns) in Table 12.2-7 and Table 12.2-8.

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

The regenerative heat exchanger is used to cool the primary coolant as it enters the CVCS using the CVCS water returning back to the RCS. The non-regenerative heat exchanger further cools the primary coolant, using reactor component cooling water, to protect the demineralizer resins. The heat exchangers are tube and shell 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 12.2-6. The source term for the RCS water is found in Table 11.1-4.

The heat exchangers are located in the RXB on the 50' elevation inside the heat exchanger rooms.

Module Heating System Heat Exchangers

The module heating system heat exchangers are modeled with the tube side filled with design basis primary coolant. No credit was given for shielding provided by the tubes or the clean steam and water filling the shell side.

Resin Transfer Pipe

A generic resin transfer line is modeled assuming it is 100 percent obstructed by spherical resin beads from the CVCS mixed bed demineralizer, which has been modeled using a bulk dry resin density. The generic resin transfer line is modeled with the parameters listed in Table 12.2-6. The source term used for the spent resin transfer line is the CVCS mixed bed demineralizer decayed for 48 hours, as provided in Table 12.2-7 and Table 12.2-8, with a spent resin volume of 8.8 ft^3.

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 assumed to remove crud particulate. The assumed filter efficiency is listed in Table 12.2-6. The filter source term and source strengths are listed in Table 12.2-7 and Table 12.2-8, respectively.

12.2.1.4 Reactor Pool Cooling, Spent Fuel Pool Cooling and Pool Cleanup Systems

The reactor pool cooling system (RPCS) is a cooling-water system that removes heat from the reactor pool, while the spent fuel pool cooling system (SFPCS) removes heat by drawing water from the spent fuel pool. The pool cleanup system (PCUS) draws water from either the SFPCS or the RPCS and removes impurities to reduce radiation exposures and to maintain water chemistry and clarity. These systems are further described in Section 9.1.3.

The RPCS and SFPCS 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 primary system components considered in designing shielding are the PCUS demineralizers and filters that accumulate activity from radioactive contamination in the reactor pool water. The PCUS demineralizers are assumed to collect the entire inventory of radioactivity in the pool water as reflected in Table 12.2-10. It is also assumed that the PCUS demineralizers operate for two years, resulting in the collection of the entire reactor pool water radionuclide inventory 12 times (assuming a plant with 12 NPMs on a 2 year refueling cycle).

The PCUS filters are assumed to collect CRUD particulates and are operated for one year.

The input assumptions used to develop these source terms are listed in Table 12.2-9. The radionuclide source terms and source strengths for this equipment are provided in Table 12.2-10 and Table 12.2-11, respectively.

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. The radionuclide inventories are listed for the major LRWS components in Table 12.2-13a and Table 12.2-13b which are the basis for the liquid radioactive waste component shielding design.

The estimated input flows from various sources to the high-conductivity waste (HCW) collection tanks, and the low-conductivity waste (LCW) collection tanks are listed in Table 11.2-3. 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. The assumed values for the LRW processing equipment radionuclide collection efficiencies are listed in Table 12.2-12. The LRWS component source terms are provided in Table 12.2-13a and Table 12.2-13b, and source strengths are provided in Table 12.2-14a and Table 12.2-14b. To establish the shielding design downstream of the GAC filter, the radionuclide concentration in the outlet stream from the GAC filter is assumed to not be reduced by the GAC filter.

12.2.1.6 Gaseous Radioactive Waste System

Radioactive fission gases are produced in the reactor core and assumed to be released to the primary coolant, as discussed in Section 11.1. 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 12.2-15 lists the assumed values pertaining to the GRWS source geometries and Table 11.3-1 describes the GRWS processing parameters. The GRWS component source terms are provided in Table 12.2-16 and the source strengths are provided in Table 12.2-17.

The radioisotopic inventory listed in Table 12.2-16 for the GRWS guard bed and decay beds results in a RG 1.143 safety classification of RW-IIb. Because an end of operating cycle degasification evolution could result in a transient radioisotopic inventory that exceeds that listed in Table 12.2-16, the RG 1.143 safety classifications for the GRWS guard bed and decay beds are increased from RW-IIb to RW-IIa to cover such transients, as reflected in Table 11.3-2.

12.2.1.7 Solid Radioactive Waste System

The solid radioactive waste system (SRWS) handles solid radioactive waste from various waste streams, as described in Section 11.4. The waste inputs to the SRWS components are collected, resulting in a radionuclide source term for the SRWS components. The assumed values used to develop the SRWS component source terms are listed in Table 12.2-18. Table 12.2-19 lists the radionuclide inventory of the major SRWS components and Table 12.2-20 lists the SRWS component source strengths. As described in Section 11.4, there is storage space provided in the Radioactive Waste Building for processed waste packages that contain spent filters, dewatered resins, and other solid wastes. For shielding design purposes, it is assumed that the Class A/B/C high integrity container storage area contains five high integrity containers loaded with Class B/C dewatered spent resins from the spent resin storage tank, which has been decayed for approximately two years (one fuel cycle), and one 55-gallon drum filled with waste from the LRWS drum dryer. Table 12.2-13b provides the radionuclide inventory of the drum dryer and Table 12.2-14b 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

The reactor pool is housed within the RXB and contains up to 12 NPMs, which are partially immersed in the reactor pool water. Because the spent fuel pool communicates with the reactor pool through the weir wall, radionuclides are mixed with the spent fuel pool water volume. 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 small 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 will operate until the projected dose rate (after NPM disassembly) to an operator on the refueling bridge is less than 2.5 mR/hr. The other major input assumptions for the pool water source term are provided in Table 12.2-9.

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 quickly decrease in the reactor pool's borated water. The small amount of neutron activation products in the reactor pool water was calculated to be insignificant compared to the amount of primary coolant radionuclides released to the reactor pool water during refueling outages. The reactor pool and RCS water chemistry limits (when the temperature of the RCS is less than 250 degrees F) are in conformance with the Electric Power Research Institute primary water chemistry guidelines (Reference 12.2-3). The reactor pool water volume dilutes inadvertently introduced impurities that could result from component failures and, because the chemistry limits in both the reactor pool and each NPM are monitored, impurities in either of the two water sources are minimized.

Between refueling outages, the radionuclides in the reactor pool water are treated by the PCUS demineralizers and filters to reduce the radionuclide content. The pool water has a negligible neutron activation source term. The major input assumptions are listed in Table 12.2-9.

The pool surge control system (PSCS) storage tank is designed to temporarily store cleaned up pool water that is displaced during drydock operations. The PSCS storage tank is modeled as a vertical cylindrical tank with the characteristics listed in Table 12.2-9.

The source terms and the source strengths for the pool water and the PSCS storage tank are provided in Table 12.2-10 and Table 12.2-11, respectively.

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's concrete 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 freshly-discharged, 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 12.2-21 for a spent fuel rack full of freshly discharged fuel assemblies. Spent fuel neutron source strengths are given in Table 12.2-22 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. The major inputs assumptions are listed in Table 12.2-23. The gamma spectra are provided in Table 12.2-24.

12.2.1.11 Control Rods and Secondary Source Rods

<u>Control Rod Assemblies</u>

The control rod assemblies are irradiated during reactor operations. 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. The major input assumptions are listed in Table 12.2-25. The CRA gamma spectra are listed in Table 12.2-26.

<u>Secondary Source Rod</u>

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

The gamma ray source strengths associated with the secondary source rods are listed in Table 12.2-27 for various times after shutdown.

12.2.1.12 Secondary Coolant System

The secondary coolant system is expected to contain minimal radioactivity during normal operations. Primary-to-secondary leaks through the steam generator can introduce primary coolant activity into the secondary system with the resultant contamination level being dependent upon 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 were 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 ten day resin regeneration period, the accumulation of radioactive material is less than 100 mCi.

12.2.1.13 Post-Accident Sources

The iodine spike design basis source term (the maximum primary coolant activity released from design basis accidents described in Section 15.0.3) is evaluated for equipment qualification (EQ) in and around an NPM. Three volumes associated with the NPM are evaluated for EQ dose consequences: 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 equipment qualification evaluation are provided on a mass basis in Table 12.2-34. The specific concentration values in Table 12.2-34 are in excess of the values that would be calculated using the methodology in TR-0915-17565, Accident Source Term Methodology, Rev. 3, with the design inputs provided in this FSAR. 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 12.2-28. The three volumes are evaluated with conservative assumptions, including instantaneous and homogeneous releases into the volume of interest.

Table 12.2-31 lists the integrated post-accident source energy deposition versus time for both photons and electrons for the three evaluated volumes. Table 12.2-31 also tabulates the integrated doses for various times post-accident. For additional details on equipment qualification, see Section 3.11 and Appendix 3.C. Consistent with 10 CFR 50.34(f)(2)(vii), areas that could contain core damage post-accident sources were evaluated for equipment protection. Information on equipment protection from a core damage source term is addressed in Section 19.2.

12.2.1.14 Other Contained Sources

There are no other identified contained sources that exceed 100 mCi, including HVAC filters. To evaluate the accumulation of radioactive material on the Reactor Building HVAC system HEPA filters, the airborne radioactivity in the Reactor Building due to pool evaporation and primary coolant leaks was deposited on filters assuming a 99 percent particulate efficiency and two years of operation. For the pool evaporation portion, the Reactor Building HVAC system provides a ventilation flow rate equivalent to one air volume change per hour. For the primary coolant leakage portion, the activity that becomes airborne is captured and filtered by the ventilation system. The resultant accumulation of radioactive material is less than 100 mCi.

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

12.2.2 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 RXB atmosphere due to reactor pool evaporation or primary coolant leakage. The airborne concentration is modeled as a buildup to an equilibrium concentration based on Bevelacqua (Reference 12.2-1) given the production rate and removal rate, and on an evaporation rate based on ASHRAE (Reference 12.2-6). 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 reactor coolant system. Each case maximizes the tritium concentration in the fluid of interest. These values are reported 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. The input parameters are listed in Table 12.2-32.

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

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.

The resultant airborne isotopic concentrations in the RXB atmosphere are listed in Table 12.2-33. Monitoring airborne radioactivity within the air spaces of the facility is described in Section 12.3.4. Monitoring gaseous effluents is described in Section 11.5.

12.2.2.2 Turbine Building Atmosphere

As discussed in Section 12.2.1.12, the secondary coolant is considered to be clean for normal operating conditions. Therefore, the Turbine Building atmosphere contains minimal airborne radioactive material.

12.2.3 References

12.2-1 Bevelacqua, J.J., *Basic Health Physics, Problems and Solutions*, Wiley-VCH Publishing, Weinheim, Germany, 2004.

12.2-2 International Atomic Energy Agency, "Combined Methods for Liquid Radioactive Waste Treatment" IAEA-TECDOC-1336, Vienna, Austria, February 2003.

12.2-3 Electric Power Research Institute, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Vols. 1 and 2, EPRI #3002000505, Rev. 7, Palo Alto, CA, 2014.

12.2-4 Not Used.

12.2-5 Not Used.

12.2-6 American Society of Heating, Refrigerating and Air-Conditioning Engineers, 2007 ASHRAE Handbook Applications, Atlanta, GA.

Table 12.2-1: Core and Coolant Source Information

Parameter	Value
Fission neutron source strength	1.91E+19 particles/sec
Fission neutron energy spectrum	Watt spectrum for U-235
Fission gamma source strength	1.45E+20 particles/sec
Hot Leg Fraction of Primary coolant	27%
Cold Leg Fraction of Primary coolant	73%
Total Primary Coolant source strength	1.06E+14 particles/sec

Table 12.2-2: Fission Gamma Energy Spectrum Probability Density Function

Energy Bin Boundaries (eV)			Probability Density Function
0.00E+00	-	5.00E+04	1.417E-07
5.00E+04	-	1.00E+05	2.367E-07
1.00E+05	-	1.50E+05	3.954E-07
1.50E+05	-	1.63E+05	6.676E-07
1.63E+05	-	1.78E+05	7.101E-07
1.78E+05	-	2.00E+05	6.123E-07
2.00E+05	-	2.12E+05	8.632E-07
2.12E+05	-	2.25E+05	1.106E-06
2.25E+05	-	2.45E+05	8.433E-07
2.45E+05	-	2.65E+05	6.094E-07
2.65E+05	-	3.00E+05	4.110E-07
3.00E+05	-	3.25E+05	6.846E-07
3.25E+05	-	3.50E+05	8.178E-07
3.50E+05	-	3.65E+05	1.030E-06
3.65E+05	-	4.00E+05	1.066E-06
4.00E+05	-	4.30E+05	9.396E-07
4.30E+05	-	4.50E+05	8.418E-07
4.50E+05	-	4.85E+05	8.844E-07
4.85E+05	-	5.00E+05	9.964E-07
5.00E+05	-	5.25E+05	9.596E-07
5.25E+05	-	5.45E+05	8.249E-07
5.45E+05	-	5.75E+05	7.327E-07
5.75E+05	-	6.05E+05	8.759E-07
6.05E+05	-	6.25E+05	9.538E-07
6.25E+05	-	6.50E+05	8.632E-07
6.50E+05	-	6.70E+05	7.285E-07
6.70E+05	-	6.83E+05	7.753E-07
6.83E+05	-	7.00E+05	7.994E-07
7.00E+05	-	7.50E+05	7.186E-07
7.50E+05	-	8.00E+05	6.973E-07
8.00E+05	-	9.00E+05	6.633E-07
9.00E+05	-	1.00E+06	5.102E-07
1.00E+06	-	1.10E+06	4.209E-07
1.10E+06	-	1.20E+06	3.586E-07
1.20E+06	-	1.25E+06	3.415E-07
1.25E+06	-	1.35E+06	3.274E-07
1.35E+06	-	1.50E+06	2.821E-07
1.50E+06	-	1.75E+06	2.013E-07
1.75E+06	-	2.00E+06	1.289E-07
2.00E+06	-	2.25E+06	9.349E-08
2.25E+06	-	2.50E+06	7.367E-08
2.50E+06	-	2.75E+06	5.866E-08
2.75E+06	-	3.00E+06	4.504E-08
3.00E+06	-	3.25E+06	3.433E-08
3.25E+06	-	3.50E+06	2.482E-08

Table 12.2-2: Fission Gamma Energy Spectrum Probability Density Function (Continued)

Energy Bin Boundaries (eV)			Probability Density Function
3.50E+06	-	3.75E+06	1.972E-08
3.75E+06	-	4.00E+06	1.562E-08
4.00E+06	-	4.25E+06	1.271E-08
4.25E+06	-	4.50E+06	9.349E-09
4.50E+06	-	4.75E+06	6.577E-09
4.75E+06	-	5.00E+06	5.075E-09
5.00E+06	-	5.25E+06	4.214E-09
5.25E+06	-	5.50E+06	3.554E-09
5.50E+06	-	5.75E+06	3.023E-09
5.75E+06	-	6.00E+06	2.583E-09
6.00E+06	-	6.25E+06	2.172E-09
6.25E+06	-	6.50E+06	1.762E-09
6.50E+06	-	6.75E+06	1.421E-09
6.75E+06	-	7.00E+06	1.101E-09
7.00E+06	-	7.50E+06	8.609E-10
7.50E+06	-	7.75E+06	4.204E-10
7.75E+06	-	8.00E+06	1.201E-10
8.00E+06	-	8.10E+06	1.001E-11

Table 12.2-3: Cold Leg Primary Coolant Gamma Source Term

Energy Group	Lower Bound (MeV)	Upper Bound (MeV)	Gamma Source (photons/sec/gram primary coolant)
1	1.00E-02	2.00E-02	3.38E+04
2	2.00E-02	3.00E-02	2.50E+04
3	3.00E-02	4.50E-02	1.03E+05
4	4.50E-02	6.00E-02	1.26E+04
5	6.00E-02	7.00E-02	6.19E+03
6	7.00E-02	7.50E-02	2.69E+03
7	7.50E-02	1.00E-01	8.49E+04
8	1.00E-01	1.50E-01	1.32E+04
9	1.50E-01	2.00E-01	1.09E+04
10	2.00E-01	2.60E-01	1.30E+04
11	2.60E-01	3.00E-01	9.78E+03
12	3.00E-01	4.00E-01	9.74E+04
13	4.00E-01	4.50E-01	3.43E+03
14	4.50E-01	5.10E-01	3.59E+03
15	5.10E-01	5.12E-01	1.37E+03
16	5.12E-01	6.00E-01	4.21E+04
17	6.00E-01	7.00E-01	1.33E+04
18	7.00E-01	8.00E-01	6.31E+03
19	8.00E-01	9.00E-01	5.59E+03
20	9.00E-01	1.00E+00	1.61E+03
21	1.00E+00	1.20E+00	3.06E+03
22	1.20E+00	1.33E+00	1.11E+04
23	1.33E+00	1.44E+00	1.46E+03
24	1.44E+00	1.50E+00	3.57E+02
25	1.50E+00	1.57E+00	4.26E+02
26	1.57E+00	1.66E+00	4.35E+02
27	1.66E+00	1.80E+00	1.50E+03
28	1.80E+00	2.00E+00	1.28E+03
29	2.00E+00	2.15E+00	4.89E+02
30	2.15E+00	2.35E+00	3.02E+02
31	2.35E+00	2.50E+00	1.16E+03
32	2.50E+00	2.75E+00	5.03E+03
33	2.75E+00	3.00E+00	1.34E+03
34	3.00E+00	3.50E+00	3.30E+02
35	3.50E+00	4.00E+00	3.71E+02
36	4.00E+00	4.50E+00	1.34E+02
37	4.50E+00	5.00E+00	1.25E+02
38	5.00E+00	5.50E+00	1.11E+02
39	5.50E+00	6.00E+00	5.49E+01
40	6.00E+00	6.50E+00	3.62E+05
41	6.50E+00	7.00E+00	2.37E+02
42	7.00E+00	7.50E+00	2.65E+04
43	7.50E+00	8.00E+00	7.58E+00
44	8.00E+00	1.00E+01	4.23E+02

Table 12.2-3: Cold Leg Primary Coolant Gamma Source Term (Continued)

Energy Group	Lower Bound (MeV)	Upper Bound (MeV)	Gamma Source (photons/sec/gram primary coolant)
45	1.00E+01	1.20E+01	1.08E-01
46	1.20E+01	1.40E+01	-
47	1.40E+01	2.00E+01	-
		Total	9.08E+05

Note: This source term is used for 73 percent by mass of the primary coolant in the NuScale operating reactor shielding calculation, for the pressurizer region above the pressurizer plate and from the steam generator to the cold inlet of the core.

Table 12.2-4: Hot Leg Primary Coolant Gamma Source Term

Energy Group	Lower Bound (MeV)	Upper Bound (MeV)	Gamma Source (photons/sec/gram primary coolant)
1	1.00E-02	2.00E-02	2.95E+05
2	2.00E-02	3.00E-02	1.77E+05
3	3.00E-02	4.50E-02	2.28E+05
4	4.50E-02	6.00E-02	1.12E+05
5	6.00E-02	7.00E-02	5.52E+04
6	7.00E-02	7.50E-02	2.41E+04
7	7.50E-02	1.00E-01	1.75E+05
8	1.00E-01	1.50E-01	1.14E+05
9	1.50E-01	2.00E-01	8.99E+04
10	2.00E-01	2.60E-01	6.06E+04
11	2.60E-01	3.00E-01	3.33E+04
12	3.00E-01	4.00E-01	1.61E+05
13	4.00E-01	4.50E-01	2.59E+04
14	4.50E-01	5.10E-01	2.80E+04
15	5.10E-01	5.12E-01	1.37E+03
16	5.12E-01	6.00E-01	5.44E+04
17	6.00E-01	7.00E-01	3.32E+04
18	7.00E-01	8.00E-01	2.17E+04
19	8.00E-01	9.00E-01	1.78E+04
20	9.00E-01	1.00E+00	1.17E+04
21	1.00E+00	1.20E+00	1.80E+04
22	1.20E+00	1.33E+00	1.95E+04
23	1.33E+00	1.44E+00	6.06E+03
24	1.44E+00	1.50E+00	2.36E+03
25	1.50E+00	1.57E+00	2.35E+03
26	1.57E+00	1.66E+00	3.82E+03
27	1.66E+00	1.80E+00	1.14E+04
28	1.80E+00	2.00E+00	9.30E+03
29	2.00E+00	2.15E+00	2.94E+03
30	2.15E+00	2.35E+00	3.02E+02
31	2.35E+00	2.50E+00	7.19E+03
32	2.50E+00	2.75E+00	4.43E+04
33	2.75E+00	3.00E+00	1.11E+04
34	3.00E+00	3.50E+00	2.97E+03
35	3.50E+00	4.00E+00	3.42E+03
36	4.00E+00	4.50E+00	1.24E+03
37	4.50E+00	5.00E+00	1.15E+03
38	5.00E+00	5.50E+00	1.04E+03
39	5.50E+00	6.00E+00	5.12E+02
40	6.00E+00	6.50E+00	3.38E+06
41	6.50E+00	7.00E+00	2.21E+03
42	7.00E+00	7.50E+00	2.47E+05
43	7.50E+00	8.00E+00	7.07E+01
44	8.00E+00	1.00E+01	3.94E+03

Table 12.2-4: Hot Leg Primary Coolant Gamma Source Term (Continued)

Energy Group	Lower Bound (MeV)	Upper Bound (MeV)	Gamma Source (photons/sec/gram primary coolant)
45	1.00E+01	1.20E+01	1.01E+00
46	1.20E+01	1.40E+01	-
47	1.40E+01	2.00E+01	-
		Total	5.50E+06

Note: This source term is used for 27 percent by mass of the primary coolant in the NuScale operating reactor shielding calculation, for primary coolant leaving the core to the top of the upper riser.

Table 12.2-5: Nitrogen-16 Primary Coolant Concentrations at Full Power

Primary Coolant Location	N-16 concentration (µCi/gram)
Core exit	139
Top of upper riser / entrance to SG	14.9
CVCS letdown line	1.80

Table 12.2-6: Chemical and Volume Control System Component Source Term Inputs and Assumptions

Model Parameter	Value
CVCS mixed bed:	
CVCS mixed bed operation time	100% of fuel cycle
Decontamination Factors	Table 11.1-2
Geometry	vertical cylinder
Source dimensions of vessel	diameter=24″; height=96″
Shielding thickness of steel shell	1.5″
Volume of resin	8.8 ft^3
CVCS Cation bed:	
CVCS Cation Bed Operation Time	50% of fuel cycle
CVCS Cation Bed Decontamination Factors:	
Halogens	1
Cs, Rb	10
Others	10
Geometry	vertical cylinder
Source dimensions of vessel	diameter=24″; height=96″
Shielding thickness of steel shell	1.5″
Regenerative and Non-Regenerative Heat Exchangers:	
Contents	100% primary coolant (see Table 11.1-4)
Geometry	vertical stack of 5 horizontal cylinders
Source dimensions of each cylinder	diameter=12″; length=11.5′
Shielding thickness of steel shell	
Regenerative heat exchanger	1"
Non-Regenerative heat exchanger	0.406"
Module Heating System Heat Exchangers	
Contents	100% primary coolant (see Table 11.1-4)
Geometry	horizontal cylinder
Source dimensions	diameter=21.5″; length=13.1″
Shielding thickness of steel shell	1.25″
CVCS flowrate	22 gpm
CVCS filter efficiency	9.1% (DF = 1.1)
Geometry	vertical cylinder
Dimensions	diameter=13″; height=48″
Shielding thickness of steel shell	0.5″
CVCS resin transfer line:	
Pipe internal diameter	2"
Pipe wall thickness	0.154"
Pipe length	20′
Pipe material	Stainless steel
Resin source term	CVCS mixed bed resin (48-hr decay)
CVCS pipe inside vertical pipe chase	
Contents	Primary coolant (see Table 12.2-8)
Geometry	vertical cylinder
Source length of pipe	20′
Shielding dimensions of pipe	inside diameter=2.5″; thickness=0.375″

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

Isotope	CVCS Mixed Bed Ci	CVCS Cation Bed Ci	CVCS Particulate Filter Ci	CVCS Mixed Bed Transfer - 48-hour decay Ci
Br82	3.40E-02	-	-	1.33E-02
Br83	1.32E-02	-	-	1.26E-08
Br84	1.36E-03	-	-	7.42E-31
Br85	1.50E-05	-	-	-
I129	2.82E-04	8.69E-11	-	2.82E-04
I130	9.62E-02	-	-	6.52E-03
I131	3.74E+01	5.99E-04	-	3.15E+01
I132	1.76E+00	1.43E-02	-	1.04E+00
I133	4.62E+00	1.20E-05	-	9.33E-01
I134	4.87E-02	1.57E-05	-	1.70E-18
I135	1.25E+00	-	-	7.88E-03
Rb86m	1.97E-09	8.85E-10	-	-
Rb86	3.08E-01	1.39E-01	-	2.86E-01
Rb88	3.48E-02	1.57E-02	-	6.03E-51
Rb89	1.36E-03	6.12E-04	-	8.10E-61
Cs132	2.06E-03	9.29E-04	-	1.67E-03
Cs134	1.03E+03	2.70E+02	-	1.03E+03
Cs135m	7.95E-05	3.58E-05	-	3.49E-21
Cs136	7.96E+00	3.58E+00	-	7.16E+00
Cs137	8.43E+02	1.92E+02	-	8.43E+02
Cs138	2.40E-02	1.08E-02	-	2.70E-28
P32	1.32E-06	1.21E-08	-	1.20E-06
Co57	1.57E-07	1.03E-09	-	1.56E-07
Sr89	2.10E-01	2.51E-03	-	2.04E-01
Sr90	4.47E-01	2.08E-03	-	4.47E-01
Sr91	8.61E-04	7.90E-06	-	2.72E-05
Sr92	1.27E-04	1.17E-06	-	4.70E-10
Y90	4.45E-01	2.06E-03	-	4.46E-01
Y91m	5.41E-04	4.97E-06	-	1.73E-05
Y91	3.55E-02	3.21E-04	-	3.46E-02
Y92	2.71E-04	2.49E-06	-	5.28E-08
Y93	1.94E-04	1.78E-06	-	7.39E-06
Zr97	4.70E-04	4.32E-06	-	6.44E-05
Nb95	7.27E-01	6.39E-03	-	7.25E-01
Mo99	3.32E+00	3.05E-02	-	2.01E+00
Mo101	4.63E-04	4.25E-06	-	2.11E-63
Tc99m	3.20E+00	2.94E-02	-	1.94E+00
Tc99	1.71E-02	7.84E-05	-	1.71E-02
Ru103	4.54E-02	4.16E-04	-	4.38E-02
Ru105	7.03E-05	6.46E-07	-	3.91E-08
Ru106	2.05E-01	1.24E-03	-	2.05E-01
Rh103m	4.49E-02	4.12E-04	-	4.33E-02

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

Isotope	CVCS Mixed Bed Ci	CVCS Cation Bed Ci	CVCS Particulate Filter Ci	CVCS Mixed Bed Transfer - 48-hour decay Ci
Rh105	1.26E-03	1.16E-05	-	4.96E-04
Rh106	2.05E-01	1.24E-03	-	2.05E-01
Ag110	1.07E-01	7.14E-04	-	1.06E-01
Sb124	1.03E-04	9.27E-07	-	1.00E-04
Sb125	5.86E-03	3.02E-05	-	5.85E-03
Sb127	2.51E-04	2.30E-06	-	1.75E-04
Sb129	1.47E-05	1.35E-07	-	7.63E-09
Te125m	1.28E-01	1.15E-03	-	1.25E-01
Te127m	7.66E-01	6.37E-03	-	7.56E-01
Te127	7.58E-01	6.32E-03	-	7.41E-01
Te129m	6.84E-01	6.28E-03	-	6.56E-01
Te129	4.32E-01	3.97E-03	-	4.14E-01
Te131m	8.32E-02	7.64E-04	-	2.74E-02
Te131	1.90E-02	1.75E-04	-	6.18E-03
Te132	1.55E+00	1.43E-02	-	1.01E+00
Te133m	1.59E-03	1.46E-05	-	3.56E-19
Te134	1.71E-03	1.57E-05	-	3.10E-24
Ba137m	7.96E+02	1.81E+02	-	7.96E+02
Ba139	6.31E-05	5.79E-07	-	2.30E-15
Ba140	7.60E-02	6.98E-04	-	6.82E-02
La140	7.90E-02	7.25E-04	-	7.48E-02
La141	5.54E-05	5.09E-07	-	1.14E-08
La142	1.03E-05	9.41E-08	-	3.12E-15
Ce141	2.99E-02	2.75E-04	-	2.87E-02
Ce143	9.57E-04	8.79E-06	-	3.50E-04
Ce144	1.81E-01	1.17E-03	-	1.80E-01
Pr143	1.21E-02	1.11E-04	-	1.09E-02
Pr144	1.79E-01	1.16E-03	-	1.78E-01
Np239	3.44E-02	3.16E-04	-	1.91E-02
Na24	6.17E-01	8.77E-03	1.77E-03	6.67E-02
Cr51	2.39E+00	2.22E-02	4.48E-03	2.28E+00
Mn54	1.11E+01	7.03E-02	1.42E-02	1.11E+01
Fe55	1.31E+01	6.75E-02	1.36E-02	1.31E+01
Fe59	3.72E-01	3.43E-03	6.92E-04	3.61E-01
Co58	9.10E+00	8.14E-02	1.65E-02	8.93E+00
Co60	6.46E+00	3.16E-02	6.37E-03	6.45E+00
Ni63	3.63E+00	1.67E-02	3.38E-03	3.63E+00
Zn65	3.02E+00	2.04E-02	4.11E-03	3.00E+00
Zr95	6.97E-01	6.30E-03	1.27E-03	6.82E-01
Ag110m	7.84E+00	5.25E-02	1.06E-02	7.79E+00
W187	5.87E-02	7.09E-04	1.43E-04	1.44E-02

Table 12.2-8: Chemical and Volume Control System Component Source Terms - Source Strengths

Energy Group	Energy Boundary (MeV)	Design Basis Primary Coolant Photon Spectra (photon/s/gram)	CVCS Mixed Bed (photon/s)	CVCS Cation Bed (photon/s)	CVCS Particulate Filter (photon/s)	CVCS Mixed Bed Transfer - 48-hour decay (photon/s)
1	1.00E-02 - 2.00E-02	2.09E+03	2.15E+11	4.69E+10	2.81E+06	2.06E+11
2	2.00E-02 - 3.00E-02	3.26E+03	2.27E+11	2.49E+10	5.68E+06	2.00E+11
3	3.00E-02 - 4.50E-02	8.54E+04	2.38E+12	5.25E+11	1.07E+06	2.36E+12
4	4.50E-02 - 6.00E-02	5.80E+02	6.18E+10	1.25E+10	1.03E+06	5.79E+10
5	6.00E-02 - 7.00E-02	2.79E+02	3.90E+10	1.21E+10	1.56E+06	3.68E+10
6	7.00E-02 - 7.50E-02	1.14E+02	1.03E+10	2.33E+09	9.24E+05	9.91E+09
7	7.50E-02 - 1.00E-01	7.16E+04	1.64E+11	1.51E+10	5.85E+05	1.45E+11
8	1.00E-01 - 1.50E-01	1.06E+03	1.63E+11	8.69E+09	1.87E+06	1.11E+11
9	1.50E-01 - 2.00E-01	1.19E+03	7.61E+10	2.47E+10	1.23E+06	6.66E+10
10	2.00E-01 - 2.60E-01	7.24E+03	1.09E+11	4.97E+09	7.87E+05	5.47E+10
11	2.60E-01 - 3.00E-01	2.69E+02	1.26E+11	1.51E+10	2.40E+05	1.06E+11
12	3.00E-01 - 4.00E-01	1.37E+03	1.33E+12	5.75E+10	1.61E+07	1.12E+12
13	4.00E-01 - 4.50E-01	4.71E+02	1.89E+10	2.28E+08	1.50E+07	1.37E+10
14	4.50E-01 - 5.10E-01	2.95E+02	5.76E+11	1.48E+11	1.58E+06	5.72E+11
15	5.10E-01 - 5.12E-01	5.37E+02	1.08E+11	9.29E+08	1.85E+08	1.04E+11
16	5.12E-01 - 6.00E-01	2.61E+03	9.38E+12	2.42E+12	5.09E+05	9.25E+12
17	6.00E-01 - 7.00E-01	2.84E+03	6.21E+13	1.52E+13	4.58E+08	6.20E+13
18	7.00E-01 - 8.00E-01	1.82E+03	3.66E+13	9.52E+12	2.16E+08	3.66E+13
19	8.00E-01 - 9.00E-01	1.37E+03	2.80E+12	5.30E+11	1.42E+09	2.75E+12
20	9.00E-01 - 1.00E+00	3.01E+02	1.14E+11	8.14E+08	1.33E+08	1.06E+11
21	1.00E+00 - 1.20E+00	9.71E+02	1.62E+12	3.72E+11	3.44E+08	1.58E+12
22	1.20E+00 - 1.33E+00	8.27E+03	2.14E+11	2.67E+10	1.36E+08	1.87E+11
23	1.33E+00 - 1.44E+00	8.41E+02	1.35E+12	2.99E+11	2.76E+08	1.33E+12
24	1.44E+00 - 1.50E+00	1.19E+02	1.76E+10	9.46E+07	1.64E+07	1.28E+10
25	1.50E+00 - 1.57E+00	1.98E+02	4.39E+10	4.39E+08	5.73E+07	4.24E+10
26	1.57E+00 - 1.66E+00	3.23E+01	3.05E+09	2.76E+07	9.52E+04	2.74E+09
27	1.66E+00 - 1.80E+00	3.20E+02	1.21E+10	2.28E+07	3.10E+06	2.02E+09
28	1.80E+00 - 2.00E+00	3.29E+02	2.21E+09	1.40E+08	6.65E+04	8.22E+08
29	2.00E+00 - 2.15E+00	1.97E+02	1.02E+09	1.01E+07	3.93E+03	3.29E+08
30	2.15E+00 - 2.35E+00	3.02E+02	7.67E+08	6.64E+07	2.71E+03	2.09E+08
31	2.35E+00 - 2.50E+00	4.47E+02	6.34E+08	2.16E+06	2.91E+00	9.77E+07
32	2.50E+00 - 2.75E+00	3.67E+02	1.22E+10	2.21E+08	3.43E+07	1.41E+09
33	2.75E+00 - 3.00E+00	1.82E+02	1.09E+10	1.56E+08	3.13E+07	1.18E+09
34	3.00E+00 - 3.50E+00	1.65E+01	1.56E+07	4.82E+06	6.00E-02	8.98E+05
35	3.50E+00 - 4.00E+00	8.39E+00	2.28E+07	1.04E+06	5.00E+04	1.88E+06
36	4.00E+00 - 4.50E+00	2.09E+00	7.21E+05	1.80E+05	5.49E+02	2.07E+04
37	4.50E+00 - 5.00E+00	3.75E+00	2.45E+06	1.10E+06	-	-
38	5.00E+00 - 5.50E+00	1.28E+00	4.62E+03	2.07E+03	-	-
39	5.50E+00 - 6.00E+00	6.30E-01	-	-	-	-
40	6.00E+00 - 6.50E+00	4.16E+03	-	-	-	-
41	6.50E+00 - 7.00E+00	2.72E+00	-	-	-	-
42	7.00E+00 - 7.50E+00	3.04E+02	-	-	-	-
43	7.50E+00 - 8.00E+00	8.70E-02	-	-	-	-

Table 12.2-8: Chemical and Volume Control System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy Boundary (MeV)	Design Basis Primary Coolant Photon Spectra (photon/s/gram)	CVCS Mixed Bed (photon/s)	CVCS Cation Bed (photon/s)	CVCS Particulate Filter (photon/s)	CVCS Mixed Bed Transfer - 48-hour decay (photon/s)
44	8.00E+00 - 1.00E+01	4.85E+00	-	-	-	-
45	1.00E+01 - 1.20E+01	1.24E-03	-	-	-	-
46	1.20E+01 - 1.40E+01	-	-	-	-	-
47	1.40E+01 - 2.00E+01	-	-	-	-	-
Total		2.02E+05	1.20E+14	2.93E+13	3.36E+09	1.19E+14

Table 12.2-9: Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup, and Pool Surge Control System Component Source Term Inputs and Assumptions

Model Parameter	Value
Reactor pool cooling heat exchanger:	
Contents	100% pool water
Source term mass	3.88E+06 grams
Geometry	horizontal cylinder
Source dimensions	diameter=2′-8″; height=24′-7″
Shielding thickness of steel shell	1/4″
Spent fuel pool cooling heat exchanger:	
Contents	100% pool water
Source term mass	3.88E+06 grams
Geometry	horizontal cylinder
Source dimensions	diameter=2′-8″; height=24′-7″
Shielding thickness of steel shell	1/4″
Pool cleanup system demineralizer:	
Geometry	vertical cylinder
Source dimensions	diameter=13′; height=5′
Shielding thickness of steel shell	1/2″
Operation time	2 years (12 refuelings)
PCU filters:	
PCU filter efficiency	9.1%
Geometry	vertical cylinder
Source dimensions	diameter=2′; height=9′-8″
Shielding thickness of steel shell	1/2″
PCU filter operation time	1 year (6 refuelings)
PSCS Surge Tank	
Contents	cleaned up pool water
Geometry	vertical cylinder
Source dimensions	diameter=61′; height=50′
Shielding thickness of steel wall	1/4″
Source volume	1.46E+05 ft^3
Source Mass	4.14E+09 grams

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-10: Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup and Pool Surge Control System Component Source Terms - Radionuclide Content

Isotope	RPCS Heat Exchanger (Ci)	Spent Fuel Pool Cooling Heat Exchanger (Ci)	PCUS Demineralizer (Ci)	Reactor Pool Water (Ci/g)	PSC Surge Tank (Ci/g)	PCU Filter (Ci)
Br82	9.09E-08	9.09E-08	2.34E-05	2.34E-14	2.34E-16	-
Br83	2.86E-12	2.86E-12	5.00E-11	7.36E-19	-	-
Br84	-	-	-	-	-	-
Br85	-	-	-	-	-	-
I129	5.44E-12	5.44E-12	4.82E-07	1.40E-18	1.40E-20	-
I130	1.42E-07	1.42E-07	1.28E-05	3.67E-14	3.67E-16	-
I131	3.89E-03	3.89E-03	5.46E+00	1.00E-09	1.00E-11	-
I132	3.24E-06	3.24E-06	1.82E-03	8.36E-13	8.36E-15	-
I133	4.61E-04	4.61E-04	6.99E-02	1.19E-10	1.19E-12	-
I134	-	-	3.05E-20	-	-	-
I135	3.76E-07	3.76E-07	1.80E-05	9.69E-14	9.69E-16	-
Rb86m	-	-	-	-	-	-
Rb86	5.71E-07	5.71E-07	1.80E-03	1.47E-13	7.35E-14	-
Rb88	-	-	-	-	-	-
Rb89	-	-	-	-	-	-
Cs132	9.64E-09	9.64E-09	1.06E-05	2.48E-15	1.24E-15	-
Cs134	1.05E-04	1.05E-04	6.59E+00	2.72E-11	1.36E-11	-
Cs135m	-	-	-	-	-	-
Cs136	2.03E-05	2.03E-05	4.52E-02	5.22E-12	2.61E-12	-
Cs137	6.47E-05	6.47E-05	5.43E+00	1.67E-11	8.33E-12	-
Cs138	-	-	-	-	-	-
P32	8.23E-13	8.23E-13	2.05E-09	2.12E-19	-	-
Co57	6.69E-15	6.69E-15	2.69E-10	-	-	-
Sr89	3.97E-08	3.97E-08	3.51E-04	1.02E-14	2.05E-16	-
Sr90	9.04E-09	9.04E-09	7.82E-04	2.33E-15	4.66E-17	-
Sr91	8.19E-10	8.19E-10	5.75E-08	2.11E-16	4.22E-18	-
Sr92	9.02E-14	9.02E-14	1.75E-12	2.33E-20	-	-
Y90	4.83E-09	4.83E-09	7.80E-04	1.25E-15	2.49E-17	-
Y91m	5.22E-10	5.22E-10	3.67E-08	1.35E-16	2.69E-18	-
Y91	5.77E-09	5.77E-09	5.91E-05	1.49E-15	2.98E-17	-
Y92	6.17E-12	6.17E-12	1.61E-10	1.59E-18	3.18E-20	-
Y93	2.08E-10	2.08E-10	1.54E-08	5.36E-17	1.07E-18	-
Zr97	1.02E-09	1.02E-09	1.24E-07	2.63E-16	5.25E-18	-
Nb95	3.81E-06	3.81E-06	1.06E+00	9.83E-13	1.97E-14	-
Mo99	7.34E-06	7.34E-06	3.53E-03	1.89E-12	3.78E-14	-
Mo101	-	-	-	-	-	-
Tc99m	7.08E-06	7.08E-06	3.41E-03	1.83E-12	3.65E-14	-
Tc99	3.37E-10	3.37E-10	2.99E-05	8.69E-17	1.74E-18	-
Ru103	1.09E-08	1.09E-08	7.49E-05	2.81E-15	5.62E-17	-
Ru105	3.29E-12	3.29E-12	1.07E-10	8.48E-19	1.70E-20	-
Ru106	7.28E-09	7.28E-09	3.53E-04	1.88E-15	3.75E-17	-
Rh103m	1.08E-08	1.08E-08	7.40E-05	2.78E-15	5.55E-17	-
Rh105	3.48E-09	3.48E-09	8.97E-07	8.96E-16	1.79E-17	-

Table 12.2-10: Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup and Pool Surge Control System Component Source Terms - Radionuclide Content (Continued)

Isotope	RPCS Heat Exchanger (Ci)	Spent Fuel Pool Cooling Heat Exchanger (Ci)	PCUS Demineralizer (Ci)	Reactor Pool Water (Ci/g)	PSC Surge Tank (Ci/g)	PCU Filter (Ci)
Rh106	7.28E-09	7.28E-09	3.53E-04	1.88E-15	3.75E-17	-
Ag110	4.81E-06	4.81E-06	1.64E-01	1.24E-12	2.48E-14	-
Sb124	1.63E-11	1.63E-11	1.71E-07	4.19E-18	8.38E-20	-
Sb125	1.46E-10	1.46E-10	1.02E-05	3.77E-17	7.53E-19	-
Sb127	4.52E-10	4.52E-10	3.05E-07	1.17E-16	2.33E-18	-
Sb129	6.50E-13	6.50E-13	2.09E-11	1.68E-19	-	-
Te125m	2.10E-08	2.10E-08	2.13E-04	5.41E-15	1.08E-16	-
Te127m	6.83E-08	6.83E-08	1.29E-03	1.76E-14	3.52E-16	-
Te127	7.47E-08	7.47E-08	1.27E-03	1.93E-14	3.85E-16	-
Te129m	1.91E-07	1.91E-07	1.12E-03	4.92E-14	9.83E-16	-
Te129	1.20E-07	1.20E-07	7.08E-04	3.10E-14	6.20E-16	-
Te131m	2.29E-07	2.29E-07	5.01E-05	5.91E-14	1.18E-15	-
Te131	5.16E-08	5.16E-08	1.13E-05	1.33E-14	2.66E-16	-
Te132	3.15E-06	3.15E-06	1.76E-03	8.11E-13	1.62E-14	-
Te133m	-	-	-	-	-	-
Te134	-	-	-	-	-	-
Ba137m	6.11E-05	6.11E-05	5.12E+00	1.57E-11	3.15E-13	-
Ba139	1.75E-18	1.75E-18	1.76E-17	-	-	-
Ba140	5.25E-08	5.25E-08	1.17E-04	1.35E-14	2.70E-16	-
La140	3.73E-08	3.73E-08	1.28E-04	9.62E-15	1.93E-16	-
La141	1.16E-12	1.16E-12	3.31E-11	2.98E-19	-	-
La142	1.89E-18	1.89E-18	2.09E-17	-	-	-
Ce141	8.61E-09	8.61E-09	4.90E-05	2.22E-15	4.44E-17	-
Ce143	2.64E-09	2.64E-09	6.35E-07	6.79E-16	1.36E-17	-
Ce144	7.48E-09	7.48E-09	3.10E-04	1.93E-15	3.86E-17	-
Pr143	7.63E-09	7.63E-09	1.88E-05	1.97E-15	3.93E-17	-
Pr144	7.41E-09	7.41E-09	3.07E-04	1.91E-15	3.82E-17	-
Np239	8.19E-08	8.19E-08	3.38E-05	2.11E-14	4.22E-16	-
Na24	1.85E-06	1.85E-06	1.82E-04	4.78E-13	9.56E-15	2.05E-05
Cr51	8.11E-04	8.11E-04	3.53E+00	2.09E-10	4.18E-12	3.98E-01
Mn54	4.35E-04	4.35E-04	1.72E+01	1.12E-10	2.24E-12	1.34E+00
Fe55	3.27E-04	3.27E-04	2.04E+01	8.43E-11	1.69E-12	1.30E+00
Fe59	7.97E-05	7.97E-05	5.58E-01	2.05E-11	4.11E-13	6.26E-02
Co58	1.23E-02	1.23E-02	1.38E+02	3.18E-09	6.36E-11	1.51E+01
Co60	1.45E-04	1.45E-04	1.01E+01	3.72E-11	7.45E-13	6.08E-01
Ni63	7.22E-05	7.22E-05	5.71E+00	1.86E-11	3.72E-13	3.23E-01
Zn65	1.38E-04	1.38E-04	4.65E+00	3.57E-11	7.14E-13	3.87E-01
Zr95	1.04E-04	1.04E-04	1.05E+00	2.69E-11	5.38E-13	1.16E-01
Ag110m	3.53E-04	3.53E-04	1.21E+01	9.11E-11	1.82E-12	9.97E-01
W187	2.04E-04	2.04E-04	3.18E-02	5.27E-11	1.05E-12	3.58E-03
H3	3.25E-01	3.25E-01	-	8.37E-08	8.37E-08	-
C14	1.41E-06	1.41E-06	-	3.63E-13	3.63E-13	-

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-11: Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup, and Pool Surge Control System Component Source Terms - Source Strengths

Energy Group	Energy Boundary (MeV)	PCUS Demineralizer (photon/s)	Reactor Pool Water (photon/s)	PSC Surge Tank (photon/s/g)	PCU Filter (photon/s)
1	1.00E-02 - 2.00E-02	1.23E+10	1.26E+10	1.20E-02	1.08E+09
2	2.00E-02 - 3.00E-02	1.88E+10	4.14E+10	1.75E-02	9.93E+08
3	3.00E-02 - 4.50E-02	2.42E+10	2.56E+10	2.12E-02	4.65E+08
4	4.50E-02 - 6.00E-02	3.84E+09	6.15E+09	4.48E-03	3.54E+08
5	6.00E-02 - 7.00E-02	2.13E+09	1.45E+10	1.47E-02	1.95E+08
6	7.00E-02 - 7.50E-02	9.30E+08	8.81E+09	6.16E-03	9.01E+07
7	7.50E-02 - 1.00E-01	8.77E+09	2.83E+10	1.57E-02	2.80E+08
8	1.00E-01 - 1.50E-01	3.93E+09	1.13E+10	8.13E-03	3.65E+08
9	1.50E-01 - 2.00E-01	3.50E+09	6.27E+09	1.81E-02	2.70E+08
10	2.00E-01 - 2.60E-01	1.84E+09	2.05E+09	1.62E-03	1.58E+08
11	2.60E-01 - 3.00E-01	1.35E+10	6.82E+10	3.42E-02	6.28E+07
12	3.00E-01 - 4.00E-01	1.87E+11	9.29E+11	3.70E-01	1.52E+09
13	4.00E-01 - 4.50E-01	1.75E+10	5.11E+09	3.14E-03	1.44E+09
14	4.50E-01 - 5.10E-01	5.07E+09	1.97E+10	2.04E-02	7.38E+07
15	5.10E-01 - 5.12E-01	1.52E+12	1.01E+12	7.03E-01	1.67E+11
16	5.12E-01 - 6.00E-01	6.15E+10	1.16E+11	1.61E-01	3.28E+07
17	6.00E-01 - 7.00E-01	9.30E+11	2.58E+11	5.91E-01	4.29E+10
18	7.00E-01 - 8.00E-01	5.08E+11	1.23E+11	5.39E-01	2.02E+10
19	8.00E-01 - 9.00E-01	5.85E+12	3.44E+12	2.49E+00	6.13E+11
20	9.00E-01 - 1.00E+00	1.52E+11	3.33E+10	2.30E-02	1.25E+10
21	1.00E+00 - 1.20E+00	5.06E+11	7.94E+10	1.30E-01	3.27E+10
22	1.20E+00 - 1.33E+00	2.07E+11	3.66E+10	4.19E-02	1.29E+10
23	1.33E+00 - 1.44E+00	2.99E+11	4.49E+10	4.55E-02	2.01E+10
24	1.44E+00 - 1.50E+00	1.88E+10	4.08E+09	2.84E-03	1.55E+09
25	1.50E+00 - 1.57E+00	6.54E+10	1.41E+10	9.94E-03	5.39E+09
26	1.57E+00 - 1.66E+00	1.14E+08	3.88E+07	2.53E-05	8.95E+06
27	1.66E+00 - 1.80E+00	2.55E+10	1.70E+10	1.18E-02	2.80E+09
28	1.80E+00 - 2.00E+00	7.77E+07	3.60E+07	1.89E-05	6.26E+06
29	2.00E+00 - 2.15E+00	5.25E+06	9.72E+06	3.94E-06	3.70E+05
30	2.15E+00 - 2.35E+00	4.66E+06	4.82E+06	2.01E-06	2.59E+05
31	2.35E+00 - 2.50E+00	2.66E+05	2.93E+06	1.04E-06	2.32E+02
32	2.50E+00 - 2.75E+00	3.73E+06	2.66E+08	1.85E-04	3.97E+05
33	2.75E+00 - 3.00E+00	3.23E+06	2.42E+08	1.69E-04	3.63E+05
34	3.00E+00 - 3.50E+00	1.54E+03	2.86E+03	1.99E-09	6.95E-04
35	3.50E+00 - 4.00E+00	5.14E+03	3.87E+05	2.70E-07	5.79E+02
36	4.00E+00 - 4.50E+00	5.64E+01	4.25E+03	2.96E-09	6.35E+00
37	4.50E+00 - 5.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
38	5.00E+00 - 5.50E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
39	5.50E+00 - 6.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
40	6.00E+00 - 6.50E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
41	6.50E+00 - 7.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
42	7.00E+00 - 7.50E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
43	7.50E+00 - 8.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 12.2-11: Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup, and Pool Surge Control System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy Boundary (MeV)	PCUS Demineralizer (photon/s)	Reactor Pool Water (photon/s)	PSC Surge Tank (photon/s/g)	PCU Filter (photon/s)
44	8.00E+00 - 1.00E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
45	1.00E+01 - 1.20E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
46	1.20E+01 - 1.40E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
47	1.40E+01 - 2.00E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total		1.04E+13	6.35E+12	5.30E+00	9.38E+11

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-12: Liquid Radioactive Waste System Component Source Term Inputs and Assumptions

Model Parameter	Value
LRWS degasifier	
Contents	CVCS letdown (see Table 11.2-11)
Geometry	vertical cylinder
Source dimensions	diameter=12′; height=14′
Shield thickness of steel shell	1″
Volume	12,500 gallons
LCW and HCW collection tanks	
Inputs	Table 11.2-3
Geometry	vertical cylinder
Source dimensions	diameter=12′; height=15.13′
Shield thickness of steel shell	0.25″
Volume	12,800 gallons
LRWS oil separator	
Inputs	Table 11.2-3
Geometry	parallelepiped
Source dimensions	length=3′; width=10′; height=4′
Shield thickness of steel shell	0.25″
LCW and HCW granulated activated charcoal (GAC) units	
Decontamination Factors	(from Reference 12.2-2)
• 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=3′; height=6′
Shield thickness of steel shell	(Table 12.3-7)
LCW and HCW tubular ultrafiltration (TUF) units	
Decontamination factors	
• All nuclides	2.5
Geometry	vertical cylinder
Source dimensions	diameter=39″; height=47.5″
Shield thickness of steel shell	(Table 12.3-7)
LCW and HCW reverse osmosis (RO) units	
Decontamination factors	
• All nuclides	10
Geometry	vertical cylinder
Source dimensions	diameter=39″; height=47.5″
Shield thickness of steel shell	(Table 12.3-7)

Table 12.2-12: Liquid Radioactive Waste System Component Source Term Inputs and Assumptions (Continued)

Model Parameter	Value
LCW cation bed	
Decontamination factors	
• Anions	1
• Cs, Rb	10
• Others	10
Geometry	vertical cylinder
Source dimensions of vessel	diameter=2.78′; height=2.9′
Shield thickness of steel shell	(Table 12.3-7)
LCW anion bed	
Decontamination factors	
• Anions	100
• Cs, Rb	1
• Others	1
Geometry	vertical cylinder
Source dimensions of vessel	diameter=2.78′; height=2.9′
Shield thickness of steel shell	(Table 12.3-7)
LCW cesium bed	
Decontamination factors	
• Anions	1
• Cs, Rb	10
• Others	10
Geometry	vertical cylinder
Source dimensions of vessel	diameter=2.78′; height=2.9′
Shield thickness of steel shell	(Table 12.3-7)
LCW mixed bed	
Decontamination Factors	
• Anion	100
• Cs, Rb	2
• Others	100
Geometry	vertical cylinder
Source dimensions of vessel	diameter=4.46′; height=1.68′
Shield thickness of steel shell	(Table 12.3-7)
LCW and HCW sample tanks	
Geometry	vertical cylinder
Source dimensions	diameter=12′; height=15.13′
Shield thickness of steel shell	(Table 12.3-7)
Drum dryer	
Inputs	TUF and RO rejects
Geometry	vertical cylinder
Source dimensions	diameter=18″; height=28″

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

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

Isotope	LCW Collection Tank (Ci)	HCW Collection Tank (Ci)	Oil Separator (Ci)	LCW GAC Unit (Ci)	LCW TUF Unit (Ci)	LCW RO Unit (Ci)	LCW Cation Bed (Ci)	LCW Anion Bed (Ci)
Br82	2.95E-05	1.22E-04	7.54E-06	-	4.71E-06	2.83E-06	-	3.19E-07
Br83	1.66E-04	6.97E-04	4.31E-05	-	1.85E-06	1.11E-06	-	1.22E-07
Br84	7.73E-05	3.24E-04	2.01E-05	-	1.90E-07	1.14E-07	-	1.25E-08
Br85	9.34E-06	3.92E-05	2.42E-06	-	2.09E-09	1.26E-09	-	1.38E-10
I129	7.48E-10	3.02E-09	1.87E-10	-	4.49E-10	2.69E-10	2.32E-13	1.39E-09
I130	2.35E-04	9.84E-04	6.09E-05	-	1.35E-05	8.09E-06	-	8.90E-07
I131	2.67E-02	2.75E-02	1.57E-03	-	1.17E-02	7.00E-03	1.54E-06	1.57E-03
I132	2.77E-03	1.15E-02	7.14E-04	-	5.25E-04	3.15E-04	3.68E-05	1.94E-06
I133	1.15E-02	3.84E-02	2.36E-03	-	1.11E-03	6.67E-04	3.07E-08	7.34E-05
I134	1.62E-03	6.78E-03	4.20E-04	-	7.23E-06	4.34E-06	4.00E-08	4.33E-07
I135	5.72E-03	2.40E-02	1.49E-03	-	1.74E-04	1.05E-04	-	1.15E-05
Rb86m	4.03E-07	6.40E-08	1.76E-09	-	3.17E-11	1.90E-11	1.90E-12	-
Rb86	2.40E-03	3.80E-04	1.04E-05	-	1.25E-03	7.49E-04	2.98E-04	-
Rb88	4.08E-01	6.48E-02	1.78E-03	-	5.60E-04	3.36E-04	3.36E-05	-
Rb89	1.87E-02	2.97E-03	8.16E-05	-	2.19E-05	1.32E-05	1.31E-06	-
Cs132	4.62E-05	7.33E-06	2.01E-07	-	1.88E-05	1.13E-05	2.00E-06	-
Cs134	4.13E-01	6.56E-02	1.80E-03	-	2.47E-01	1.48E-01	5.93E-01	-
Cs135m	3.13E-04	4.97E-05	1.36E-06	-	1.28E-06	7.68E-07	7.68E-08	-
Cs136	8.76E-02	1.39E-02	3.82E-04	-	4.31E-02	2.59E-02	7.70E-03	-
Cs137	2.53E-01	4.02E-02	1.10E-03	-	1.52E-01	9.12E-02	4.23E-01	-
Cs138	1.50E-01	2.38E-02	6.53E-04	-	3.86E-04	2.32E-04	2.32E-05	-
P32	3.30E-10	4.88E-10	2.98E-11	-	1.65E-10	9.88E-11	3.14E-11	-
Co57	2.46E-12	3.64E-12	2.22E-13	-	1.46E-12	8.77E-13	2.70E-12	-
Sr89	1.47E-05	2.18E-05	1.33E-06	-	1.06E-05	6.35E-06	6.23E-06	-
Sr90	3.31E-06	4.90E-06	2.99E-07	-	1.98E-06	1.19E-06	5.52E-06	-
Sr91	7.54E-06	1.13E-05	6.91E-07	-	3.37E-07	2.02E-07	2.02E-08	-
Sr92	4.03E-06	6.04E-06	3.69E-07	-	4.97E-08	2.98E-08	2.98E-09	-
Y90	8.17E-07	1.19E-06	7.26E-08	-	1.34E-06	8.06E-07	5.48E-06	-
Y91m	4.04E-06	6.06E-06	3.70E-07	-	2.12E-07	1.27E-07	1.27E-08	-
Y91	2.14E-06	3.16E-06	1.93E-07	-	1.24E-06	7.42E-07	8.32E-07	-
Y92	3.42E-06	5.13E-06	3.14E-07	-	1.06E-07	6.35E-08	6.35E-09	-
Y93	1.61E-06	2.41E-06	1.47E-07	-	7.59E-08	4.55E-08	4.55E-09	-
Zr97	2.37E-06	3.55E-06	2.17E-07	-	1.84E-07	1.10E-07	1.10E-08	-
Nb95	2.37E-05	7.33E-06	3.14E-07	1.53E-04	3.82E-05	2.30E-05	2.46E-04	-
Mo99	4.29E-03	6.38E-03	3.90E-04	-	1.13E-03	6.76E-04	7.86E-05	-
Mo101	1.60E-04	2.40E-04	1.47E-05	-	1.81E-07	1.09E-07	1.09E-08	-
Tc99m	3.97E-03	5.90E-03	3.61E-04	-	1.08E-03	6.50E-04	7.56E-05	-
Tc99	1.23E-07	1.83E-07	1.12E-08	-	7.41E-08	4.44E-08	2.09E-07	-
Ru103	4.12E-06	6.10E-06	3.73E-07	-	2.31E-06	1.39E-06	1.08E-06	-
Ru105	1.34E-06	2.00E-06	1.22E-07	-	2.75E-08	1.65E-08	1.65E-09	-
Ru106	2.67E-06	3.96E-06	2.42E-07	-	1.59E-06	9.56E-07	3.28E-06	-
Rh103m	4.08E-06	6.03E-06	3.68E-07	-	2.28E-06	1.37E-06	1.07E-06	-
Rh105	2.86E-06	4.26E-06	2.60E-07	-	4.83E-07	2.90E-07	2.97E-08	-

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

Isotope	LCW Collection Tank (Ci)	HCW Collection Tank (Ci)	Oil Separator (Ci)	LCW GAC Unit (Ci)	LCW TUF Unit (Ci)	LCW RO Unit (Ci)	LCW Cation Bed (Ci)	LCW Anion Bed (Ci)
Rh106	2.67E-06	3.96E-06	2.42E-07	-	1.59E-06	9.56E-07	3.28E-06	-
Ag110	2.86E-05	7.38E-06	2.82E-07	1.25E-03	1.62E-05	9.71E-06	2.90E-05	-
Sb124	6.08E-09	9.00E-09	5.49E-10	5.83E-08	3.49E-09	2.09E-09	2.41E-09	-
Sb125	5.36E-08	7.93E-08	4.84E-09	6.15E-06	3.21E-08	1.92E-08	8.00E-08	-
Sb127	2.31E-07	3.43E-07	2.10E-08	1.42E-07	7.46E-08	4.48E-08	5.94E-09	-
Sb129	2.81E-07	4.22E-07	2.58E-08	8.21E-09	5.74E-09	3.44E-09	3.44E-10	-
Te125m	7.86E-06	1.16E-05	7.10E-07	1.39E-06	4.50E-06	2.70E-06	2.99E-06	-
Te127m	2.53E-05	3.75E-05	2.29E-06	2.51E-08	1.48E-05	8.89E-06	1.66E-05	-
Te127	9.93E-05	1.48E-04	9.07E-06	1.17E-07	1.78E-05	1.07E-05	1.65E-05	-
Te129m	7.25E-05	1.07E-04	6.56E-06	1.86E-09	4.02E-05	2.41E-05	1.63E-05	-
Te129	1.02E-04	1.52E-04	9.27E-06	6.35E-09	2.57E-05	1.54E-05	1.03E-05	-
Te131m	2.35E-04	3.51E-04	2.14E-05	-	3.22E-05	1.93E-05	1.96E-06	-
Te131	1.15E-04	1.73E-04	1.05E-05	-	7.38E-06	4.43E-06	4.49E-07	-
Te132	1.72E-03	2.56E-03	1.56E-04	-	4.99E-04	2.99E-04	3.68E-05	-
Te133m	1.45E-04	2.18E-04	1.33E-05	-	6.21E-07	3.73E-07	3.73E-08	-
Te134	2.07E-04	3.10E-04	1.89E-05	-	6.67E-07	4.00E-07	4.00E-08	-
Ba137m	1.16E-02	1.70E-02	1.04E-03	-	1.44E-01	8.61E-02	4.00E-01	-
Ba139	3.84E-06	5.77E-06	3.52E-07	-	2.47E-08	1.48E-08	1.48E-09	-
Ba140	2.12E-05	3.15E-05	1.92E-06	-	1.04E-05	6.23E-06	1.81E-06	-
La140	6.28E-06	9.14E-06	5.57E-07	-	8.71E-06	5.22E-06	1.88E-06	-
La141	1.19E-06	1.79E-06	1.09E-07	-	2.17E-08	1.30E-08	1.30E-09	-
La142	5.69E-07	8.54E-07	5.22E-08	-	4.01E-09	2.40E-09	2.40E-10	-
Ce141	3.27E-06	4.85E-06	2.96E-07	-	1.81E-06	1.09E-06	7.11E-07	-
Ce143	2.46E-06	3.67E-06	2.24E-07	-	3.69E-07	2.21E-07	2.26E-08	-
Ce144	2.75E-06	4.07E-06	2.49E-07	-	1.64E-06	9.81E-07	3.08E-06	-
Pr143	2.91E-06	4.31E-06	2.63E-07	-	1.54E-06	9.23E-07	2.86E-07	-
Pr144	2.73E-06	4.03E-06	2.46E-07	-	1.62E-06	9.72E-07	3.05E-06	-
Np239	5.17E-05	7.70E-05	4.70E-06	-	1.22E-05	7.31E-06	8.14E-07	-
Na24	5.39E-03	8.07E-03	4.93E-04	-	3.74E-04	2.24E-04	2.24E-05	-
Cr51	4.61E-03	9.26E-04	2.81E-05	2.36E-02	2.52E-03	1.51E-03	8.53E-04	-
Mn54	2.47E-03	4.87E-04	1.45E-05	1.39E-01	1.47E-03	8.81E-04	2.86E-03	-
Fe55	1.85E-03	3.65E-04	1.08E-05	-	1.11E-03	6.66E-04	2.77E-03	-
Fe59	4.53E-04	9.02E-05	2.71E-06	-	2.56E-04	1.54E-04	1.34E-04	-
Co58	6.60E-02	7.78E-03	4.16E-05	8.02E-01	3.81E-02	2.29E-02	3.03E-02	-
Co60	8.19E-04	1.61E-04	4.78E-06	1.20E-01	4.91E-04	2.95E-04	1.30E-03	-
Ni63	4.09E-04	8.06E-05	2.39E-06	-	2.46E-04	1.47E-04	6.89E-04	-
Zn65	7.85E-04	1.55E-04	4.60E-06	-	4.66E-04	2.80E-04	8.26E-04	-
Zr95	5.93E-04	1.18E-04	3.52E-06	-	3.41E-04	2.05E-04	2.49E-04	-
Ag110m	2.01E-03	3.95E-04	1.17E-05	9.22E-02	1.19E-03	7.14E-04	2.13E-03	-
W187	1.36E-03	5.29E-04	2.52E-05	-	1.49E-04	8.93E-05	8.97E-06	-
H3	7.08E+01	1.84E+01	9.96E-01	-	-	-	-	-
C14	5.51E-03	6.91E-04	1.21E-05	-	-	-	-	-

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

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

Isotope	LCW Mixed Bed (Ci)	LCW Cesium Bed (Ci)	LCW Sample Tank (Ci)	HCW GAC Unit (Ci)	HCW TUF Unit (Ci)	HCW RO Unit (Ci)	HCW Sample Tank (Ci)	Drum Dryer (Ci)
Br82	3.19E-09	-	9.27E-12	-	2.01E-05	1.21E-05	3.83E-07	4.09E-05
Br83	1.22E-09	-	-	-	8.05E-06	4.83E-06	1.55E-21	1.58E-05
Br84	1.25E-10	-	-	-	8.27E-07	4.96E-07	-	1.63E-06
Br85	1.38E-12	-	-	-	9.11E-09	5.47E-09	-	1.79E-08
I129	1.39E-11	2.32E-16	2.99E-15	-	1.81E-09	1.09E-09	1.21E-10	1.75E-07
I130	8.90E-09	-	6.57E-13	-	5.85E-05	3.51E-05	2.74E-08	1.15E-04
I131	1.59E-05	1.54E-09	6.70E-08	-	1.22E-02	7.31E-03	6.92E-04	7.90E-02
I132	4.06E-06	3.68E-08	2.20E-09	-	8.81E-04	5.29E-04	3.28E-05	2.75E-03
I133	7.38E-07	3.07E-11	6.15E-10	-	3.84E-03	2.30E-03	2.05E-05	7.93E-03
I134	8.73E-09	4.00E-11	-	-	2.96E-05	1.78E-05	-	5.89E-05
I135	1.15E-07	-	2.64E-14	-	7.59E-04	4.55E-04	1.11E-09	1.49E-03
Rb86m	1.06E-13	9.50E-14	-	-	5.22E-12	3.13E-12	-	5.90E-11
Rb86	1.66E-05	1.49E-05	3.92E-07	-	1.99E-04	1.20E-04	1.25E-05	9.26E-03
Rb88	1.87E-06	1.68E-06	-	-	9.24E-05	5.54E-05	-	1.04E-03
Rb89	7.30E-08	6.57E-08	-	-	3.61E-06	2.17E-06	-	4.08E-05
Cs132	1.11E-07	9.98E-08	5.18E-09	-	3.02E-06	1.81E-06	1.64E-07	6.20E-05
Cs134	3.30E-02	2.97E-02	8.23E-05	-	3.93E-02	2.36E-02	2.61E-03	1.84E+01
Cs135m	4.27E-09	3.84E-09	-	-	2.11E-07	1.27E-07	-	2.39E-06
Cs136	4.28E-04	3.85E-04	1.32E-05	-	6.90E-03	4.14E-03	4.19E-04	2.39E-01
Cs137	2.35E-02	2.12E-02	5.07E-05	-	2.41E-02	1.45E-02	1.61E-03	1.32E+01
Cs138	1.29E-06	1.16E-06	-	-	6.37E-05	3.82E-05	-	7.20E-04
P32	3.45E-12	3.14E-14	1.01E-15	-	2.46E-10	1.47E-10	1.50E-11	2.12E-09
Co57	2.97E-13	2.70E-15	9.71E-18	-	2.16E-12	1.30E-12	1.44E-13	1.83E-10
Sr89	6.15E-07	7.02E-08	7.79E-10	-	1.28E-05	7.66E-06	8.32E-07	3.74E-04
Sr90	6.08E-07	5.52E-09	1.32E-11	-	2.94E-06	1.76E-06	1.96E-07	3.74E-04
Sr91	2.22E-09	2.02E-11	2.68E-15	-	5.24E-07	3.14E-07	4.02E-11	1.38E-06
Sr92	3.28E-10	2.98E-12	-	-	7.74E-08	4.64E-08	5.21E-22	2.03E-07
Y90	6.03E-07	5.48E-09	1.08E-11	-	1.96E-06	1.17E-06	1.59E-07	3.71E-04
Y91m	1.40E-09	1.27E-11	1.71E-15	-	3.29E-07	1.98E-07	2.56E-11	8.65E-07
Y91	9.16E-08	8.32E-10	8.11E-12	-	1.84E-06	1.10E-06	1.20E-07	5.63E-05
Y92	6.98E-10	6.35E-12	5.95E-22	-	1.65E-07	9.89E-08	8.92E-18	4.33E-07
Y93	5.01E-10	4.55E-12	9.46E-16	-	1.18E-07	7.09E-08	1.42E-11	3.10E-07
Zr97	1.22E-09	1.10E-11	4.44E-14	-	2.86E-07	1.72E-07	6.64E-10	7.53E-07
Nb95	2.70E-05	2.46E-07	2.31E-10	4.93E-05	8.89E-06	5.34E-06	4.59E-07	7.92E-03
Mo99	8.65E-06	7.86E-08	4.39E-09	-	1.72E-03	1.03E-03	6.53E-05	5.34E-03
Mo101	1.19E-09	1.09E-11	-	-	2.82E-07	1.69E-07	-	7.40E-07
Tc99m	8.32E-06	7.56E-08	4.24E-09	-	1.65E-03	9.91E-04	6.31E-05	5.13E-03
Tc99	2.29E-08	2.09E-10	4.94E-13	-	1.10E-07	6.58E-08	7.31E-09	1.41E-05
Ru103	1.19E-07	1.08E-09	1.50E-11	-	3.43E-06	2.06E-06	2.22E-07	7.30E-05
Ru105	1.81E-10	1.65E-12	8.73E-21	-	4.28E-08	2.57E-08	1.31E-16	1.12E-07
Ru106	3.61E-07	3.28E-09	1.06E-11	-	2.36E-06	1.42E-06	1.57E-07	2.22E-04
Rh103m	1.17E-07	1.07E-09	1.48E-11	-	3.39E-06	2.03E-06	2.19E-07	7.21E-05
Rh105	3.27E-09	2.97E-11	9.62E-13	-	7.45E-07	4.47E-07	1.43E-08	2.02E-06

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

Isotope	LCW Mixed Bed (Ci)	LCW Cesium Bed (Ci)	LCW Sample Tank (Ci)	HCW GAC Unit (Ci)	HCW TUF Unit (Ci)	HCW RO Unit (Ci)	HCW Sample Tank (Ci)	Drum Dryer (Ci)
Rh106	3.61E-07	3.28E-09	1.06E-11	-	2.36E-06	1.42E-06	1.57E-07	2.22E-04
Ag110	3.19E-06	2.90E-08	3.31E-14	2.57E-04	3.19E-06	1.92E-06	6.52E-11	9.30E-04
Sb124	2.65E-10	2.41E-12	7.15E-18	8.96E-08	5.17E-09	3.10E-09	4.76E-11	1.63E-07
Sb125	8.80E-09	8.00E-11	6.68E-17	9.45E-06	4.74E-08	2.85E-08	4.45E-10	5.41E-06
Sb127	6.53E-10	5.94E-12	1.09E-16	2.19E-07	1.13E-07	6.79E-08	7.32E-10	4.03E-07
Sb129	3.79E-11	3.44E-13	4.79E-25	1.28E-08	8.94E-09	5.36E-09	3.23E-18	2.35E-08
Te125m	3.29E-07	2.99E-09	2.95E-11	2.13E-06	6.68E-06	4.01E-06	4.36E-07	2.02E-04
Te127m	1.83E-06	1.66E-08	9.78E-11	3.87E-08	2.20E-05	1.32E-05	1.45E-06	1.12E-03
Te127	1.81E-06	1.65E-08	9.58E-11	1.80E-07	2.66E-05	1.60E-05	1.42E-06	1.11E-03
Te129m	1.79E-06	1.63E-08	2.60E-10	2.89E-09	5.97E-05	3.58E-05	3.84E-06	1.10E-03
Te129	1.13E-06	1.03E-08	1.64E-10	9.90E-09	3.82E-05	2.29E-05	2.42E-06	6.95E-04
Te131m	2.16E-07	1.96E-09	4.71E-11	-	4.99E-05	2.99E-05	7.03E-07	1.33E-04
Te131	4.93E-08	4.49E-10	1.06E-11	-	1.14E-05	6.85E-06	1.58E-07	3.05E-05
Te132	4.04E-06	3.68E-08	2.14E-09	-	7.59E-04	4.55E-04	3.18E-05	2.49E-03
Te133m	4.10E-09	3.73E-11	-	-	9.68E-07	5.81E-07	-	2.54E-06
Te134	4.40E-09	4.00E-11	-	-	1.04E-06	6.24E-07	-	2.73E-06
Ba137m	2.22E-02	2.00E-02	4.78E-05	-	2.28E-02	1.37E-02	1.52E-03	1.24E+01
Ba139	1.63E-10	1.48E-12	-	-	3.84E-08	2.31E-08	-	1.01E-07
Ba140	1.99E-07	1.81E-09	6.33E-11	-	1.55E-05	9.30E-06	9.38E-07	1.22E-04
La140	2.07E-07	1.88E-09	6.51E-11	-	1.28E-05	7.71E-06	9.64E-07	1.27E-04
La141	1.43E-10	1.30E-12	5.32E-22	-	3.38E-08	2.03E-08	7.98E-18	8.87E-08
La142	2.64E-11	2.40E-13	-	-	6.24E-09	3.74E-09	-	1.64E-08
Ce141	7.82E-08	7.11E-10	1.17E-11	-	2.69E-06	1.62E-06	1.73E-07	4.81E-05
Ce143	2.48E-09	2.26E-11	6.48E-13	-	5.70E-07	3.42E-07	9.67E-09	1.54E-06
Ce144	3.39E-07	3.08E-09	1.09E-11	-	2.42E-06	1.45E-06	1.61E-07	2.08E-04
Pr143	3.15E-08	2.86E-10	9.61E-12	-	2.29E-06	1.38E-06	1.42E-07	1.94E-05
Pr144	3.36E-07	3.05E-09	1.08E-11	-	2.40E-06	1.44E-06	1.59E-07	2.06E-04
Np239	8.95E-08	8.14E-10	4.23E-11	-	1.87E-05	1.12E-05	6.29E-07	5.52E-05
Na24	2.47E-06	2.24E-08	5.32E-11	-	5.81E-04	3.49E-04	7.96E-07	1.53E-03
Cr51	9.38E-05	8.53E-07	6.30E-11	4.92E-03	5.07E-04	3.04E-04	1.26E-07	2.75E-02
Mn54	3.14E-04	2.86E-06	9.12E-11	2.85E-02	2.90E-04	1.74E-04	1.80E-07	9.18E-02
Fe55	3.05E-04	2.77E-06	7.39E-09	-	2.19E-04	1.31E-04	1.46E-05	8.89E-02
Fe59	1.47E-05	1.34E-07	1.66E-09	-	5.11E-05	3.06E-05	3.32E-06	4.31E-03
Co58	3.34E-03	3.03E-05	1.90E-08	9.81E-02	4.50E-03	2.70E-03	2.24E-05	9.08E-01
Co60	1.43E-04	1.30E-06	4.88E-10	2.45E-02	9.66E-05	5.80E-05	9.61E-07	4.17E-02
Ni63	7.58E-05	6.89E-07	1.64E-09	-	4.83E-05	2.90E-05	3.22E-06	2.21E-02
Zn65	9.08E-05	8.26E-07	3.09E-09	-	9.20E-05	5.52E-05	6.10E-06	2.65E-02
Zr95	2.73E-05	2.49E-07	2.24E-09	-	6.78E-05	4.07E-05	4.44E-06	8.00E-03
Ag110m	2.34E-04	2.13E-06	2.43E-12	1.89E-02	2.35E-04	1.41E-04	4.79E-09	6.84E-02
W187	9.87E-07	8.97E-09	1.23E-10	-	6.01E-05	3.61E-05	4.80E-07	3.36E-04
H3	-	-	7.07E+01	-	-	-	1.84E+01	-
C14	-	-	5.51E-03	-	-	-	6.91E-04	-

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-14a: Liquid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Energy Boundary (MeV)			LCW Collection Tank (photon/s)	HCW Collection Tank (photon/s)	Oil Separator (photon/s)	LCW GAC Unit (photon/s)	LCW TUF Unit (photon/s)	LCW RO Unit (photon/s)	LCW Cation Bed (photon/s)	LCW Anion Bed (photon/s)
1	1.00E-02	-	2.00E-02	9.05E+08	1.99E+08	7.49E+06	6.33E+07	5.42E+07	3.25E+07	1.05E+08	2.01E+05
2	2.00E-02	-	3.00E-02	5.91E+08	2.23E+08	1.08E+07	7.19E+07	5.47E+07	3.28E+07	5.69E+07	1.94E+06
3	3.00E-02	-	4.50E-02	8.62E+08	2.23E+08	9.27E+06	2.70E+07	5.40E+08	3.25E+08	1.16E+09	1.12E+06
4	4.50E-02	-	6.00E-02	3.28E+08	7.41E+07	2.84E+06	1.99E+07	1.63E+07	9.75E+06	2.83E+07	5.46E+04
5	6.00E-02	-	7.00E-02	3.25E+08	5.88E+07	1.86E+06	9.41E+06	8.58E+07	5.16E+07	2.67E+07	2.46E+04
6	7.00E-02	-	7.50E-02	7.43E+07	1.54E+07	5.48E+05	4.02E+06	3.33E+06	2.00E+06	5.32E+06	1.03E+04
7	7.50E-02	-	1.00E-01	4.68E+08	1.06E+08	4.01E+06	1.58E+07	1.01E+08	6.08E+07	3.35E+07	1.43E+06
8	1.00E-01	-	1.50E-01	5.76E+08	3.30E+08	1.80E+07	1.94E+07	5.55E+07	3.34E+07	2.03E+07	3.44E+04
9	1.50E-01	-	2.00E-01	8.11E+08	1.62E+08	5.65E+06	1.06E+07	2.61E+08	1.57E+08	5.39E+07	1.76E+05
10	2.00E-01	-	2.60E-01	2.98E+08	1.53E+08	8.14E+06	1.07E+07	2.28E+07	1.37E+07	1.14E+07	1.95E+04
11	2.60E-01	-	3.00E-01	5.07E+08	1.85E+08	8.81E+06	3.98E+06	2.02E+08	1.21E+08	3.25E+07	3.68E+06
12	3.00E-01	-	4.00E-01	2.44E+09	1.15E+09	5.82E+07	9.22E+07	1.04E+09	6.26E+08	1.27E+08	4.95E+07
13	4.00E-01	-	4.50E-01	3.63E+08	1.59E+08	7.98E+06	1.32E+08	4.03E+06	2.42E+06	3.41E+06	8.01E+04
14	4.50E-01	-	5.10E-01	2.00E+09	3.65E+08	1.17E+07	2.58E+06	1.61E+08	9.62E+07	3.25E+08	2.25E+05
15	5.10E-01	-	5.12E-01	7.37E+08	1.15E+08	2.27E+06	8.84E+09	4.21E+08	2.53E+08	3.35E+08	5.01E+04
16	5.12E-01	-	6.00E-01	4.75E+09	2.11E+09	1.06E+08	1.83E+06	2.25E+09	1.35E+09	5.31E+09	2.34E+06
17	6.00E-01	-	7.00E-01	1.47E+10	3.56E+09	1.43E+08	3.96E+09	1.33E+10	7.95E+09	3.36E+10	4.37E+06
18	7.00E-01	-	8.00E-01	1.49E+10	2.87E+09	9.69E+07	1.49E+09	8.78E+09	5.26E+09	2.10E+10	1.15E+06
19	8.00E-01	-	9.00E-01	7.56E+09	1.71E+09	6.27E+07	3.62E+10	3.18E+09	1.91E+09	2.38E+09	2.30E+05
20	9.00E-01	-	1.00E+00	1.22E+09	3.29E+08	1.39E+07	1.16E+09	2.08E+07	1.25E+07	2.71E+07	3.34E+04
21	1.00E+00	-	1.20E+00	4.80E+09	1.20E+09	4.86E+07	4.74E+09	1.35E+09	8.11E+08	8.80E+08	1.94E+05
22	1.20E+00	-	1.33E+00	1.11E+09	5.07E+08	2.58E+07	2.34E+09	3.32E+08	1.99E+08	8.35E+07	2.38E+05
23	1.33E+00	-	1.44E+00	5.32E+09	1.16E+09	4.36E+07	2.99E+09	3.19E+08	1.91E+08	6.97E+08	1.44E+04
24	1.44E+00	-	1.50E+00	9.60E+07	1.06E+08	6.19E+06	1.43E+08	2.94E+06	1.76E+06	3.33E+06	4.15E+04
25	1.50E+00	-	1.57E+00	6.47E+07	3.41E+07	1.79E+06	4.98E+08	8.52E+06	5.12E+06	1.18E+07	1.05E+04
26	1.57E+00	-	1.66E+00	1.90E+07	1.69E+07	9.62E+05	8.31E+05	3.98E+05	2.39E+05	8.91E+04	1.09E+03
27	1.66E+00	-	1.80E+00	1.41E+08	2.14E+08	1.27E+07	1.49E+08	8.71E+06	5.24E+06	5.64E+06	9.16E+04
28	1.80E+00	-	2.00E+00	3.35E+09	5.65E+08	1.67E+07	5.82E+05	5.10E+06	3.06E+06	3.18E+05	6.32E+03
29	2.00E+00	-	2.15E+00	1.34E+08	3.46E+07	1.43E+06	3.58E+04	4.26E+05	2.56E+05	2.44E+04	4.63E+03
30	2.15E+00	-	2.35E+00	9.43E+08	1.58E+08	4.65E+06	5.11E+04	2.41E+06	1.45E+06	1.44E+05	3.08E+03

Table 12.2-14a: Liquid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy Boundary (MeV)			LCW Collection Tank (photon/s)	HCW Collection Tank (photon/s)	Oil Separator (photon/s)	LCW GAC Unit (photon/s)	LCW TUF Unit (photon/s)	LCW RO Unit (photon/s)	LCW Cation Bed (photon/s)	LCW Anion Bed (photon/s)
31	2.35E+00	-	2.50E+00	2.19E+07	1.45E+07	7.96E+05	7.34E+02	1.42E+05	8.54E+04	5.19E+03	4.58E+03
32	2.50E+00	-	2.75E+00	1.02E+09	3.02E+08	1.36E+07	4.25E+01	9.05E+06	5.42E+06	5.44E+05	6.74E+01
33	2.75E+00	-	3.00E+00	1.28E+08	1.48E+08	8.88E+06	2.28E+01	6.67E+06	3.99E+06	3.99E+05	1.12E+01
34	3.00E+00	-	3.50E+00	1.02E+08	1.71E+07	5.01E+05	-	1.73E+05	1.04E+05	1.04E+04	3.49E+01
35	3.50E+00	-	4.00E+00	1.96E+07	4.13E+06	1.51E+05	-	3.96E+04	2.37E+04	2.34E+03	3.28E+01
36	4.00E+00	-	4.50E+00	3.91E+06	6.54E+05	1.92E+04	-	6.49E+03	3.89E+03	3.88E+02	1.24E+00
37	4.50E+00	-	5.00E+00	2.87E+07	4.55E+06	1.25E+05	-	3.94E+04	2.36E+04	2.36E+03	1.24E-06
38	5.00E+00	-	5.50E+00	5.41E+04	8.59E+03	2.36E+02	-	7.43E+01	4.46E+01	4.46E+00	1.12E-06
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				7.17E+10	1.86E+10	7.66E+08	6.30E+10	3.26E+10	1.95E+10	6.62E+10	6.73E+07

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-14b: Liquid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Energy Boundary (MeV)	LCW Mixed Bed (photon/s)	LCW Cesium Bed (photon/s)	LCW Sample Tank (photon/s)	HCW GAC Unit (photon/s)	HCW TUF Unit (photon/s)	HCW RO Unit (photon/s)	HCW Sample Tank (photon/s)	Drum Dryer (photon/s)
1	1.00E-02 - 2.00E-02	6.00E+06	5.15E+06	2.21E+06	8.61E+06	1.85E+07	1.11E+07	1.41E+06	3.30E+09
2	2.00E-02 - 3.00E-02	3.38E+06	2.67E+06	5.46E+04	1.25E+07	4.28E+07	2.57E+07	2.05E+06	1.93E+09
3	3.00E-02 - 4.50E-02	6.45E+07	5.80E+07	1.99E+05	3.65E+06	9.85E+07	5.91E+07	6.18E+06	3.60E+10
4	4.50E-02 - 6.00E-02	1.62E+06	1.37E+06	1.62E+04	2.64E+06	7.18E+06	4.30E+06	3.23E+05	8.90E+08
5	6.00E-02 - 7.00E-02	1.51E+06	1.32E+06	2.97E+04	1.24E+06	1.45E+07	8.69E+06	8.37E+05	8.32E+08
6	7.00E-02 - 7.50E-02	3.06E+05	2.57E+05	2.07E+03	5.27E+05	9.14E+05	5.49E+05	3.15E+04	1.66E+08
7	7.50E-02 - 1.00E-01	1.91E+06	1.65E+06	3.00E+04	2.05E+06	2.62E+07	1.57E+07	1.51E+06	1.12E+09
8	1.00E-01 - 1.50E-01	1.34E+06	8.27E+05	3.68E+03	2.76E+06	6.91E+07	4.15E+07	2.65E+06	7.45E+08
9	1.50E-01 - 2.00E-01	3.04E+06	2.66E+06	7.84E+04	1.36E+06	4.74E+07	2.84E+07	2.74E+06	1.69E+09
10	2.00E-01 - 2.60E-01	7.20E+05	4.96E+05	1.80E+03	1.75E+06	2.69E+07	1.61E+07	1.10E+06	4.01E+08
11	2.60E-01 - 3.00E-01	1.86E+06	1.62E+06	5.33E+04	6.05E+05	5.87E+07	3.52E+07	3.33E+06	1.20E+09
12	3.00E-01 - 4.00E-01	7.73E+06	6.17E+06	2.06E+05	1.87E+07	4.96E+08	2.97E+08	2.84E+07	6.43E+09
13	4.00E-01 - 4.50E-01	3.59E+05	1.83E+04	4.22E+01	2.70E+07	4.40E+06	2.64E+06	4.41E+04	1.18E+08
14	4.50E-01 - 5.10E-01	1.81E+07	1.63E+07	5.00E+04	4.42E+05	2.99E+07	1.79E+07	1.80E+06	1.01E+10
15	5.10E-01 - 5.12E-01	3.69E+07	3.35E+05	2.15E+02	1.08E+09	5.24E+07	3.14E+07	2.69E+05	1.00E+10
16	5.12E-01 - 6.00E-01	2.95E+08	2.66E+08	7.36E+05	3.58E+05	4.80E+08	2.88E+08	2.42E+07	1.65E+11
17	6.00E-01 - 7.00E-01	1.87E+09	1.68E+09	4.38E+06	8.12E+08	2.18E+09	1.31E+09	1.43E+08	1.04E+12
18	7.00E-01 - 8.00E-01	1.17E+09	1.05E+09	2.90E+06	3.05E+08	1.44E+09	8.65E+08	9.42E+07	6.51E+11
19	8.00E-01 - 9.00E-01	2.00E+08	5.85E+07	5.43E+05	5.08E+09	4.71E+08	2.83E+08	1.82E+07	7.30E+10
20	9.00E-01 - 1.00E+00	2.97E+06	3.19E+04	1.79E+01	2.37E+08	1.11E+07	6.64E+06	2.49E+05	8.84E+08
21	1.00E+00 - 1.20E+00	5.27E+07	4.06E+07	4.08E+05	9.67E+08	2.31E+08	1.39E+08	1.33E+07	2.74E+10
22	1.20E+00 - 1.33E+00	6.15E+06	2.82E+06	9.59E+04	4.78E+08	6.86E+07	4.11E+07	3.20E+06	2.65E+09
23	1.33E+00 - 1.44E+00	4.12E+07	3.28E+07	9.06E+04	6.11E+08	7.38E+07	4.43E+07	3.04E+06	2.17E+10
24	1.44E+00 - 1.50E+00	3.66E+05	3.84E+03	1.39E+00	2.93E+07	3.57E+06	2.14E+06	2.14E+04	1.13E+08
25	1.50E+00 - 1.57E+00	1.28E+06	2.81E+04	5.62E+02	1.02E+08	2.30E+06	1.38E+06	1.90E+04	3.81E+08
26	1.57E+00 - 1.66E+00	9.72E+03	1.78E+02	2.39E+00	1.70E+05	5.80E+05	3.49E+05	3.54E+04	5.41E+06
27	1.66E+00 - 1.80E+00	6.22E+05	6.05E+03	4.04E+00	1.83E+07	7.08E+06	4.25E+06	1.20E+04	1.81E+08
28	1.80E+00 - 2.00E+00	2.00E+04	1.39E+04	1.75E+00	1.19E+05	1.76E+06	1.06E+06	2.39E+04	1.17E+07
29	2.00E+00 - 2.15E+00	2.09E+03	5.99E+02	6.73E-01	7.50E+03	5.82E+05	3.49E+05	1.00E+04	1.79E+06
30	2.15E+00 - 2.35E+00	8.44E+03	6.87E+03	3.20E-01	1.04E+04	6.96E+05	4.18E+05	4.76E+03	5.12E+06
31	2.35E+00 - 2.50E+00	5.06E+02	1.04E+02	1.77E-01	1.51E+02	3.67E+05	2.20E+05	2.63E+03	8.70E+05

Table 12.2-14b: Liquid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy Boundary (MeV)			LCW Mixed Bed (photon/s)	LCW Cesium Bed (photon/s)	LCW Sample Tank (photon/s)	HCW GAC Unit (photon/s)	HCW TUF Unit (photon/s)	HCW RO Unit (photon/s)	HCW Sample Tank (photon/s)	Drum Dryer (photon/s)
32	2.50E+00	-	2.75E+00	5.41E+04	5.77E+03	1.15E+00	8.81E+00	1.16E+07	6.96E+06	1.73E+04	3.32E+07
33	2.75E+00	-	3.00E+00	4.39E+04	5.52E+02	9.43E-01	4.73E+00	1.03E+07	6.18E+06	1.41E+04	2.72E+07
34	3.00E+00	-	3.50E+00	5.79E+02	5.17E+02	6.74E-04	-	3.10E+04	1.86E+04	9.98E+00	3.27E+05
35	3.50E+00	-	4.00E+00	1.65E+02	8.62E+01	1.50E-03	-	2.33E+04	1.40E+04	2.25E+01	1.01E+05
36	4.00E+00	-	4.50E+00	2.20E+01	1.91E+01	1.65E-05	-	1.31E+03	7.86E+02	2.47E-01	1.25E+04
37	4.50E+00	-	5.00E+00	1.31E+02	1.18E+02	-	-	6.49E+03	3.89E+03	-	7.31E+04
38	5.00E+00	-	5.50E+00	2.48E-01	2.23E-01	-	-	1.23E+01	7.35E+00	-	1.38E+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	-	-	-	-	-	-	-	-
45	1.00E+01	-	1.20E+01	-	-	-	-	-	-	-	-
46	1.20E+01	-	1.40E+01	-	-	-	-	-	-	-	-
47	1.40E+01	-	2.00E+01	-	-	-	-	-	-	-	-
Total				3.79E+09	3.22E+09	1.21E+07	9.81E+09	5.99E+09	3.59E+09	3.52E+08	2.06E+12

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

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

Model Parameter	Value
GRWS guard bed	
Contents	
Geometry	vertical cylinder
Source dimensions of vessel	diameter=1.79′; height=47.5″
Shield thickness of steel shell	0.25″
GRWS decay bed (four each per train; two trains)	
Inputs	
Geometry	vertical cylinder
Source dimensions of vessel	diameter=1.79′; height=15′
Shield thickness of steel shell	0.25″

Table 12.2-16: Gaseous Radioactive Waste System Component Source Term Radionuclide Content

Isotope	Guard Bed (Ci)	Decay Bed 5A (Ci)	Decay Bed 6A (Ci)	Decay Bed 7A (Ci)	Decay Bed 8A (Ci)
Kr83m	1.37E-03	1.93E-03	2.81E-05	6.06E-07	3.12E-08
Kr85m	7.81E-03	1.67E-02	2.86E-03	4.91E-04	8.41E-05
Kr85	2.93E+00	1.06E+01	1.06E+01	1.05E+01	1.05E+01
Kr87	2.57E-03	3.12E-03	6.24E-06	1.25E-08	2.50E-11
Kr88	1.09E-02	1.91E-02	1.18E-03	7.31E-05	4.52E-06
Kr89	8.59E-06	8.59E-06	-	-	-
Xe131m	8.22E-01	2.37E+00	1.23E+00	6.36E-01	3.29E-01
Xe133m	5.23E-01	8.10E-01	2.31E-02	6.58E-04	1.90E-05
Xe133	5.05E+01	1.16E+02	2.67E+01	6.04E+00	1.37E+00
Xe135m	4.22E-04	4.22E-04	3.72E-05	3.72E-06	3.72E-07
Xe135	3.64E-01	3.66E-01	6.43E-04	5.94E-05	5.94E-06
Xe137	3.34E-05	3.34E-05	-	-	-
Xe138	4.22E-04	4.22E-04	-	-	-
Br82	5.56E-06	5.56E-06	5.56E-07	5.56E-08	5.56E-09
Br83	2.16E-06	2.16E-06	2.16E-07	2.16E-08	2.16E-09
Br84	2.22E-07	2.22E-07	2.22E-08	2.22E-09	2.22E-10
Br85	2.45E-09	2.45E-09	2.45E-10	2.45E-11	2.45E-12
I129	1.42E-06	1.42E-06	1.42E-07	1.42E-08	1.42E-09
I130	1.57E-05	1.57E-05	1.57E-06	1.57E-07	1.57E-08
I131	6.30E-03	6.30E-03	6.30E-04	6.30E-05	6.30E-06
I132	3.43E-05	3.43E-05	3.43E-06	3.43E-07	3.43E-08
I133	1.03E-03	1.03E-03	1.03E-04	1.03E-05	1.03E-06
I134	7.68E-06	7.68E-06	7.68E-07	7.68E-08	7.68E-09
I135	2.04E-04	2.04E-04	2.04E-05	2.04E-06	2.04E-07
Rb88	1.09E-02	1.91E-02	1.18E-03	7.31E-05	4.52E-06
Rb89	8.59E-06	8.59E-06	-	-	-
Cs137	2.50E-05	2.50E-05	-	-	-
Cs138	4.22E-04	4.22E-04	-	-	-
Sr89	8.59E-06	8.59E-06	-	-	-
Ba137m	2.36E-05	2.36E-05	-	-	-
Ar41	1.37E-02	3.96E-02	2.06E-02	1.07E-02	5.55E-03

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-17: Gaseous Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Energy Boundary (MeV)	Guard Bed (photons/s)	Decay Bed 5A (photons/s)	Decay Bed 6A (photons/s)	Decay Bed 7A (photons/s)	Decay Bed 8A (photons/s)
1	1.00E-02 - 2.00E-02	3.46E+09	8.42E+09	3.23E+09	2.06E+09	1.80E+09
2	2.00E-02 - 3.00E-02	1.98E+10	4.67E+10	1.83E+10	9.51E+09	5.30E+09
3	3.00E-02 - 4.50E-02	7.84E+11	1.80E+12	4.19E+11	9.79E+10	2.40E+10
4	4.50E-02 - 6.00E-02	7.64E+08	1.90E+09	8.07E+08	5.61E+08	5.05E+08
5	6.00E-02 - 7.00E-02	3.25E+08	8.12E+08	3.56E+08	2.54E+08	2.31E+08
6	7.00E-02 - 7.50E-02	1.32E+08	3.30E+08	1.47E+08	1.06E+08	9.69E+07
7	7.50E-02 - 1.00E-01	6.62E+11	1.52E+12	3.50E+11	7.95E+10	1.82E+10
8	1.00E-01 - 1.50E-01	4.50E+08	1.10E+09	4.76E+08	3.51E+08	3.25E+08
9	1.50E-01 - 2.00E-01	2.07E+09	4.99E+09	1.68E+09	7.51E+08	4.29E+08
10	2.00E-01 - 2.60E-01	1.52E+10	1.65E+10	1.97E+08	8.43E+07	7.84E+07
11	2.60E-01 - 3.00E-01	8.28E+07	1.76E+08	6.32E+07	4.04E+07	3.53E+07
12	3.00E-01 - 4.00E-01	3.92E+08	5.96E+08	1.30E+08	6.26E+07	5.06E+07
13	4.00E-01 - 4.50E-01	1.01E+08	1.17E+08	7.60E+06	7.11E+06	7.05E+06
14	4.50E-01 - 5.10E-01	1.32E+07	1.98E+07	4.42E+06	3.77E+06	3.70E+06
15	5.10E-01 - 5.12E-01	2.37E+08	8.49E+08	8.48E+08	8.48E+08	8.48E+08
16	5.12E-01 - 6.00E-01	2.67E+08	8.30E+08	7.84E+08	7.80E+08	7.80E+08
17	6.00E-01 - 7.00E-01	3.97E+08	4.02E+08	3.37E+06	6.93E+05	4.30E+05
18	7.00E-01 - 8.00E-01	1.90E+07	2.27E+07	1.23E+06	1.16E+05	1.74E+04
19	8.00E-01 - 9.00E-01	1.09E+08	1.77E+08	9.87E+06	6.29E+05	4.31E+04
20	9.00E-01 - 1.00E+00	3.75E+07	6.48E+07	3.99E+06	2.52E+05	1.69E+04
21	1.00E+00 - 1.20E+00	2.67E+07	3.92E+07	2.10E+06	1.44E+05	1.09E+04
22	1.20E+00 - 1.33E+00	5.27E+08	1.51E+09	7.74E+08	4.01E+08	2.08E+08
23	1.33E+00 - 1.44E+00	2.53E+07	3.42E+07	1.27E+06	7.96E+04	5.10E+03
24	1.44E+00 - 1.50E+00	1.56E+06	1.99E+06	1.34E+05	1.11E+04	1.00E+03
25	1.50E+00 - 1.57E+00	5.33E+07	9.29E+07	5.72E+06	3.55E+05	2.20E+04
26	1.57E+00 - 1.66E+00	2.84E+06	4.76E+06	2.71E+05	1.69E+04	1.10E+03
27	1.66E+00 - 1.80E+00	1.13E+07	1.58E+07	1.06E+06	2.46E+05	1.06E+05
28	1.80E+00 - 2.00E+00	9.06E+07	1.58E+08	9.75E+06	6.04E+05	3.74E+04
29	2.00E+00 - 2.15E+00	4.09E+07	6.79E+07	3.82E+06	2.37E+05	1.47E+04
30	2.15E+00 - 2.35E+00	7.12E+07	1.23E+08	7.39E+06	4.58E+05	2.83E+04
31	2.35E+00 - 2.50E+00	1.40E+08	2.46E+08	1.52E+07	9.39E+05	5.81E+04
32	2.50E+00 - 2.75E+00	2.60E+07	3.80E+07	1.38E+06	8.39E+04	5.18E+03
33	2.75E+00 - 3.00E+00	1.66E+06	2.69E+06	1.37E+05	8.44E+03	5.22E+02
34	3.00E+00 - 3.50E+00	2.43E+06	3.91E+06	1.99E+05	1.23E+04	7.58E+02
35	3.50E+00 - 4.00E+00	3.62E+05	6.08E+05	3.51E+04	2.17E+03	1.35E+02
36	4.00E+00 - 4.50E+00	6.84E+04	1.16E+05	6.83E+03	4.23E+02	2.62E+01
37	4.50E+00 - 5.00E+00	7.66E+05	1.34E+06	8.29E+04	5.14E+03	3.18E+02
38	5.00E+00 - 5.50E+00	1.45E+03	2.53E+03	1.57E+02	9.69E+00	5.99E-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 12.2-17: Gaseous Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy Boundary (MeV)	Guard Bed (photons/s)	Decay Bed 5A (photons/s)	Decay Bed 6A (photons/s)	Decay Bed 7A (photons/s)	Decay Bed 8A (photons/s)
45	1.00E+01 - 1.20E+01	-	-	-	-	-
46	1.20E+01 - 1.40E+01	-	-	-	-	-
47	1.40E+01 - 2.00E+01	-	-	-	-	-
Total		1.49E+12	3.41E+12	7.97E+11	1.93E+11	5.29E+10

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-18: Solid Radioactive Waste System Source Term Inputs

Model Parameter	Value
Spent resin storage tank:	
Contents	spent resins from CVCS and PCUS
Geometry	vertical cylinder
Source dimensions of vessel	diameter=12.0′; height=7.94′
Shield thickness of steel shell	0.25″
Phase separator tank:	
Inputs	spent resins from LRWS
Geometry	vertical cylinder
Source dimensions of vessel	diameter=10.0′; height=1.36′
Shield thickness of steel shell	0.25″
High Integrity Container (HIC):	
Inputs	spent resins from SRST
Geometry	vertical cylinder
Source dimensions of container	diameter=4.92' height=5.83'
Array of HICs	one layer of five Class B/C HICs

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

Isotope	SRST (Ci)	PST (Ci)	HIC (Ci)
Kr83m	1.75E-10	-	-
Kr85m	3.98E-11	-	-
Kr85	4.57E-08	1.48E-19	3.37E-09
Xe131m	2.60E-01	2.03E-14	1.03E-19
Xe133m	9.05E-03	-	-
Xe133	7.41E-01	-	-
Xe135m	1.18E-05	-	-
Xe135	7.85E-04	-	-
Br82	4.86E-04	3.22E-07	-
Br83	7.99E-11	1.23E-07	-
Br84	-	1.27E-08	-
Br85	-	1.40E-10	-
I129	3.39E-03	2.81E-09	2.82E-04
I130	9.24E-05	8.99E-07	-
I131	1.14E+01	1.59E-03	-
I132	8.18E-02	4.28E-05	-
I133	8.91E-02	7.42E-05	-
I134	2.43E-08	4.82E-07	-
I135	6.92E-05	1.16E-05	-
Rb86m	1.48E-14	2.10E-12	-
Rb86	1.90E-01	3.30E-04	9.83E-14
Rb88	4.58E-06	3.71E-05	-
Rb89	1.53E-07	1.45E-06	-
Cs132	4.09E-04	2.21E-06	-
Cs134	1.14E+04	1.13E+00	5.01E+02
Cs135m	3.12E-08	8.49E-08	-
Cs136	3.40E+00	8.51E-03	3.78E-17
Cs137	1.22E+04	9.26E-01	9.69E+02
Cs138	5.93E-06	2.56E-05	-
P32	4.11E-07	3.49E-11	7.72E-23
Co57	8.56E-07	4.19E-12	1.21E-08
Sr89	2.48E-01	6.97E-06	1.52E-06
Sr90	5.26E+00	1.21E-05	4.19E-01
Sr91	3.91E-07	2.24E-08	-
Sr92	3.07E-09	3.31E-09	-
Y90	5.26E+00	1.21E-05	4.19E-01
Y91m	2.49E-07	1.41E-08	-
Y91	4.85E-02	9.37E-07	1.09E-06
Y92	1.21E-08	7.05E-09	-
Y93	1.08E-07	5.06E-09	-
Zr97	1.26E-06	1.23E-08	-
Nb95	2.70E+00	2.85E-04	2.08E-04
Mo99	1.36E-01	8.74E-05	-
Mo101	1.02E-09	1.21E-08	-

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

Isotope	SRST (Ci)	PST (Ci)	HIC (Ci)
Tc99m	1.31E-01	8.40E-05	-
Tc99	2.06E-01	4.64E-07	1.72E-02
Ru103	4.12E-02	1.20E-06	1.64E-08
Ru105	3.11E-09	1.83E-09	-
Ru106	1.35E+00	5.50E-06	3.11E-02
Rh103m	4.08E-02	1.19E-06	1.62E-08
Rh105	1.86E-05	3.31E-08	-
Rh106	1.35E+00	5.50E-06	3.11E-02
Ag110	7.12E-01	4.39E-05	8.65E-03
Sb124	1.45E-04	2.71E-09	4.08E-09
Sb125	5.55E-02	1.58E-07	2.87E-03
Sb127	1.65E-05	6.60E-09	-
Sb129	6.40E-10	3.82E-10	-
Te125m	1.83E-01	3.38E-06	7.07E-04
Te127m	1.95E+00	2.03E-05	1.97E-03
Te127	1.91E+00	2.01E-05	1.93E-03
Te129m	5.29E-01	1.81E-05	2.67E-08
Te129	3.34E-01	1.14E-05	1.69E-08
Te131m	8.85E-04	2.18E-06	-
Te131	1.99E-04	4.98E-07	-
Te132	7.93E-02	4.09E-05	-
Te133m	1.33E-08	4.14E-08	-
Te134	1.08E-08	4.45E-08	-
Ba137m	1.15E+04	8.74E-01	9.15E+02
Ba139	7.92E-10	1.64E-09	-
Ba140	2.09E-02	2.01E-06	7.24E-20
La140	2.40E-02	2.09E-06	8.33E-20
La141	2.05E-09	1.44E-09	-
La142	1.41E-10	2.67E-10	-
Ce141	2.23E-02	7.90E-07	6.98E-10
Ce143	1.23E-05	2.51E-08	-
Ce144	1.02E+00	4.83E-06	1.57E-02
Pr143	3.58E-03	3.18E-07	1.20E-19
Pr144	1.01E+00	4.78E-06	1.55E-02
Np239	1.12E-03	9.04E-07	-
Na24	1.30E-03	2.49E-05	-
Cr51	5.04E+00	9.47E-04	1.21E-08
Mn54	8.33E+01	4.59E-03	1.49E+00
Fe55	1.44E+02	5.46E-03	7.42E+00
Fe59	9.42E-01	1.49E-04	1.58E-06
Co58	1.53E+02	3.47E-02	1.43E-02
Co60	7.85E+01	2.71E-03	5.10E+00
Ni63	4.92E+01	1.53E-03	4.05E+00
Zn65	1.99E+01	1.24E-03	2.32E-01
Zr95	2.10E+00	2.82E-04	9.54E-05

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

Isotope	SRST (Ci)	PST (Ci)	HIC (Ci)
Ag110m	5.24E+01	3.22E-03	6.36E-01
W187	3.21E-02	9.97E-06	-
Total	**3.56E+04**	**2.99E+00**	**2.40E+03**

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-20: Solid Radioactive Waste System Component Source Terms - Source Strengths

Energy Group	Energy Boundary (MeV)	Spent Resin Storage Tank (photon/s)	Phase Separator Tank (photon/s)	High Integrity Container (HIC) (photon/s)
1	1.00E-02 - 2.00E-02	2.41E+12	2.11E+08	1.51E+11
2	2.00E-02 - 3.00E-02	1.33E+12	1.14E+08	7.86E+10
3	3.00E-02 - 4.50E-02	3.11E+13	2.46E+09	2.35E+12
4	4.50E-02 - 6.00E-02	6.40E+11	5.64E+07	3.99E+10
5	6.00E-02 - 7.00E-02	2.94E+11	4.08E+07	1.79E+10
6	7.00E-02 - 7.50E-02	1.20E+11	1.06E+07	7.45E+09
7	7.50E-02 - 1.00E-01	4.56E+11	5.53E+07	2.66E+10
8	1.00E-01 - 1.50E-01	3.83E+11	3.70E+07	2.29E+10
9	1.50E-01 - 2.00E-01	2.09E+11	6.70E+07	1.14E+10
10	2.00E-01 - 2.60E-01	2.05E+11	2.08E+07	1.00E+10
11	2.60E-01 - 3.00E-01	7.66E+10	4.11E+07	2.20E+09
12	3.00E-01 - 4.00E-01	5.40E+11	1.95E+08	5.41E+09
13	4.00E-01 - 4.50E-01	8.24E+10	5.31E+06	1.34E+09
14	4.50E-01 - 5.10E-01	6.18E+12	6.16E+08	2.71E+11
15	5.10E-01 - 5.12E-01	1.72E+12	3.84E+08	6.37E+08
16	5.12E-01 - 6.00E-01	1.02E+14	1.01E+10	4.48E+12
17	6.00E-01 - 7.00E-01	7.75E+14	6.77E+10	4.79E+13
18	7.00E-01 - 8.00E-01	4.03E+14	3.99E+10	1.77E+13
19	8.00E-01 - 9.00E-01	2.74E+13	3.48E+09	8.33E+11
20	9.00E-01 - 1.00E+00	6.57E+11	4.08E+07	7.98E+09
21	1.00E+00 - 1.20E+00	1.55E+13	1.53E+09	7.31E+11
22	1.20E+00 - 1.33E+00	1.57E+12	1.17E+08	9.93E+10
23	1.33E+00 - 1.44E+00	1.44E+13	1.32E+09	6.48E+11
24	1.44E+00 - 1.50E+00	8.13E+10	5.07E+06	9.87E+08
25	1.50E+00 - 1.57E+00	2.83E+11	1.78E+07	3.44E+09
26	1.57E+00 - 1.66E+00	1.32E+09	1.08E+05	6.12E+06
27	1.66E+00 - 1.80E+00	2.87E+10	6.56E+06	7.63E+06
28	1.80E+00 - 2.00E+00	4.30E+08	3.63E+05	4.97E+06
29	2.00E+00 - 2.15E+00	6.65E+07	3.20E+04	7.38E+05
30	2.15E+00 - 2.35E+00	3.11E+08	1.63E+05	6.28E+06
31	2.35E+00 - 2.50E+00	3.95E+07	1.04E+04	6.71E+05
32	2.50E+00 - 2.75E+00	6.31E+07	6.04E+05	1.39E+05
33	2.75E+00 - 3.00E+00	2.61E+07	4.44E+05	5.64E+04
34	3.00E+00 - 3.50E+00	1.12E+06	1.15E+04	2.00E+04
35	3.50E+00 - 4.00E+00	3.73E+04	2.63E+03	8.31E+00
36	4.00E+00 - 4.50E+00	4.71E+02	4.30E+02	-
37	4.50E+00 - 5.00E+00	3.22E+02	2.61E+03	-
38	5.00E+00 - 5.50E+00	6.07E-01	4.92E+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 12.2-20: Solid Radioactive Waste System Component Source Terms - Source Strengths (Continued)

Energy Group	Energy Boundary (MeV)	Spent Resin Storage Tank (photon/s)	Phase Separator Tank (photon/s)	High Integrity Container (HIC) (photon/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		1.39E+15	1.29E+11	7.54E+13

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-21: Spent Fuel Gamma Source Strength

Gamma Energy Boundaries (MeV)			Single Spent Fuel Assembly Gamma Source (photons/sec)	Gamma Source for Full Fuel Storage Racks (1184) (photons/sec)
1.00E-02	-	2.00E-02	2.79E+17	3.30E+20
2.00E-02	-	3.00E-02	1.13E+17	1.34E+20
3.00E-02	-	4.50E-02	1.15E+17	1.36E+20
4.50E-02	-	6.00E-02	5.75E+16	6.80E+19
6.00E-02	-	7.00E-02	3.59E+16	4.25E+19
7.00E-02	-	7.50E-02	8.02E+16	9.50E+19
7.50E-02	-	1.00E-01	7.87E+16	9.31E+19
1.00E-01	-	1.50E-01	1.79E+17	2.12E+20
1.50E-01	-	2.00E-01	7.69E+16	9.11E+19
2.00E-01	-	2.60E-01	8.77E+16	1.04E+20
2.60E-01	-	3.00E-01	6.90E+16	8.17E+19
3.00E-01	-	4.00E-01	1.05E+17	1.24E+20
4.00E-01	-	4.50E-01	4.35E+16	5.16E+19
4.50E-01	-	5.10E-01	5.75E+16	6.81E+19
5.10E-01	-	5.12E-01	2.47E+15	2.93E+18
5.12E-01	-	6.00E-01	8.58E+16	1.02E+20
6.00E-01	-	7.00E-01	7.91E+16	9.36E+19
7.00E-01	-	8.00E-01	8.90E+16	1.05E+20
8.00E-01	-	9.00E-01	7.69E+16	9.11E+19
9.00E-01	-	1.00E+00	5.22E+16	6.18E+19
1.00E+00	-	1.20E+00	6.59E+16	7.81E+19
1.20E+00	-	1.33E+00	4.06E+16	4.81E+19
1.33E+00	-	1.44E+00	3.94E+16	4.67E+19
1.44E+00	-	1.50E+00	7.86E+15	9.30E+18
1.50E+00	-	1.57E+00	1.15E+16	1.36E+19
1.57E+00	-	1.66E+00	1.93E+16	2.28E+19
1.66E+00	-	1.80E+00	1.87E+16	2.22E+19
1.80E+00	-	2.00E+00	1.78E+16	2.10E+19
2.00E+00	-	2.15E+00	1.20E+16	1.42E+19
2.15E+00	-	2.35E+00	1.35E+16	1.59E+19
2.35E+00	-	2.50E+00	9.65E+15	1.14E+19
2.50E+00	-	2.75E+00	1.38E+16	1.63E+19
2.75E+00	-	3.00E+00	8.41E+15	9.96E+18
3.00E+00	-	3.50E+00	1.01E+16	1.19E+19
3.50E+00	-	4.00E+00	5.51E+15	6.52E+18
4.00E+00	-	4.50E+00	3.55E+15	4.20E+18
4.50E+00	-	5.00E+00	1.41E+15	1.67E+18
5.00E+00	-	5.50E+00	1.15E+15	1.36E+18
5.50E+00	-	6.00E+00	6.38E+14	7.55E+17
6.00E+00	-	6.50E+00	1.71E+14	2.02E+17
6.50E+00	-	7.00E+00	1.01E+13	1.19E+16
7.00E+00	-	7.50E+00	4.15E+12	4.92E+15
7.50E+00	-	8.00E+00	9.11E+10	1.08E+14
8.00E+00	-	1.00E+01	6.02E+10	7.13E+13
1.00E+01	-	1.20E+01	9.10E+06	1.08E+10

Table 12.2-21: Spent Fuel Gamma Source Strength (Continued)

Gamma Energy Boundaries (MeV)			Single Spent Fuel Assembly Gamma Source (photons/sec)	Gamma Source for Full Fuel Storage Racks (1184) (photons/sec)
1.20E+01	-	1.40E+01	1.08E+05	1.28E+08
1.40E+01	-	2.00E+01	1.87E+00	2.21E+03
Total			**2.06E+18**	**2.444E+21**

Table 12.2-22: Spent Fuel Neutron Energy Spectrum

Neutron Energy Boundaries (MeV)			Single Spent Fuel Assembly Neutron Source (neutrons/sec)	Neutron Source for Full Fuel Storage Racks (1184 assemblies) (neutrons/sec)
1.00E-11	-	1.00E-08	4.46E+07	5.28E+10
1.00E-08	-	3.00E-08	8.93E+07	1.06E+11
3.00E-08	-	5.00E-08	8.93E+07	1.06E+11
5.00E-08	-	1.00E-07	2.23E+08	2.64E+11
1.00E-07	-	2.25E-07	5.58E+08	6.61E+11
2.25E-07	-	3.25E-07	4.46E+08	5.29E+11
3.25E-07	-	4.14E-07	3.97E+08	4.70E+11
4.14E-07	-	8.00E-07	1.72E+09	2.04E+12
8.00E-07	-	1.00E-06	8.93E+08	1.06E+12
1.00E-06	-	1.13E-06	5.59E+08	6.62E+11
1.13E-06	-	1.30E-06	7.80E+08	9.23E+11
1.30E-06	-	1.86E-06	2.48E+09	2.94E+12
1.86E-06	-	3.06E-06	5.37E+09	6.36E+12
3.06E-06	-	1.07E-05	3.40E+10	4.03E+13
1.07E-05	-	2.90E-05	8.19E+10	9.70E+13
2.90E-05	-	1.01E-04	3.23E+11	3.82E+14
1.01E-04	-	5.83E-04	2.15E+12	2.55E+15
5.83E-04	-	3.04E-03	1.10E+13	1.30E+16
3.04E-03	-	1.50E-02	5.72E+13	6.78E+16
1.50E-02	-	1.11E-01	4.69E+14	5.55E+17
1.11E-01	-	4.08E-01	1.15E+15	1.36E+18
4.08E-01	-	9.07E-01	7.85E+14	9.29E+17
9.07E-01	-	1.42E+00	1.70E+14	2.02E+17
1.42E+00	-	1.83E+00	2.36E+13	2.79E+16
1.83E+00	-	3.01E+00	9.15E+12	1.08E+16
3.01E+00	-	6.38E+00	1.48E+12	1.76E+15
6.38E+00	+		3.94E+07	4.66E+10
		Total	**2.68E+15**	**3.168E+18**

Table 12.2-23: In-Core Instrument Source Term Input Assumptions

Component (Quantity)	Material
Emitter (4)	Rh-103
Signal wire (4)	Inconel 600
Insulation (1)	Al_2O_3
Outer sheath (1)	Inconel 600
Inner sheath (6)	Inconel 600
Thermocouple: type K chromel-alumel (2)	Chromel Alumel
Parameter	**Value**
Number of irradiation cycles	1-30
Neutron flux	2.30E+14 n/cm^2-sec

Table 12.2-24: In-Core Instrumentation Gamma Spectra

In-Core Instrumentation - Gamma Spectra (gamma/sec/assembly)				
Energy Group	Energy Boundaries (MeV)		Cycle 1	Cycle 30
			Discharge	3 Day Decay
1	2.00E-02	3.50E-02	2.06E+13	1.57E+11
2	3.50E-02	5.00E-02	9.86E+12	8.17E+10
3	5.00E-02	7.50E-02	1.22E+13	7.01E+10
4	7.50E-02	1.25E-01	1.14E+13	2.45E+11
5	1.25E-01	1.75E-01	4.27E+12	4.62E+10
6	1.75E-01	2.50E-01	3.70E+12	3.65E+10
7	2.50E-01	4.00E-01	6.22E+12	4.73E+11
8	4.00E-01	9.00E-01	3.20E+13	9.49E+12
9	9.00E-01	1.35E+00	2.88E+12	1.13E+13
10	1.35E+00	1.80E+00	2.68E+12	3.75E+10
11	1.80E+00	2.20E+00	3.31E+12	3.79E+08
12	2.20E+00	2.60E+00	1.27E+11	4.54E+06
13	2.60E+00	3.00E+00	1.18E+11	1.44E+08
14	3.00E+00	3.50E+00	2.10E+10	6.72E+06
15	3.50E+00	4.00E+00	9.09E+06	1.98E+05
16	4.00E+00	4.50E+00	2.59E+06	1.17E+03
17	4.50E+00	5.00E+00	9.23E+05	-
18	5.00E+00	1.00E+01	4.31E+08	-

Table 12.2-25: Control Rod Assembly Tip Source Term Input Assumptions

Parameter	Value
Neutron flux at control rod tip	1.40E+12 n/cm^2-sec
CRA tip irradiation duration	2 cycles
CRA Component	**Material**
Absorber	Ag-In-Cd
Stack support	X-750
Cladding	304 SS

Table 12.2-26: Control Rod Assembly Tip Gamma Spectra (End of Cycle 2)

Group	E_{low} (MeV)	E_{high} (MeV)	Discharge (γ/sec/kg)	3 Day Decay (γ/sec/kg)	30 Day Decay (γ/sec/kg)
1	2.00E-02	3.50E-02	2.08E+13	1.61E+11	1.23E+11
2	3.50E-02	5.00E-02	1.18E+13	3.10E+10	2.52E+10
3	5.00E-02	7.50E-02	1.11E+13	2.92E+10	2.37E+10
4	7.50E-02	1.25E-01	1.31E+13	3.08E+10	2.49E+10
5	1.25E-01	1.75E-01	5.99E+12	1.27E+10	1.03E+10
6	1.75E-01	2.50E-01	4.60E+12	3.84E+10	2.83E+10
7	2.50E-01	4.00E-01	7.01E+12	3.45E+10	1.87E+10
8	4.00E-01	9.00E-01	2.64E+13	1.02E+13	9.47E+12
9	9.00E-01	1.35E+00	3.81E+13	1.09E+12	1.01E+12
10	1.35E+00	1.80E+00	4.49E+12	1.47E+12	1.37E+12
11	1.80E+00	2.20E+00	3.78E+12	6.71E+08	6.01E+08
12	2.20E+00	2.60E+00	2.03E+10	2.77E+06	2.57E+06
13	2.60E+00	3.00E+00	1.85E+09	2.07E+05	6.72E+04
14	3.00E+00	3.50E+00	2.21E+08	1.98E+04	1.49E+04
15	3.50E+00	4.00E+00	3.23E+04	3.67E+02	2.12E+02
16	4.00E+00	4.50E+00	4.25E+03	1.05E+00	1.04E-13
17	4.50E+00	5.00E+00	5.40E+00	-	-
18	5.00E+00	1.00E+01	3.64E+00	-	-

Table 12.2-27: Secondary Neutron Source Gamma Spectra (End of Cycle 1)

Group	E_{low} (MeV)	E_{high} (MeV)	E_{avg} (MeV)	Discharge (γ/sec/assy)	3 Day Decay (γ/sec/assy)	30 Day Decay (γ/sec/assy)
1	2.00E-02	3.50E-02	2.75E-02	2.11E+15	9.11E+14	3.09E+14
2	3.50E-02	5.00E-02	4.25E-02	7.45E+14	4.12E+14	1.32E+14
3	5.00E-02	7.50E-02	6.25E-02	1.13E+15	3.75E+14	1.18E+14
4	7.50E-02	1.25E-01	1.00E-01	8.10E+14	4.17E+14	1.31E+14
5	1.25E-01	1.75E-01	1.50E-01	3.50E+14	2.18E+14	1.03E+14
6	1.75E-01	2.50E-01	2.13E-01	2.38E+14	1.29E+14	4.13E+13
7	2.50E-01	4.00E-01	3.25E-01	3.42E+14	1.96E+14	7.10E+13
8	4.00E-01	9.00E-01	6.50E-01	1.44E+16	9.74E+15	4.78E+15
9	9.00E-01	1.35E+00	1.13E+00	6.19E+14	5.04E+14	3.26E+14
10	1.35E+00	1.80E+00	1.58E+00	3.35E+15	3.16E+15	2.32E+15
11	1.80E+00	2.20E+00	2.00E+00	4.82E+14	3.38E+14	2.47E+14
12	2.20E+00	2.60E+00	2.40E+00	7.56E+12	2.32E+12	1.70E+12
13	2.60E+00	3.00E+00	2.80E+00	4.96E+12	3.30E+11	2.41E+11
14	3.00E+00	3.50E+00	3.25E+00	8.48E+11	2.28E+07	1.71E+07
15	3.50E+00	4.00E+00	3.75E+00	5.36E+09	4.00E+05	2.44E+05
16	4.00E+00	4.50E+00	4.25E+00	1.96E+09	9.77E+02	9.62E-11
17	4.50E+00	5.00E+00	4.75E+00	1.87E+09	-	-
18	5.00E+00	1.00E+01	7.50E+00	3.97E+09	-	-

Table 12.2-28: Post-Accident Equipment Qualification Source Term Input Assumptions

Parameter	Value
Containment release delay	0 hours
Containment release duration	1.0E-05 hours
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	6475 ft^3
Primary coolant water density	43.6 lb/ft^3
Air density	0.07 lb/ft^3
Containment air volume	3635 ft^3
Combined water volume	2500 ft^3

Table 12.2-29: Not Used

Table 12.2-30: Not Used

Table 12.2-31: Post-Accident Integrated Energy Deposition and Integrated Dose

Volume	Medium	Time	Integrated Dose (Rad)
Reactor and containment	Water	1 hour	2.73E+03
		36 hour	2.99E+04
		3 day	4.01E+04
		30 day	9.15E+04
		100 day	1.17E+05
Containment vessel	Air	1 hour	1.19E+06
		36 hour	1.35E+07
		3 day	1.84E+07
		30 day	4.85E+07
		100 day	7.78E+07
Bioshield envelope	Air and water	1 hour	7.35E+03
		36 hour	8.02E+04
		3 day	1.11E+05
		30 day	4.30E+05
		100 day	1.30E+06

Table 12.2-32: Input Parameters for Determining Facility Airborne Concentrations

Parameter	Value
Primary coolant leak rate	160 lb/day/unit
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)	 1 100 200 2000
Primary coolant source term	Table 11.1-4 Table 11.1-8
Pool water source term	Table 12.2-9 Table 12.2-10 Table 12.2-11
Pool evaporation rate: • Pool surface water temperature • Area of pool water surface • Air velocity over water surface • Room air temperature • Room air relative humidity	1705 lb/hour 100°F 11,845 ft^2 30 ft/min 85°F 60%
CVCS pump/valve room leak	4 lb/day
Degasifier room leak	13 lb/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/hour 2 air-changes/hour 2 air-changes/hour
Pool air space volume	4.42E+10 ml
CVCS pump/valve room volume	1.12E+08 ml
Degasifier room volume	3.52E+08 ml

Table 12.2-33: Reactor Building Airborne Concentrations

Radionuclide	CVCS Pump / Valve Room (µCi/ml)	Degasifier Room (µCi/ml)	Air Space above Reactor Pool (µCi/ml)
Noble Gases			
Kr83m	1.49E-09	1.49E-09	1.07E-13
Kr85m	6.92E-09	6.92E-09	-
Kr85	2.21E-06	2.21E-06	-
Kr87	3.20E-09	3.20E-09	-
Kr88	1.06E-08	1.06E-08	-
Xe131m	2.86E-08	2.86E-08	1.52E-08
Xe133m	2.61E-08	2.61E-08	2.36E-08
Xe133	1.95E-06	1.95E-06	3.35E-07
Xe135m	1.07E-09	1.07E-09	6.21E-09
Xe135	6.37E-08	6.37E-08	3.45E-09
Xe138	1.20E-09	1.20E-09	-
Halogens			
Br82	1.95E-13	1.95E-13	2.01E-16
Br83	9.82E-13	9.82E-13	-
Br84	3.16E-13	3.16E-13	-
I129	4.86E-18	4.86E-18	1.23E-20
I130	1.54E-12	1.54E-12	3.04E-16
I131	4.07E-11	4.07E-11	8.73E-12
I132	1.64E-11	1.64E-11	2.19E-14
I133	6.05E-11	6.05E-11	1.01E-12
I134	7.89E-12	7.89E-12	-
I135	3.68E-11	3.68E-11	7.67E-16
Cs, Rb			
Rb86	1.39E-13	1.39E-13	1.28E-14
Rb88	5.71E-09	5.71E-09	-
Rb89	2.11E-11	2.11E-11	-
Cs132	2.67E-15	2.67E-15	2.16E-16
Cs134	2.40E-11	2.40E-11	2.37E-12
Cs135m	1.30E-14	1.30E-14	-
Cs136	5.08E-12	5.08E-12	4.56E-13
Cs137	1.47E-11	1.47E-11	1.46E-12
Cs138	4.66E-10	4.66E-10	-
Other FP			
P32	3.97E-19	3.97E-19	1.85E-20
Co57	2.96E-21	2.96E-21	-
Sr89	2.37E-14	2.37E-14	8.94E-16
Sr90	3.98E-15	3.98E-15	2.04E-16
Sr91	8.88E-15	8.88E-15	1.72E-17
Sr92	4.35E-15	4.35E-15	-
Y90	9.82E-16	9.82E-16	1.10E-16
Y91m	5.00E-15	5.00E-15	1.10E-17
Y91	2.57E-15	2.57E-15	1.30E-16
Y92	4.19E-15	4.19E-15	1.17E-19
Y93	1.90E-15	1.90E-15	4.39E-18

Table 12.2-33: Reactor Building Airborne Concentrations (Continued)

Radionuclide	CVCS Pump / Valve Room (µCi/ml)	Degasifier Room (µCi/ml)	Air Space above Reactor Pool (µCi/ml)
Zr97	2.83E-15	2.83E-15	2.20E-17
Nb95	4.20E-15	4.20E-15	8.78E-14
Mo99	5.16E-12	5.16E-12	1.64E-13
Mo101	8.08E-14	8.08E-14	-
Tc99m	4.79E-12	4.79E-12	1.58E-13
Tc99	1.49E-16	1.49E-16	7.60E-18
Ru103	4.96E-15	4.96E-15	2.46E-16
Ru105	1.51E-15	1.51E-15	6.42E-20
Ru106	3.22E-15	3.22E-15	1.64E-16
Rh103m	4.90E-15	4.90E-15	2.43E-16
Rh105	3.45E-15	3.45E-15	7.69E-17
Sb124	7.31E-18	7.31E-18	3.66E-19
Sb125	6.44E-17	6.44E-17	3.29E-18
Sb127	2.78E-16	2.78E-16	1.01E-17
Sb129	3.18E-16	3.18E-16	1.27E-20
Te125m	9.46E-15	9.46E-15	4.73E-16
Te127m	3.05E-14	3.05E-14	1.54E-15
Te127	1.18E-13	1.18E-13	1.67E-15
Te129m	8.73E-14	8.73E-14	4.30E-15
Te129	1.08E-13	1.08E-13	2.71E-15
Te131m	2.82E-13	2.82E-13	5.05E-15
Te131	1.05E-13	1.05E-13	1.14E-15
Te132	2.07E-12	2.07E-12	7.03E-14
Te133m	1.29E-13	1.29E-13	-
Te134	1.68E-13	1.68E-13	-
Ba139	3.75E-15	3.75E-15	-
Ba140	2.56E-14	2.56E-14	1.18E-15
La140	7.58E-15	7.58E-15	8.48E-16
La141	1.34E-15	1.34E-15	2.22E-20
La142	5.66E-16	5.66E-16	-
Ce141	3.94E-15	3.94E-15	1.94E-16
Ce143	2.95E-15	2.95E-15	5.82E-17
Ce144	3.31E-15	3.31E-15	1.69E-16
Pr143	3.50E-15	3.50E-15	1.72E-16
Pr144	3.28E-15	3.28E-15	1.67E-16
Np239	6.22E-14	6.22E-14	1.82E-15
Crud			
Na24	6.42E-12	6.42E-12	3.99E-14
Cr51	3.74E-13	3.74E-13	1.83E-11
Mn54	1.93E-13	1.93E-13	9.81E-12
Fe55	1.44E-13	1.44E-13	7.37E-12
Fe59	3.61E-14	3.61E-14	1.79E-12
Co58	5.54E-13	5.54E-13	2.78E-10
Co60	6.37E-14	6.37E-14	3.26E-12
Ni63	3.18E-14	3.18E-14	1.63E-12
Zn65	6.13E-14	6.13E-14	3.12E-12

Table 12.2-33: Reactor Building Airborne Concentrations (Continued)

Radionuclide	CVCS Pump / Valve Room (µCi/ml)	Degasifier Room (µCi/ml)	Air Space above Reactor Pool (µCi/ml)
Zr95	4.69E-14	4.69E-14	2.35E-12
Ag110m	1.56E-13	1.56E-13	7.97E-12
W187	3.30E-13	3.30E-13	4.48E-12
Water Activation Products			
H3	4.04E-07	4.04E-07	1.46E-06
C14	3.21E-11	3.21E-11	6.34E-12
Ar41	6.28E-08	6.28E-08	-

Note: Assumes the plant consists of 12 NPMs operating on a two-year refueling cycle.

Table 12.2-34: Maximum Post-Accident Radionuclide Concentrations

Isotope	RCS Peak Concentration (μCi/g)
Kr-83m	2.69E+00
Kr-85m	2.14E-01
Kr-85	3.73E+01
Kr-87	6.52E-02
Kr-88	1.90E-01
Xe-131m	1.87E+00
Xe-133m	5.47E+00
Xe-133	2.07E+02
Xe-135m	1.73E+01
Xe-135	1.05E+02
Xe-137	1.39E-02
Xe-138	4.77E-02
Br-82	5.67E-01
Br-83	2.67E+00
Br-84	9.06E-01
Br-85	9.52E-02
I-130	4.45E+00
I-131	1.20E+02
I-132	4.60E+01
I-133	1.75E+02
I-134	2.09E+01
I-135	1.04E+02
Rb-88	3.31E-01
Rb-89	1.08E-02
Cs-134	1.84E-01
Cs-136	3.86E-02
Cs-137	1.27E-01
Cs-138	9.98E-02
Ni-63	4.41E-02
Sr-89	1.10E-02
Zr-95	6.51E-02
Nb-95	6.44E-02
Mo-99	3.88E-02
Tc-99m	7.01E-02
Ag-110m	2.17E-01
Te-132	1.56E-02
Ba-137m	2.26E-01
Na-24	9.11E+00
Cr-51	5.19E-01
Mn-54	2.67E-01
Fe-55	2.00E-01
Fe-59	5.01E-02
Co-58	7.68E+00
Co-60	8.83E-02
W-187	4.65E-01
Zn-65	8.50E-02

Table 12.2-34: Maximum Post-Accident Radionuclide Concentrations (Continued)

Isotope	RCS Peak Concentration (µCi/g)
H-3	9.90E-01
Ar-41	2.07E-01

12.3 Radiation Protection Design Features

Radiation protection design features incorporated into the design of the NuScale Power Plant facilities are described in this section. These include features to reduce both onsite and offsite exposures, and to protect the environment. These design features are incorporated into the facility 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 Power Plant design. These design features are incorporated during the design process and the application of industry 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

Radiation sources from sedimentation in tanks are reduced by using tanks with bottoms that slope toward outlets and, where practicable, providing built-in spray features, spargers and eductors for mixing tank contents.

The tanks in the liquid radioactive waste and solid radioactive waste systems are stainless-steel with sloped bottoms. The liquid radioactive waste 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.

Tanks that store spent resins have break pot tanks in the vent line to prevent the tank contents from contaminating the ventilation system.

The pool surge control system (PSCS) storage tank is located in a steel-lined catch basin with sufficient capacity to contain the tank volume along with its associated piping, with a weather enclosure to prevent precipitation from mixing with potentially contaminated fluids.

12.3.1.1.2 Valves

Remotely actuated valves are used to minimize personnel exposures, where practical. 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 solid radioactive waste system (SRWS) and the pool cleanup system (PCUS) 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 high radiation areas, in some applications.

Where possible, 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

For the liquid radioactive waste system (LRWS), piping is seamless stainless steel with large radius bends and butt welded to minimize crud traps. Piping used for slurry transfers associated with the solid radioactive waste system (SRWS) is also butt welded stainless steel pipe with five-diameter bends.

The LRWS and SRWS piping is provided with clean-in-place (CIP) and flushing capabilities to reduce the buildup of crud and other contaminates. The chemical and volume control system (CVCS) and the PCUS 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 practical. Whenever possible, 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 PSCS piping, are enclosed within structured pipe chases or are double-walled. Chemical and volume control system, PSS, and resin sluice pipes that are expected to contain highly contaminated fluids are routed through shielded pipe chases, as much as practical.

12.3.1.1.4 Pumps

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

The LRWS uses double diaphragm pumps to reduce leakage and minimize repair times.

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 prior to 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 possible, heat exchangers are designed such that the contaminated fluid is on the tube side.

12.3.1.1.6 Instrumentation

Whenever practical, 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 to prevent blowback in the event of an exhaust system trip.

More discussion on ventilation systems is provided in Section 12.3.3.

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

The filter design used 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 prior to 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 series of charcoal beds of the gaseous radioactive waste system are located in individual shielded cubicles.

The removal of activated charcoal from charcoal beds is done remotely to minimize the occupational exposures to plant personnel.

Radiation detectors monitor each charcoal bed cubicle 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. Radioactive 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 radioactive fluid is achieved by routing sample lines in shielded pipe chases to the extent practical and locating sample coolers behind the concrete and steel partition wall as necessary. 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 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 steam generator tubing due to 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 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 CRA hub connection couplings (Haynes Alloy-25). The cobalt content of austenitic stainless steel and Ni-Cr-Fe weld filler metals is limited to a maximum of 0.05 percent. The cobalt content of austenitic stainless steel base materials is limited to 0.15 percent. Steam generator (SG) heat transfer tubing is limited to a cobalt content of a maximum average of 0.014 percent, with zero heats exceeding 0.020 percent. Table 12.3-4 summarizes the typical cobalt content for materials and components.

12.3.1.2 Plant and Layout Design Features for ALARA

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 possible.

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 will be sufficiently 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 practical. 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 NuScale Power Plant is analyzed for expected radiation levels resulting from normal operation. Since 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.

Normal operation radiation zones for the RXB are provided in Figure 12.3-1a through Figure 12.3-1i. Areas that have the potential for airborne radiation in the RXB and the Radioactive Waste Building (RWB) are listed in Table 12.3-5a and Table 12.3-5b, respectively. Normal operation radiation zones for the RWB are provided in Figure 12.3-2a and Figure 12.3-2b. These 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 (RCA) 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 Annex Building. 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 will be 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: A COL applicant that references the NuScale Power Plant design certification will develop the administrative controls regarding access to high radiation areas per the guidance of Regulatory Guide 8.38.

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. The locations of very high-radiation areas are listed on Table 12.3-3.

COL Item 12.3-2: A COL applicant that references the NuScale Power Plant design certification 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: A COL applicant that references the NuScale Power Plant design certification will specify personnel exposure monitoring hardware, specify contamination identification and removal hardware, and establish administrative controls and procedures to control access into and exiting the radiologically controlled area.

12.3.1.3.2 Accident Conditions

Post-accident access is discussed in Section 12.4.1.8 and equipment qualification is addressed in Section 12.2.1.13 and Section 3.11. A radiation and shielding design review has been 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 prior to area entry.

See Section 7.1 for 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 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. The radiation zone maps described in Section 12.3.1 indicate the radiation levels for plant areas.

As described in Section 12.3.1, shielding design features include permanent shielding and separation of components that constitute substantial radiation sources, the use of shielded cubicles, labyrinths, and shielded entrances to minimize dose. 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 ANSI/ANS 6.4-2006 (Reference 12.3-1). Table 12.3-6 and Table 12.3-7 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.

For shield walls that contain a door, the door provides an equivalent radiation attenuation as the shield wall that contains the door. A listing of radiation shield doors is provided in Table 12.3-8 for the RXB and Table 12.3-9 for the RWB.

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 used 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 a Monte Carlo radiation transport code designed to track a variety of particles over a broad spectrum of energies. The MCNP6 code is used for shielding calculations and for dose rate determinations. ANSI/ANS 6.1.1-1977, "Gamma Flux to Dose Conversion Factors," (Reference 12.3-3) is used to convert gamma flux at each detector location to a corresponding dose rate.

Radioactive components in the RXB and RWB are modeled using MCNP6. The codes used to prepare source strength input data are described in Section 12.2. 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 which incorporate the full volume of the component.

Shielding credit and material selections for MCNP6 cells are conservatively applied. The material compositions for air, concrete, water, and stainless steel are taken from PNNL-25870 (Reference 12.3-4). Structural steel composition is in accordance with plant drawings and ASTM standards. Credit is not taken for reinforcing steel bars in the concrete.

The operating NuScale Power Module (NPM) dose rate at full power is also calculated using MCNP6. The reactor shielding calculations consider dose rates from fission neutrons, fission photons, and gamma output from buildup of radioisotopes in the reactor coolant. The NPM model is conservatively developed using methods similar to the building evaluations. The NPM model determines dose rates for components located below the bioshield.

The fission neutron and fission photon output is based on a total power output of 160 MWt and energy spectrums are based on MCNP6 and SCALE6.1 (Reference 12.3-11) fission neutron and fission gamma distributions. 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. Figure 12.3-3 shows the homogenized regions and the general arrangement of the NPM shielding model.

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 (see Figure 12.3-1a though Figure 12.3-1i). 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

An NPM is a self-contained nuclear steam supply system composed of a reactor core, a pressurizer, two steam generators integrated within the reactor pressure vessel, CRDMs and valves, and is housed in a compact steel containment vessel. The containment vessel is partially immersed in the reactor pool as shown in Figure 1.2-5.

Biological shielding is provided above each NPM to allow personnel access above the 126' elevation in the RXB. The bioshield provides shielding using concrete 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 containment vessel, 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.

COL Item 12.3-8: A COL applicant that references the NuScale Power Plant design certification will describe the radiation shielding design measures used to compensate for major shield wall penetrations in accordance with FSAR Section 12.1.2.3.2 "Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment," Section 12.3.1.2.3 "Penetrations," and Section 12.3.2.2 "Design Considerations." Penetration compensatory measures will account for the protection of equipment, and exposures to workers and the public.

12.3.2.4.2 Main Control Room

The dose rate in the main control room 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 15.0-12, 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 12.3-1i).

The RXB includes systems that contain radioactive components. The major radiation sources in the RXB are associated with the NPM (see Section 12.3.2.4.1), chemical volume and control system, PCUS, 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 regenerative and non-regenerative heat exchangers are located at elevation 50'-0". The module heatup system heat exchangers are located at elevation 62'-0". The CVCS reactor coolant particulate filters and the CVCS ion exchangers are located at elevation 24'-0". Access to these areas is restricted and is not required for normal operation of this equipment.

The filters, ion exchangers, and heat exchangers are located in shielded cubicles with knockout panels, which provide equivalent shielding as the wall in which they are located. The CVCS filters and resin traps are accessible via removable floor shield plugs at elevation 35'-8" for maintenance purposes. The cubicle walls are concrete supported by carbon steel plates, called structural steel partition walls. 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 ion exchanger spectra. 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 RCS discharge lines, which travel from the modules to the CVCS heat exchangers and purification equipment through a concrete-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.

Pool Cooling and Cleanup Systems

The pool cooling and cleanup systems, which include the spent fuel cooling system, reactor pool cooling system, and the PCUS are located below grade in the RXB. The PCUS demineralizers and filters are located at elevation 24'-0".

The PCUS demineralizers and filters are located in shielded cubicles. The dose rates in surrounding areas are acceptable for operations or maintenance activities. The filters are changed in accordance with ALARA principles to minimize personnel exposure.

For purposes of radiation shielding, the spent fuel pool cooling and reactor pool cooling heat exchangers are a negligible 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 degasifiers and their transfer pumps (both liquid and gaseous) are located within shielded cubicles in the RXB at elevation 24'-0". The degasifier contributes minor dose rates to the adjacent labyrinth and surrounding corridors and is acceptable for operations and maintenance activities.

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, refueling pool, 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 storage racks and the refueling pool 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 fuel storage racks in the spent fuel pool, as described in Section 9.1. Due to the depth of water above the stored fuel, the dose rate at the water's surface from the stored fuel is negligible. Likewise, 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 acceptably small.

12.3.2.4.4 Radioactive Waste Building

The RWB houses significant radiation sources that belong to the radioactive waste processing systems. The specifics of these systems are discussed below. The radiation zone maps are located in Figure 12.3-2a and Figure 12.3-2b.

Liquid Radioactive Waste System

The LRWS is primarily located in the RWB. The low-conductivity waste (LCW) and high-conductivity waste (HCW) sample tanks (two of each) contain liquid radioactive waste 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 at elevation 71'-0". The respective transfer pumps for these tanks are located in shared compartments in the RWB at elevation 71'-0". 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 other LRWS components that are important radiation sources are located in the RWB at elevation 100'-0". Liquid radioactive waste demineralizers and some of the filters 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, GAC filters, and drum dryer, as noted in Table 12.3-7.

Gaseous Radioactive Waste System

The GRWS system is located in the RWB at elevation 71'-0". GRWS components are generally located in separate, shielded compartments.

The redundant charcoal decay beds each consists of four charcoal vessels that share a shielded compartment, separated by a shield wall. The decay beds are protected by a single guard bed that is in a separately shielded compartment located in the RWB at elevation 71'-0".

The remaining GRW 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 at elevation 71'-0".

Solid Radioactive Waste System

The SRWS is located in the RWB at elevations elevation 71'-0" and elevation 100'-0". SRW 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 at elevation 71'-0". The respective transfer pumps for these tanks are located in shared compartments in the RWB at elevation 71'-0". 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 SRW consists of both Class A and Class B/C waste storage areas. A Class A waste package storage area is located at elevation 100'-0". The Class A high integrity container (HIC) and Class B/C HIC storage area is located at elevation 71'-0". Access to the HIC storage area is through floor shield plugs at elevation 100'-0".

Additional shielding is modeled for the HIC process shield, as noted in Table 12.3-7.

12.3.3 Ventilation

The plant heating, ventilating, and air-conditioning systems are designed to provide a controlled environment for personnel and equipment during normal operation. In areas subject to airborne activity, the ventilation systems are designed to collect, process, and exhaust airborne radioactive material, including directing airflow to processed exhausts (Section 9.4.) This section discusses the radiation control considerations of the HVAC systems design.

12.3.3.1 Design Objectives

Design objectives for the plant heating ventilation and air conditioning 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 building ventilation systems are designed to maintain a negative pressure with respect to the outside environs and create air flow inside the building from areas of low airborne potential to areas of higher airborne potential.

Other design features that are incorporated to 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 heating ventilation and air conditioning system is designed to allow rapid replacement of components. Filter-adsorber unit conformance 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 RXB is normally 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 HEPA filters and charcoal adsorbers. See Section 9.4.2 for additional details.

The dry dock area is provided with exhaust flow to entrain airborne contamination that may result from NPM components being exposed to air during maintenance activities.

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

In response to a high-radiation signal from the spent fuel exhaust ductwork, the RBVS will change into its high-radiological mode. In this mode, the spent fuel pool exhaust flow is diverted through both the HEPA filters and charcoal adsorbers. The general exhaust fans will reduce capacity and maintain the design exhaust airflows for the RWB and Annex Building. The RBVS supply will also reduce its capacity to provide ventilation air while maintaining a negative pressure in the RXB.

Adequate space for temporary shielding is provided to minimize personnel exposures during maintenance of ventilation equipment, including filters, inspection, and testing. In addition, the filter units are designed with 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 and is exhausted through the RBVS exhaust system. The main supply air handling unit contains both low and high efficiency outside air filters, a heating coil, and a chilled water cooling coil. Supply air from the main RWBVS is distributed throughout the RWB. Exhaust air is collected and conveyed to the RBVS general exhaust filter units 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. See Section 9.4.3 for 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 has been filtered (low and high efficiency) to maintain a suitable environment for personnel and equipment. The CRVS is designed to maintain a positive pressure inside the main control room (MCR) with respect to adjacent spaces. See Section 9.4.1 for 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 additional HEPA and charcoal filtration. Areas served by the CRVS (MCR and technical support center) are designed to maintain operator doses within PDC 19 limits.

If power is not available, or if a high radiation indication is received from the radiation monitors downstream of the CRVS filter unit, the control room envelope (CRE) isolation dampers close and the control room habitability system is initiated.

12.3.3.6 Control Room Habitability System

The control room habitability system uses a set of compressed air storage tanks to supply the CRE in case of an emergency. Upon receiving an initiation signal, the control room habitability system supplies the MCR control room envelope with clean air and maintains the CRE at a positive pressure with respect to adjacent areas. See Section 6.4 for additional details.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The following instrumentation is provided to monitor area radiation and airborne radioactivity within the facility.

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 are used to detect fuel cladding breach under potential core damaging accident conditions. The use of these monitors is discussed in Section 9.3.2.

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 prior to reaching predetermined thresholds for radiation. The CVCS reactor coolant ARMs are described in Section 11.5.

The ARMs located in the reactor pool area and 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. In addition, a local area radiation monitor 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 continuous air monitor (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.

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 as low as reasonably achievable 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 area radiation monitors 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 prior to 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. This ensures operator and worker awareness of changing radiological conditions that could indicate system leakage or component malfunction, and provides a warning to plant personnel prior to entry into the affected areas. Where appropriate, local visual alarms are provided outside of the monitored area to ensure worker awareness prior to 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. This ensures personnel working in the vicinity are able to determine easily the status of an area radiation monitor channel when in the vicinity of the local indication devices.

Fixed area radiation monitor 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 12.3-6). 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 practical, detectors are located to best measure the representative exposure rates within a given area or specified location.

The following criteria were 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 practical to provide easy maintenance access in an unobstructed area.

The fixed area radiation monitor 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 assure 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 fixed area radiation monitor 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 Electric Power Research Institute (EPRI) TR-102644 Revision 1, "Calibration of Radiation Monitors at Nuclear Power Plants" (Reference 12.3-7). 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 ARMs is performed to verify the operability of the channel, including alarm functions in accordance with manufacturer's requirements and using the guidance of EPRI TR-104862, Revision 2, "Area and Process Radiation Monitoring

System Guide" (Reference 12.3-8). 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.

Selected ARMs support accident condition response and are PAM system variables, as described in Table 12.3-10 and Table 7.1-7. The fixed area radiation monitors 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 are used to detect fuel damage under accident conditions, and are considered PAM system B and C variables. Two 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 497-2002 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" (Reference 12.3-9). 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 497-2002, Section 6.2, Common Cause Failure.

Fixed area radiation monitoring data is capable of being supplied to the NRC Operations Center through the ERDS via a secure direct electronic data link in the event of an emergency. The ERDS connection is discussed in Section 7.2.

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

- Fixed area radiation monitors that are classified as a Type B post-accident monitoring (PAM) variable receive power from the highly reliable DC power system.
- Fixed area radiation monitors that are classified as a Type C PAM variable receive power from the highly reliable DC power system.
- Fixed area radiation monitors that are classified as a Type E PAM variable receive power from the normal DC power system (EDNS).
- Fixed area radiation monitors that are not used for PAM variables receive power from the EDNS.

The quality and safety classification of the fixed area radiation monitors is provided in Table 3.2-1.

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

Table 12.3-12 provides information about the area radiation 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 monitors that serve a PAM function.

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

COL Item 12.3-5: A COL applicant that references the NuScale Power Plant design certification will describe design criteria for locating additional area radiation monitors.

12.3.4.3 Airborne Radioactivity Monitoring Instrumentation

The fixed continuous airborne radiation monitors (CAMs) 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 prior to 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. This ensures operator and worker awareness of changing radiological conditions that could indicate system leakage or component malfunction, and provides a warning to plant personnel prior to entry into the affected areas. Where appropriate, local visual alarms are provided outside of the monitored area to ensure worker awareness prior to 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 support accident condition response and are PAM system variables, as described in Table 12.3-10 and Table 7.1-7. The fixed CAMs used 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 CAMs data are capable of being supplied to the NRC Operations Center through the ERDS via a secure direct electronic data link in the event of an emergency. The ERDS connection is discussed in Section 7.2.

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. This ensures personnel working in the vicinity are able to determine easily the status of fixed CAM 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 12.3-10) and RG 8.25. The following criteria were 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 easy maintenance access in an unobstructed area. The electronic controls for the monitors are placed in the lowest dose area practical.
- Proximity of the airflow path that is as close as practical to potential release points.

The range of the fixed CAM is chosen so that the upper end of the scale is high enough to assure 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 12.3-7). 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 12.3-8). 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.

Electrical power to the fixed CAMs is provided by the following systems

- 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 quality and safety classification of the fixed area radiation monitors is provided in Table 3.2-1.

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

Table 12.3-11 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 12.3-6: A COL applicant that references the NuScale Power Plant design certification will develop the processes and programs necessary for the use of portable airborne monitoring instrumentation, including accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

12.3.5 Dose Assessment

See Section 12.4.

12.3.6 Minimization of Contamination and Radioactive Waste Generation

Design features incorporated into the NuScale Power 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 was 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 NuScale Power Plant SSC that have the potential to contain contaminated fluids have design features to reduce the likelihood of leaks, have provisions to detect leaks that do occur and reduce the spread of contamination, thus reducing the need for decontamination and the generation of waste. The list of structures and systems in a NuScale Power Plant that have been evaluated for design features consistent with RG 4.21 is provided in Table 12.3-13. The results of the system evaluations are provided in Table 12.3-14 through Table 12.3-44.

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. To address these phases, NuScale has divided the discussion into the following four design objectives and two operational objectives. Application of these six objectives demonstrates compliance with 10 CFR 20.1406 requirements.

The design features incorporated in the NuScale facility include measures to minimize and detect leakage, reduce radioactive waste generation, and facilitate decommissioning. These measures were developed to 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 are incorporated, as appropriate:

- proper selection of materials that is commensurate with the SSC service conditions to reduce the effects of corrosion, temperature, pressure, etc.
- 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

Certain design considerations can be employed to facilitate the eventual decommission process. The following facility design features are included, as appropriate:

- use of modular construction
- minimize 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 to be 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 12.3-7: A COL applicant that references the NuScale Power Plant design certification will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary to demonstrate compliance with 10 CFR 20.1406, and the guidance of 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, Oak Ridge, TN.

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12.3-4 Pacific Northwest National Laboratory, "Compendium of Material Composition for Radiation Transport Modeling," PNNL-25870, Rev. 1, March 2011.

12.3-5 American National Standards Institute/American Nuclear Society, "Design Requirement for Light Water Reactor Fuel Handling Systems," ANSI/ANS-57.1-1992, La Grange Park, IL.

12.3-6 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.

12.3-7 Electric Power Research Institute, "Calibration of Radiation Monitors at Nuclear Power Plants," EPRI #102644, Rev. 1, December 2005, Palo Alto, CA.

12.3-8 Electric Power Research Institute, "Area and Process Radiation Monitoring System Guide," EPRI #104862, Rev. 2, August 2003, Palo Alto, CA.

12.3-9 Institute of Electrical and Electronics Engineers, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," IEEE Standard 497-2002, New York, NY.

12.3-10 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.

12.3-11 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)
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	$\leq$ 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	$\geq$ 0.01 DAC $\leq$ 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	$\geq$ 0.1 DAC $\leq$ 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			$\geq$ 0.3 DAC $\leq$ 1 DAC	Areas of the plant that require respiratory protective equipment.	
IV	Airborne radioactivity area	Posting required	$\geq$ 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: Very High-Radiation Areas (>500 Rad/hr)

Room #	Description
Reactor Building	
None	
Radioactive Waste Building EL 71'-0"	
030-034 (see Figure 1.2-28)	Class A/B/C HIC Room
Radioactive Waste Building EL 100'-0"	
None	

Table 12.3-4: Typical Cobalt Content of Materials

Material or Application	Maximum Weight Percent of Cobalt
Austenitic stainless steel weld filler metals (including cladding) [1]	0.05
Austenitic stainless steel base materials [1]	0.15
Reactor vessel internal core reflector blocks	0.05
Ni-Cr-Fe base metals and weld filler metals[1] (except Alloy 690 SG tubing below)	0.05
Alloy 690 SG tubing	0.014 max average (with zero heat to exceed 0.020)
Other small components in contact with primary coolant	Not limited, however low or zero cobalt materials will be used as available
CRDM internals springs in contact with primary coolant (Inconel X-750)	1.00

Note 1: For RCS piping, reactor vessel internals (except core reflector blocks), and RPV

Table 12.3-5a: Reactor Building Areas of Potential Airborne Radioactive Material

Location & Room # (see Note 1)	Description	Source of Airborne Radioactive Material
Elevation 24’ west wing (010-007 & 010-008)	PCU filter rooms	Pool water
Elevation 24’ south wing (010-052, 052a; 010-053, 053a, & 010-054)	PCU demineralizers and adjacent valve galleries	Pool water, and PCU demineralizer resin
Elevation 24' northwest corner (010-009 & 010-012)	Degasifier Rooms	CVCS letdown
Elevation 24' north wing (010-040, 041, 042, 043, 044 & 045)	CVCS valve gallery	Primary coolant
Elevation 24' south wing (010-046, 047, 048, 049, 050 & 051)	CVCS valve gallery	Primary coolant
Elevation 35'-8" north wing (010-026, 027, 028, 029, 030 & 031)	CVCS recirculation pump room	Primary coolant
Elevation 35'-8" south wing (010-032, 033, 034, 035, 036 & 037)	CVCS recirculation pump room	Primary coolant
Elevation 50’ west wing (010-106)	Spent fuel pool cooling area	Pool water
Elevation 50’ south west side (010-134)	Reactor pool cooling area	Pool water
Elevation 50’ north and south wings (010-114, & 010-125)	Utility areas adjacent to CVCS heat exchanger valve galleries	Leaked airborne from CVCS valve galleries
Elevation 50’ east wing (010-121)	Hot lab	Primary coolant samples
Elevation 100’ north wing west half (010-409)	Containment flood and drain area	Pool water
Elevation 100’ south wing west half (010-420)	Containment flood and drain area	Pool water
Elevation 62' north wing (010-139)	MHS heat exchanger	Primary coolant
Elevation 62' south wing (010-140)	MHS heat exchanger	Primary coolant
Elevation 100' (010-422 & 423)	RXB area above pool	Pool water evaporation
Elevation 100’ north wing east half (010-411)	Hallway with CES	Vented RCS leaks into containment vessel
Elevation 100’ south wing east half (010-418)	Hallway with CES	Vented RCS leaks into containment vessel

Note 1: Refer to Figure 1.2-10 through Figure 1.2-18 for room locations.

Table 12.3-5b: Radioactive Waste Building Areas of Potential Airborne Radioactive Material

Location & Room # (see Note 1)	Description	Source of Airborne Radioactive Material
Elevation 71' center (030-011, 030-012, 030-013, 030-026, 030-027, & 030-028)	SRW resin tanks and transfer pump rooms	Spent resin
Elevation 71' center (030-014 through 030-025)	LRWS tanks and transfer pumps	Liquid radioactive waste
Elevation 71' south west (030-004, 030-005, & 030-006)	GRWS vessels and cooler rooms	Gaseous radioactive waste
Elevation 71' south west (030-033)	HIC fill station room	Spent resin
Elevation 100' (030-105)	LRW equipment room	Liquid radioactive waste
Elevation 100' (030-106, & 030-107)	LRWS drum dryers	Liquid radioactive waste
Elevation 100' (030-111, and 030-112)	Class A storage and sorting rooms	Solid radioactive waste
Elevation 100' (030-101)	Mechanical Room (RBVS and RWBVS exhaust systems)	RBVS and RWBVS Exhaust

Note 1: Refer to Figure 1.2-28, Figure 1.2-29 and Figure 1.2-30 for room locations.

Table 12.3-6: Reactor Building Shield Wall Geometry

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
24’	010-040	Module 1 CVCS ion exchanger sluice room	20” Structural steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-041	Module 2 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20”Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-042	Module 3 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20”Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-043	Module 4 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-044	Module 5 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-045	Module 6 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-051	Module 7 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-050	Module 8 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-049	Module 9 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20”Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-048	Module 10 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24’	010-047	Module 11 CVCS ion exchanger sluice room	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	20” Concrete/steel partition wall	10’ Concrete (ground floor)	20” Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed

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

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
24′	010-046	Module 12 CVCS ion exchanger sluice room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	10′ Concrete (ground floor)	20″ Concrete/steel composite slab	CVCS mixed bed and CVCS cation bed
24′	010-012	Degasifier room “A”	5′ Concrete, RXB exterior wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation)	Degasifier
24′	010-009	Degasifier room “B”	5′ Concrete, RXB exterior wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation)	Degasifier
24′	010-008	Pool cleanup filter room “A”	5' Concrete, RXB wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ concrete, RXB exterior wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation)	The dose rate from the PCU filters is assumed to be 10% of that from a PCU demineralizer.
24′	010-007	Pool cleanup filter room “B”	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ concrete, RXB exterior wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation)	The dose rate from the PCU filters is assumed to be 10% of that from a PCU demineralizer.
24′	010-052	PCUS demin room #1	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete, RXB exterior wall	20″ Concrete/steel partition wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation	PCUS demineralizer
24′	010-054	PCUS demin room #2	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete, RXB exterior wall	20″ Concrete/steel partition wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation	PCUS demineralizer
24′	010-053	PCUS demin room #3	20″ Concrete/steel partition wall	20″Concrete/steel partition wall	5′ Concrete, RXB exterior wall	20″ Concrete/steel partition wall	10′ Concrete (ground floor)	3′ Concrete (floor of 50′ elevation	PCUS demineralizer
35′-8″	N/A	Horizontal Pipe Chase	20″ Concrete	20″ Concrete	20″ Concrete	20″ Concrete	20″ Concrete	20″ Concrete	CVCS resin transfer pipe

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

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
35′-8″	010-026	Module 1 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-027	Module 2 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-028	Module 3 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-029	Module 4 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.

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

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
35′-8″	010-030	Module 5 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-031	Module 6 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-037	Module 7 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-036	Module 8 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.

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

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
35′-8″	010-035	Module 9 CVCS recirc. pump room	20″Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-034	Module 10 CVCS recirc. pump room	20″Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-033	Module 11 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.
35′-8″	010-032	Module 12 CVCS recirc. pump room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	20″ Concrete/steel composite slab	3′ Concrete (floor of 50′ elevation	The predominant radiation source for this area originates from the 24' elevation CVCS ion exchanger room.

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

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
50′	010-115	Module 1 CVCS heat exchanger room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-116	Module 2 CVCS heat exchanger room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-117	Module 3 CVCS heat exchanger room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-118	Module 4 CVCS heat exchanger room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-119	Module 5 CVCS heat exchanger room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-120	Module 6 CVCS heat exchanger room	20″ Concrete/steel partition wall	20″ Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-126	Module 7 CVCS heat exchanger room	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	20″ Structural steel partition wall	20″Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Structural steel partition wall	CVCS heat exchanger
50′	010-127	Module 8 CVCS heat exchanger room	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	20″ Structural steel partition wall	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″Concrete/steel composite slab	CVCS heat exchanger
50′	010-128	Module 9 CVCS heat exchanger room	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	20″ Structural steel partition wall	20″Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-129	Module 10 CVCS heat exchanger room	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	20″ Structural steel partition wall	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-130	Module 11 CVCS heat exchanger room	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	20″ Structural steel partition wall	20″ Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger
50′	010-131	Module 12 CVCS heat exchanger room	5′ Concrete (reactor pool wall)	20″ Concrete/steel partition wall	20″ Structural steel partition wall	20″Concrete/steel partition wall	3′ Concrete (floor of 50′ elevation)	20″ Concrete/steel composite slab	CVCS heat exchanger

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

Elev.	Room # (Note 1)	Room Name	North Wall (Note 2)	East Wall (Note 2)	South Wall (Note 2)	West Wall (Note 2)	Floor (Note 3)	Ceiling (Note 3)	Source Term
50′	010-106	Vertical pipe chase	20″ Concrete	20″ Concrete	20″ Concrete	5′ Concrete (RXB exterior)	N/A	N/A	CVCS pipe
62′	010-139	Modules 1-6 heatup heat exchangers	5′ Concrete (reactor pool wall)	20″Concrete/steel partition wall	20″Concrete/steel partition wall	20″Concrete/steel partition wall	20″Concrete/steel composite slab	3′ Concrete (floor of 75′ elevation)	CVCS heat exchanger
62′	010-140	Modules 7-12 heatup heat exchangers	20″Concrete/steel partition wall	20″Concrete/steel partition wall	5′ Concrete (reactor pool wall)	20″Concrete/steel partition wall	20″Concrete/steel composite slab	3′ Concrete (floor of 75′ elevation)	CVCS heat exchanger
75′	N/A	Modules 1-6 CVCS vertical pipe chases	20″ Concrete	20″ Concrete	5′ Concrete (reactor pool wall)	20″ Concrete	N/A	N/A	CVCS pipe
75′	N/A	Modules 7-12 CVCS vertical pipe chases	5′ Concrete (reactor pool wall)	20″ Concrete	20″ Concrete	20″ Concrete	N/A	N/A	CVCS pipe
86′	N/A	Modules 1-6 CVCS vertical pipe chases	20″ Concrete	20″ Concrete	5′ Concrete (reactor pool wall)	20″ Concrete	N/A	N/A	CVCS pipe
86′	N/A	Modules 7-12 CVCS vertical pipe chases	5′ Concrete (reactor pool wall)	20″ Concrete	20″ Concrete	20″ Concrete	N/A	N/A	CVCS pipe
100′	N/A	Modules 1-6 CVCS vertical pipe chases	20″ Concrete	20″ Concrete	5′ Concrete (reactor pool wall)	20″Concrete	N/A	N/A	CVCS pipe
100′	N/A	Modules 7-12 CVCS vertical pipe chases	5′ Concrete (reactor pool wall)	20″ Concrete	20″ Concrete	20″Concrete	N/A	N/A	CVCS pipe
126	010-022	Reactor pool area	5' Concrete wall, 4" HDPE, 5% boron content (Bioshield vertical portion in front of operating bay)	5' Concrete wall	5' Concrete wall, 4" HDPE, 5% boron content (Bioshield vertical portion in front of operating bay)	5' Concrete wall	23.5" Concrete 2 x 0.25" Steel (Bioshield)	4' Concrete roof	NPM

Note 1: Refer to Figure 1.2-10 through Figure 1.2-18 for room locations.

Note 2: A 20" concrete/steel partition wall consists of two one-half inch steel plates with 19" of concrete in between.

Note 3: A 20" concrete/steel composite slab consists of two one-half inch steel plates with 19" of concrete in between.

Table 12.3-7: Radioactive Waste Building Shield Wall Geometry

Elev.	Room #	Room Name	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling	Source Term
71′	030-004	Tank Room	20″ Concrete	20″ Concrete	20″ Concrete	48″ Concrete Wall (Facility External Wall)	60″ Concrete (Facility Basemat)	24″ Concrete	GRWS Charcoal Beds
71′	030-005	Tank Room	20″ Concrete	20″ Concrete	20″ Concrete	48″ Concrete Wall (Facility External Wall)	60″ Concrete (Facility Basemat)	24″ Concrete	GRWS Charcoal Beds
71′	030-012	Tank Room	36″ Concrete	48″ Concrete Wall (Facility External Wall)	15″ Concrete	15″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Phase Separator Tank
71′	030-013	Tank Room	36″ Concrete	15″ Concrete	15″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Phase Separator Tank
71′	030-015	Tank Room	36″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Collection Tank
71′	030-016	Tank Room	36″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Collection Tank
71′	030-018	Tank Room	24″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Collection Tank
71′	030-019	Tank Room	24″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Collection Tank
71′	030-020	Tank Room	24″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Sample Tank
71′	030-021	Tank Room	24″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Sample Tank
71′	030-024	Tank Room	24″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Sample Tank
71′	030-025	Tank Room	24″ Concrete	34″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Sample Tank
71′	030-026	Tank Room	36″ Concrete	34″ Concrete	36″ Concrete	34″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Spent Resin Storage Tank
71′	030-027	Tank Room	36″ Concrete	48″ Concrete Wall (Facility External Wall)	36″ Concrete	34″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Spent Resin Storage Tank
71′	030-033	HIC Filling Room	36″ Concrete (Note 1)	36″ Concrete (Note 1)	36″ Concrete (Note 1)	36″ Concrete (Note 1)	60″ Concrete (Facility Basemat)	24″ Concrete	HIC

Table 12.3-7: Radioactive Waste Building Shield Wall Geometry (Continued)

Elev.	Room #	Room Name	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling	Source Term
71′	030-034	Class A/B/C HIC Room	36″ Concrete	36″ Concrete	36″ Concrete	48″ Concrete Wall (Facility External Wall)	60″ Concrete (Facility Basemat)	24″ Concrete	HIC
82′	---	Pipe Chase	24″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	20″ Concrete	24″ Concrete	Resin transfer pipe
100′	030-105	LRW Processing Area	24″ Concrete	36″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	12″ Concrete - Facility Ceiling (Note 2)	LCW GAC (vessel steel wall thickness = 1.3"); LCW TUF (vessel steel wall thickness = 0.25"); LCW RO (vessel steel wall thickness = 0.25"); HCW GAC (vessel steel wall thickness = 1.3"); HCW TUF (vessel steel wall thickness = 0.25"); HCW RO (vessel steel wall thickness = 0.25"); LCW Cation Demineralizer (vessel steel wall thickness = 1.3"); LCW Anion Demineralizer (vessel steel wall thickness = 1.3"); LCW Mixed Bed Demineralizer (vessel steel wall thickness = 0.25"); LCW Cesium Demineralizer (vessel steel wall thickness = 1.3").
100′	030-106	Drum Dryer Room A	24″ Concrete	12″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	12″ Concrete - Facility Ceiling (Note 3)	Drum Dryer
100′	030-107	Drum Dryer Room B	24″ Concrete	36″ Concrete	24″ Concrete	12″ Concrete	24″ Concrete	12″ Concrete - Facility Ceiling (Note 3)	Drum Dryer

Note 1: The equivalent attenuation to an additional 4.5 inches of lead is provided for a HIC process shield.

Note 2: The equivalent attenuation to an additional one inch of steel on top of the LCW demineralizers and GAC processing skids is provided.

Note 3: The equivalent attenuation to an additional two inches of steel on top of the drum dryer skid is provided.

Table 12.3-8: Reactor Building Radiation Shield Doors

Reactor Building Door #	Reactor Building Room #[1]	Reactor Building Room Name	Reactor Building Door #	Reactor Building Room #[1]	Reactor Building Room Name
007A	010-007	Pool Cleanup Filter Rm. B	033B	010-033	Module 11 CVCS Recirc. Pump Room (middle door)
008A	010-008	Pool Cleanup Filter Rm. A	033C	010-033	Module 11 CVCS Recirc. Pump Room (east door)
009A	010-009	Degasifier Room B	034A	010-034	Module 10 CVCS Recirc. Pump Room (west door)
012A	010-012	Degasifier Room A	034B	010-034	Module 10 CVCS Recirc. Pump Room (middle door)
016A	Corridor	24′ Elevation; North corridor (southeast door)	034C	010-034	Module 10 CVCS Recirc. Pump Room (east door)
016B	Corridor	24′ Elevation; North corridor (northeast door)	035A	010-035	Module 9 CVCS Recirc. Pump Room (west door)
016C	Corridor	24′ Elevation; North corridor (northwest door)	035B	010-035	Module 9 CVCS Recirc. Pump Room (middle door)
016D	Corridor	24′ Elevation; North corridor (southwest door)	035C	010-035	Module 9 CVCS Recirc. Pump Room (east door)
018A	Corridor	24′ Elevation; South corridor (northeast door)	036A	010-036	Module 8 CVCS Recirc. Pump Room (west door)
018B	Corridor	24′ Elevation; South corridor (southeast door)	036B	010-036	Module 8 CVCS Recirc. Pump Room (middle door)
018C	Corridor	24′ Elevation; South corridor (southwest door)	036C	010-036	Module 8 CVCS Recirc. Pump Room (east door)
018D	Corridor	24′ Elevation; South corridor (northwest door)	037A	010-037	Module 7 CVCS Recirc. Pump Room (west door)
026A	010-026	Module 1 CVCS Recirc. Pump Room (east door)	037B	010-037	Module 7 CVCS Recirc. Pump Room (middle door)
026B	010-026	Module 1 CVCS Recirc. Pump Room (middle door)	037C	010-037	Module 7 CVCS Recirc. Pump Room (east door)
026C	010-026	Module 1 CVCS Recirc. Pump Room (west door)	040A	010-040	Module 1 Ion Exchanger Room
027A	010-027	Module 2 CVCS Recirc. Pump Room (east door)	041A	010-041	Module 2 Ion Exchanger Room
027B	010-027	Module 2 CVCS Recirc. Pump Room (middle door)	042A	010-042	Module 3 Ion Exchanger Room
027C	010-027	Module 2 CVCS Recirc. Pump Room (west door)	043A	010-043	Module 4 Ion Exchanger Room
028A	010-028	Module 3 CVCS Recirc. Pump Room (east door)	044A	010-044	Module 5 Ion Exchanger Room
028B	010-028	Module 3 CVCS Recirc. Pump Room (middle door)	045A	010-045	Module 6 Ion Exchanger Room
028C	010-028	Module 3 CVCS Recirc. Pump Room (west door)	046A	010-046	Module 12 Ion Exchanger Room
029A	010-029	Module 4 CVCS Recirc. Pump Room (east door)	047A	010-047	Module 11 Ion Exchanger Room
029B	010-029	Module 4 CVCS Recirc. Pump Room (middle door)	048A	010-048	Module 10 Ion Exchanger Room

Table 12.3-8: Reactor Building Radiation Shield Doors (Continued)

Reactor Building Door #	Reactor Building Room #[1]	Reactor Building Room Name	Reactor Building Door #	Reactor Building Room #[1]	Reactor Building Room Name
029C	010-029	Module 4 CVCS Recirc. Pump Room (west door)	049A	010-049	Module 9 Ion Exchanger Room
030A	010-030	Module 5 CVCS Recirc. Pump Room (east door)	050A	010-050	Module 8 Ion Exchanger Room
030B	010-030	Module 5 CVCS Recirc. Pump Room (middle door)	051A	010-051	Module 7 Ion Exchanger Room
030C	010-030	Module 5 CVCS Recirc. Pump Room (west door)	052A	010-052	PCU Demin Room #1
031A	010-031	Module 6 CVCS Recirc. Pump Room (east door)	053A	010-053	PCU Demin Room #3
031B	010-031	Module 6 CVCS Recirc. Pump Room (middle door)	114C	010-114	50’ Elevation North utilities area (west door)
031C	010-031	Module 6 CVCS Recirc. Pump Room (west door)	114D	010-114	50’ Elevation North utilities area (middle door)
032A	010-032	Module 12 CVCS Recirc. Pump Room (west door)	114E	010-114	50’ Elevation North utilities area (east door)
032B	010-032	Module 12 CVCS Recirc. Pump Room (middle door)	125C	010-125	50’ Elevation South utilities area (west door)
032C	010-032	Module 12 CVCS Recirc. Pump Room (east door)	125D	010-125	50’ Elevation South utilities area (middle door)
033A	010-033	Module 11 CVCS Recirc. Pump Room (west door)	125E	010-125	50’ Elevation South utilities area (east door)

Note 1: Refer to Figure 1.2-10, Figure 1.2-11 and Figure 1.2-12 for room locations.

Table 12.3-9: Radioactive Waste Building Radiation Shield Doors

Radioactive Waste Building Door #	Radioactive Waste Building Room #[1]	Radioactive Waste Building Room Name	Radioactive Waste Building Door #	Radioactive Waste Building Room #[1]	Radioactive Waste Building Room Name
020A	030-020	Tank room (82’ Elev.)	018A	030-018	Tank room (82’ Elev.)
021A	030-021	Tank room (82’ Elev.)	016A	030-016	Tank room (82’ Elev.)
024A	030-024	Tank room (82’ Elev.)	015A	030-015	Tank room (82’ Elev.)
025A	030-025	Tank room (82’ Elev.)	106A	030-106	Drum dryer room
026A	030-026	Tank room (82’ Elev.)	107A	030-107	Drum dryer room
027A	030-027	Tank room (82’ Elev.)	105A	030-105	LRWS equipment room
028A	030-028	Pump room (71’ Elev)	033A	030-033	HIC filling room
019A	030-019	Tank room (82’ Elev.)	034A	030-034	Class A/B/C HIC Storage

Note 1: Refer to Figure 1.2-28, Figure 1.2-29 and Figure 1.2-30 for room locations.

Table 12.3-10: 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. 3 Equipment Qualification Post-Accident Radiological Source Term	Type B Type C
Hot lab	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 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. 3 Radiological Source Term	Type E
Radiation monitors in route to hot lab	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Safety instrument rooms	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
EDSS switchgear rooms	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Radiation monitors in route to safety instrumentation rooms and EDSS switchgear rooms	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
RXB access tunnel	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
MCR envelope - main control room area radiation monitor	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
MCR envelope - main control room 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. 3 ANSI/HPS 13.1-2011	Type E
Technical support center	Fixed area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Primary Sampling Equipment	Fixed Area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E

Table 12.3-10: Fixed Area and Airborne Radiation Monitors Post-Accident Monitoring Variables (Continued)

Area	Monitor	Estimated Dynamic Detection Range	Principal Parameter measured	Basis for Dynamic Range	PAM Variable Type
Containment Sampling System Equipment	Fixed Area	1E-4 to 1E+4 rem/hr	gamma	RG 1.97, Rev. 3 ANSI/ANS-HPSSC-6.8.1-1981	Type E
Reactor Building 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. 3 Radiological Source Term ANSI 42.18-2004	Type E

Table 12.3-11: Fixed Airborne Radiation Monitors

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 24′	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 50′	Hot lab	1 / Particulate	Cs-137: γ	3E-10 to 1E-6 µCi /cc	Yes / E
Reactor Building Elevation 100'-0"	Reactor 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
Annex Building elevation 100′	Hot shop	1 / Particulate	Cs-137: γ	3E-10 to 1E-6 µCi /cc	No
	Decontamination shop	1 / Particulate	Cs-137: γ	3E-10 to 1E-6 µCi /cc	No
Radioactive Waste Building elevation 71′	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 76'-6"	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 12.3-12: Fixed Area Radiation Monitors

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 24′	NW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	West passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Pool cleanup filter rooms	2 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Pool cleanup demineralizers	3 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	East passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	CVCS ion exchanger proximity (serve the CVCS reactor coolant filters)	12 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
Reactor Building elevation 35′-8″	CVCS pump rooms	12 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No

Table 12.3-12: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 50′	NW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	West passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	CVCS heat exchanger rooms	12 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	East passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	Hot lab	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
Reactor Building elevation 62′	NW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	West passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	East passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Module heatup system heat exchanger rooms	2 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No

Table 12.3-12: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 75′	NW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	West passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	Safety I&C and EDSS equipment rooms	4 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	East passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
Reactor Building elevation 86′	NW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	Safety I&C and EDSS equipment rooms	4 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E

Table 12.3-12: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 100′	NW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NW refuel area floor	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Spent fuel mast bridge (used during fuel movement)	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
Reactor Building elevation 100′	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	East passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	NE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	SW refuel area floor	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Inside bioshield monitor	24 / gamma-sensitive	gamma γ	1E0 to 1E+7 rem/hr	Yes / B & C

Table 12.3-12: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Reactor Building elevation 126′	NW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Spent fuel pool area	2 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Reactor pool monitor	10 / gamma-sensitive	gamma γ	1E0 to 1E+7 rem/hr	No
Control Building elevation 50′	North entrance space	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	South entrance space	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	South bottle storage room	1 / gamma-sensitive	gamma ?	1E-4 to 1E+4 rem/hr	No
	Entrance bottle storage room	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	North bottle storage room	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Central stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	North stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	South stairwell	1 / gamma-sensitive	gamma ?	1E-4 to 1E+4 rem/hr	No
Control Building elevation 76'-6"	Reactor Building access tunnel	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	South stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Main control room envelope	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E

Table 12.3-12: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Control Building elevation 100′	Technical support center	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	Technical support center hallway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	Yes / E
	North stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	South stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
Control Building elevation 120′	Central area	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	North stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	South stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
Annex Building elevation 100′	Entry and exit vestibule	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Hot workshop	1 / gamma-sensitive	gamma γ	1E-5 to 1E+4 rem/hr	No
	Exit turnstiles	1 / gamma-sensitive	gamma γ	1E-5 to 1E+4 rem/hr	No
	Connector to Reactor Building and Radioactive Waste Building	1 / gamma-sensitive	gamma γ	1E-5 to 1E+4 rem/hr	No
Radioactive Waste Building elevation 71′	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Telecom room area	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	LRWS storage tank pump rooms	4 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SRWS storage tank pump rooms.	2 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	GRWS heat exchanger room	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	GRWS vessel tank room	2 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Waste drum storage room	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
Radioactive Waste Building elevation 82′	LRWS storage tank rooms	8 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SRWS storage tank rooms	4 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No

Table 12.3-12: Fixed Area Radiation Monitors (Continued)

Building and Elevation	Area	Detector Quantity / Type	Principal Parameter Measured	Nominal Range	PAM / Type
Radioactive Waste Building elevation 100′	Truck bay	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	SW stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	NE stairwell	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	Waste management control room	1 / gamma-sensitive	gamma γ	1E-5 to 1E+4 rem/hr	No
	NE passageway	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	East rollup door area	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	LRWS skids area	1 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
	HIC tank rooms storage area	3 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No
Turbine Building elevation 100′	General area	2 / gamma-sensitive	gamma γ	1E-4 to 1E+4 rem/hr	No

Table 12.3-13: NuScale Power Plant Systems with NRC Regulatory Guide 4.21 Evaluation

System Code	System Name	System Code	System Name
ABS	auxiliary boiler system	PSCS	pool surge control 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
CFWS	condensate and feedwater system	RCS	reactor coolant system
CPS	condensate polishing system	RPCS	reactor pool cooling system
CRVS	normal control room HVAC system	RWBVS	Radioactive Waste Building HVAC system
CVCS	chemical and volume control system	RWB	Radioactive Waste Building
CWS	circulating water system	RWDS	radioactive waste drain system
DHRS	decay heat removal system	RXB	Reactor Building
		SCWS	site cooling water system
GRWS	gaseous radioactive waste system	SFPCS	spent fuel pool cooling system
LRWS	liquid radioactive waste system	SRWS	solid radioactive waste system
MSS	main steam system	UHS	ultimate heat sink
PCUS	pool cleanup system	UWS	utility water system
PLDS	pool leakage detection system	DWS	demineralized water system

Table 12.3-14: 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 auxiliary boiler system (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.
Objective 2: Provide leak detection capability	The ABS is provided with several pressure and level instruments throughout the system to monitor system performance.
	The ABS has four radiation monitors: One on the vent of the high pressure condensate collection tank to monitor the exhaust from the collection tank and prevent the release of radioactive effluents into the atmosphere. The second and third are on the returns from each of the two module heatup system (MHS) heaters to monitor the condensate exiting the MHS heaters to prevent the high pressure boiler system from becoming contaminated in the event of a tube leak in the MHS heaters. The fourth radiation monitor is on the cross-over from the high pressure boiler system to the low pressure boiler system to prevent the low pressure system from becoming contaminated in the event the high pressure system becomes contaminated.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	Isolation valves between the clean and contaminated systems minimize potential for cross-contamination.
	Drains are provided for the ABS system and direct the drainage to the balance-of-plant drain system (BPDS) or to the LRWS if it becomes contaminated.
Objective 4: Facilitate decommissioning	Parts of the ABS (boiler skids and chemical addition skid) 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 12.3-15: 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 American Society of Mechanical Engineers (ASME) B31.1. This design feature reduces the potential for leaks.
	The BPDS uses double-walled pipes with leak detection in the underground piping from the underground sump tanks to the LRWS. This also applies to Objective 2.
	The underground BPDS sump tanks are located in stainless steel lined concrete sump pits to provide two boundaries to contain the contents from leaking out of the tank into the concrete or the environment.
	The BPDS is a normally non-radioactive system that serves areas that are outside of the RCA. However, design features are included to contain contaminated fluid, if leaks develop from contaminated systems, including sump tanks and sump pumps to prevent overflow conditions.
Objective 2: Provide leak detection capability	The BPDS interfaces with drains from systems that can become contaminated (i.e. condensate polishing system, ABS, north and south Turbine Generator Building floor drains). Process radiation monitors are provided in the drain lines from these sources to monitor for radioactive contamination. The sump tanks are provided with level instruments to assist in detecting leaks that are drained to the BPDS.
	The underground BPDS sump tanks are equipped with the moisture detectors between the sump tanks and the stainless steel lined concrete sumps to monitor for leaks from the sump tanks.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The BPDS components are designed for service for the life of the plant. The wetted parts of the piping, valves, tanks, instrumentation, and controls are made of either carbon steel or stainless steel material, depending on the type of chemicals or fluid being processed.
	The BPDS drain lines are sloped to facilitate gravity draining and minimize the accumulation of solids.
	The BPDS contents are kept separate from RWDS to minimize waste generation. Liquids from BPDS are sent to LRWS if the radioactive contamination levels warrant.
Objective 4: Facilitate decommissioning	The major, large components of the BPDS are located outside of buildings, allowing easy access to facilitate decommissioning.
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 Evacuation System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The containment evacuation system (CES) uses welded components and welded piping to minimize potential for leaks and spills.
	The use of stainless steel condenser tubes, piping and valve material prevent corrosion leading to pressure boundary failure. This also applies to Objective 3.
	The CES equipment vents and drains are directed to LRWS and RWDS. The leaks and spills are routed to the RWDS floor drain hubs via RXB sloped floors.
Objective 2: Provide leak detection capability	The fixed radiation monitors in proximity of the CES equipment rooms detect high radiation levels in the RXB areas.
	The CES control system uses input from pressure, temperature, flow, and radiation instruments to monitor system performance. The pressure transmitters on the CES condenser and CES sample vessel provide indication that the system may have developed a leak.
	The RWDS sumps located in the RXB are equipped with the level transmitter to detect leakage from CES components.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The CES sample vessel has provisions for flushing with demineralized water and nitrogen.
	The clean utility systems that interface with the CES have a minimum of two barriers to prevent cross contamination.
Objective 4: Facilitate decommissioning	The CES components are designed for the life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning. The majority of the CES components are located outside of rooms and cubicles that contain radiation sources. This design feature minimizes equipment contamination and reduces the effort for 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 Containment Flooding and Drain System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The CFD containment drain separator tanks are located in a curbed area of the RXB to contain leaks or spills. The drains collected in the curbed area are transferred to the RWDS. This also applies to Objective 3.
	The CFDS uses corrosion resistant materials that reduce the potential for leaks and contamination.
	Major CFDS components are located inside a curbed area to collect and drain leaks to the RWDS. This also applies to Objective 3.
	CFDS floors are sloped to collect leaks and direct it to the drain sump within the curbed area.
Objective 2: Provide leak detection capability	Pressure, flow, temperature, and level transmitters provided throughout the system are used to monitor system performance and possible system pressure boundary failure.
	The RWDS sumps located in the RXB are equipped with the level transmitters to detect the piping and valve, pump or tank leakages. This feature warns operators of equipment leakage or spills.
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.
Objective 4: Facilitate decommissioning	The CFDS piping is above the ground and 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 12.3-18: 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 condensate and feedwater system (CFWS) piping and components are designed to appropriate industry codes and standards, and are made of corrosion resistant material that is compatible with the operating environment of CFWS. This design feature reduces the potential for leaks.
	The welded construction of the CFWS piping, valves, and components reduces the probability of leaks and the spread of contamination. This also applies to Objective 3.
Objective 2: Provide leak detection capability	Grab sampling locations are provided in several locations throughout the system to monitor the chemical composition and radiation contamination level of the CFWS.
	The BPDS monitors the Turbine-Generator Building floor drains for radioactive contamination.
Objective 3: Reduce contamination to minimize releases, cross-contamination & waste generation	The CFWS components are selected for reliable service for the life of the plant.
	The BPDS handles the drainage from the CFWS, including the chemical wastes drainage or spills. The BPDS system has the capability to detect radiation and redirect contaminated drainage to the LRWS.
Objective 4: Facilitate decommissioning	The CFWS piping is routed above the ground, as much as possible. This facilitates the decommissioning effort.
	The CFWS components are mostly located outside the RCA, making it easier to decontaminate and decommission.
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 Condensate Polishing System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The condensate polishing system (CPS) demineralizers, regeneration tanks & chemical tanks, pumps, piping, and valves are designed with corrosion resistant materials in accordance to the applicable ASME and American Petroleum Institute codes. The welded construction for these components reduces equipment failure and leaks.
	The water chemistry exiting the polishers will be maintained to reduce corrosion in the condensate and feedwater systems, steam generators and the steam turbine.
	The CPS relief valves on the demineralizers protect the system from over pressurization and prevent damage to the components. The discharge lines from these relief valves are routed to the BPD system.
Objective 2: Provide leak detection capability	The CPS demineralizer skid, filters and rinse recycle pump skid are provided with a dedicated CPS sump that drains to BPDS drains and to the BPDS sump. The BPDS sump has level detection to detect leakage.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The floor and equipment drains are provided for the CPS equipment to direct spills and leaks to the nearest BPDS sump, by gravity, to minimize spread of contamination.
	The sluice lines are sized to ensure sufficient velocity is available to prevent resin from settling in the piping system. The resin transfer lines can be flushed with the feedwater system to minimize personnel exposure when these lines become contaminated.
	The resins in the cation and anion regeneration vessels are washed with acid and caustic and recycled back to the CPS. This minimizes waste resin generation.
	Utility connections such as service air and demineralized water to CPS are designed with a minimum of two barriers to prevent cross contamination of non-radioactive systems.
	The demineralizers and filters/strainers remove radioactivity resulting from primary to secondary leakage to minimize the potential spread of contamination.
Objective 4: Facilitate decommissioning	The CPS components are designed for the life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning.
	The piping systems are above the ground and embedded piping is minimized, to the extent practical.
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 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, and fan coil units, located in the CRB provide filtered and 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 minimizes the potential for contaminated air from entering the system. This also applies to Objective 3.
	The CRVS components are designed, fabricated and tested in accordance with industry codes and standards and constructed using corrosion resistant materials.
	The condensation from CRVS components are hard piped to the BPDS. This also applies to Objective 3.
Objective 2: Provide leak detection capability	The process radiation monitors are provided downstream of the CRVS filter unit. The radiation monitors provide the radiological contamination levels in the air after being processed by CRVS HEPA filter/charcoal unit.
	Duct and housings leak testing is performed after installation in accordance with ASME AG-1, Section TA and Article SA-4500.
	Periodic in-place testing of the atmospheric cleanup portions are tested in accordance with RG 1.140 and ASME N510.
	The BPD sumps located in the CRB are equipped with level transmitters to detect for leakage.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The CRVS maintains the areas within the CRB at a positive pressure with respect to the outside environment to minimize the infiltration of air.
	The CRVS detects and limits the introduction of airborne radioactivity utilizing radiation monitors and filtration, or isolation dampers, in the intake duct.
Objective 4: Facilitate decommissioning	The piping and ductwork associated with the CRV system 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 12.3-21: 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 resistance materials to minimize degradation over the life of the plant. Welded construction is used as practicable, which minimizes the potential for leaks. The CVCS components are located in a Seismic Category I building (RXB).
	A chemical mixing tank is provided to allow 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 RWD system.
	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 recirculation pumps are canned motor pumps, which minimizes the potential for leaks.
	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 for each module are placed in cubicles on elevation 24′ of the RXB. These cubicles are equipped with moisture sensors for early leak detection. The CVCS and MHS heat exchangers and pumps, located on higher elevations, drain to the RWDS sump on elevation 24′.
	The CVCS expansion tank has a level transmitter that monitors the water level in the tank to provide indication of potential CVCS leakage.
	The CVCS system provides the capability to inject argon into the RCS which increases the sensitivity to detect SG tube leaks using argon-41.
	The CVCS is provided with 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 LRWS, and the NPM injection and discharge lines. The mass flow mismatch indication provides leak detection capability to warn operators of abnormal conditions.
	The area radiation monitors near the components can assist in detecting potential leaks.
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 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.
	An automatic diverting three-way valve diverts flow around the ion exchangers if the fluid temperature exceeds the high temperature setpoint for Ion Exchanger resin protection, preventing resin damage, and minimizing waste generation.
	The equipment drains and vents from the CVCS are sent to the RWDS and eventually transferred to the LRWS, and can be recycled back into the CVCS or RCS to minimize waste generation.
	The ion exchange and reactor coolant filters remove particulate and impurities from the CVCS water. This reduces the contamination of downstream equipment.
	Drain and vent lines for CVCS and MHS major equipment (pumps, tanks, heat exchangers, filters, and resin traps) allow for the direct draining into the equipment drain system prior to maintenance activities. This reduces the potential spread of contamination.

Table 12.3-21: Regulatory Guide 4.21 Design Features for Chemical and Volume Control System (Continued)

Objective	Design Features
Objective 4: Facilitate decommissioning	The CVCS components are designed for the life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning.
	The CVCS components are designed with flushing capabilities. Design features, such as welding techniques and surface finishes, are included to facilitate decontamination processes.
	Embedded CVCS piping is minimized, as much as practical.
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 Circulating Water System

Objective	Design Features
Objective 1: Minimize the potential for leaks/ spills and provide containment areas	The circulating water system (CWS) is designed to minimize the effects of corrosion using a combination of corrosion resistant materials, cathodic protection and water treatment with corrosion inhibitors. This also applies to Objective 3.
	The CWS provides cooling water to the tube side of the turbine-generator's main condensers and operates at a higher pressure than the condenser shell side to keep leakage out of CWS and into the condenser. This also applies to Objective 3.
Objective 2: Provide leak detection capability	Grab samples are periodically taken from the cooling tower basin to monitor the water chemistry and for radioactive contamination.
	CWS leakage into the Turbine Generator Building will be collected by the BPDS, which is monitored for radionuclide contamination.
Objective 3: Reduce contamination to minimize releases, cross-contamination & waste generation	The CWS components are designed for service for the life of the plant.
Objective 4: Facilitate decommissioning	The CWS is a normally non-contaminated system that is protected by multiple barriers from contaminated systems (SG tubes, condenser tubes, and higher operating pressure than the condensing steam) and is periodically monitored for contamination, reducing the probability and magnitude of contamination that will facilitate eventual 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 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 system 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 full service life of the plant and is designed, to the extent practicable, for easy removal.
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, such as packless metal diaphragm valves, wherever possible to minimize the potential for leaks.
	Proven materials and welded construction are used to minimize leaks.
	Embedded piping is minimized and liquid containing lines are sloped to facilitate drainage. This also applies to Objective 4.
	Adequate charcoal is provided in each train of the GRWS to reduce the gaseous radioactive effluents to minimize amount of contaminated gas released to the RBVS and or the environment. This also applies to Objective 3.
Objective 2: Provide leak detection capability	Gas analyzers will detect potentially explosive gas mixtures to enable operators to prevent an event that would challenge system boundary integrity.
	Leak detection is provided by area airborne radiation monitors.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The GRWS component materials are selected for reliable service for the life of the plant, reducing waste generation from replacing these components.
	Components are designed with materials that are compatible with the operating environment, as well as meeting codes and standards identified in RG 1.143.
	Connections with clean utility systems are designed with at least two barriers to prevent cross-contamination of the clean systems.
	The process piping containing contaminated fluid is sloped to facilitate flow and reduce traps, thus reducing waste generation.
	Welding techniques and material finishes are designed to provide smooth internal surfaces for easy decontamination. This also applies to Objective 4.
	The GRWS piping arrangements use manifolds, as much as practical, to reduce the amount of piping.
	The GRWS process radiation monitors return the sampled stream back to the process stream. This feature reduces waste generation.
	In-line radiation monitors minimize the spread of contamination and waste generation by returning sampled gas back to the process stream.
Objective 4: Facilitate decommissioning	The GRWS components are designed, to the extent practical, as individual elements, for ease of removal during decommissioning.
	Nitrogen is provided to purge and decontaminate the system internals prior to disassembly and decommissioning.
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 containment areas	The LRWS uses proven materials and welded construction in accordance with applicable codes and standards for piping, valves and components. This design minimizes leaks and contamination of the facility and the environment. This feature is also applicable to Objective 3.
	The LRWS tanks are designed to provide adequate holdup capacity of the liquid waste generated in the plant. The HCW and LCW collection and sample tanks are cross-tied to provide operational flexibility and the space needed for storing waste during abnormal conditions. This feature prevents tank overflow and spread of contamination.
	The LRWS is provided with two processing skids that are similar in design and can process either waste type (HCW or LCW). This design feature minimizes the tank overflows, and spread of contamination.
	The tank overflow lines are sized for maximum surge input. The discharge lines are routed close to the floor to prevent tank surface, adjacent walls or the personnel becoming contaminated.
	The LRWS piping located in RWB is above ground to the extent practicable, except for the double-walled buried pipe located in the discharge pipe from the sample collection tanks. The double-walled pipe annulus is pressurized to prevent leakage to the environment.
	The tank level instruments in the LRWS facilitate automated pump starting and stopping. This design approach minimizes tank overflow and prevents the spread of contamination.
	The LRWS collection, and sample tanks are located in a separate shielded concrete cubicle with stainless steel liner to contain the entire volume of each tank.
Objective 2: Provide leak detection capability	Pressure transmitters on the pressurized tanks such as degasifier, ion exchangers, and charcoal bed (granulated activated charcoal) provide system pressure boundary failure information.
	The LRWS tank cubicles are provided with drains that are directed to the RWDS sump tanks, which have level instruments to detect leaks.
	The LRWS tanks are equipped with the level detectors to monitor tank level changes, leaks or overflow conditions.

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

Objective	Design Features
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The LRWS components are selected for reliable service for the life of the plant, reducing waste generation. The wetted parts of piping, valves, tanks, filter housing, instrumentation, and controls are made of corrosion resistant material.
	The LRWS has the capability to recycle processed water back into the plant systems. This feature minimizes the waste generation.
	The LRWS has CIP skids for cleaning or flushing collection tanks, sample tanks, equipment processing skids and sluicing resin out of the LRWS demineralizer vessels. The CIP prevents cross contamination of the demineralized water system (DWS).
	The LRWS tanks are either vented directly into HVAC ductworks, or vented into a canopy type vent to prevent contaminating the air space.
	The level transmitter instruments on the collection, sample, and detergent tanks have diaphragm seal design to prevent instruments from becoming contaminated. This feature reduces the quantities of contaminated equipment requiring decontamination or disposal during the decommissioning period.
	The LRWS collection tanks, sample tanks, and detergent tanks have conical bottom design to reduce crud or solid buildup at the bottom.
	Pipe connections with the clean utility systems are designed with at least two barriers to prevent cross-contamination of the clean system.
	The relief valves are provided on the degasifier vessels and the demineralizers to prevent tank over pressurizations. The relief valve discharge lines are routed close to the floor to reduce the spread of contamination.
	The process piping that contains contaminated fluid is sloped to facilitate flow and reduce fluid traps, thus reducing contamination.
	Welding techniques and material finishes are designed to minimize leaks and provide for easy decontamination. This design feature also applies to Objective 4.
	The LRWS tanks are strategically located to collect wastes from various sources. These wastes are segregated for different handling and processing requirements to minimize cross-contamination.
	The LRWS is designed with redundant equipment. The pumps are sized for maximum input to prevent overflow; one pump is normally used during normal operation. If one pump is out of service, the backup pump can be used to prevent overflow condition and the spread of contamination.
	The demineralizer skid has 5 vessels with different media. Each vessel has a bypass line to bypass it if not needed for processing the water. This feature minimizes resin usage (waste generation).
Objective 4: Facilitate decommissioning	The LRWS components are designed for full service life. The LRWS processing equipment is skid mounted which facilitates easy removal for decommissioning.
	The LRWS is designed with minimum embedded or buried piping. The contaminated lines are routed through pipe chases as much as practicable. This design approach facilitates decommissioning.
	The collection and sample tanks are enclosed in shielded concrete cubicles with a stainless steel liner. The liner extends to a level sufficient to contain the volume of liquid in the tank. This reduces contamination of the concrete and facilitates decommissioning.
	CIP flush water or the DWS break tank is provided to flush and decontaminate the equipment in the LRWS prior to maintenance or preparation for decommissioning.
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 main steam system (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 also applies to Objective 3.
	MSS uses welded connections for piping and equipment, except where threaded or flanged joints are required for maintenance. This 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 area radiation monitors can assist in detecting steam leaks into the RXB.
	The condenser air removal system radiation monitors are designed to monitor effluents coming from the condenser air removal system 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 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.
	A minimum of two barriers is provided between clean systems (non-radioactive systems), such as the nitrogen distribution system and the ABS, and the MSS to prevent cross-contamination.
Objective 4: Facilitate decommissioning	The MSS components are designed for the 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 have smooth surfaces and are sloped to reduce decontamination efforts and 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 Cleanup System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The PCUS uses proven materials for piping and components including valves, filter housings, and demineralizers.
	The PCUS is designed with minimum embedded or buried piping. The contaminated lines are routed through the pipe chases as much as practicable. This feature also applies to Objective 4.
	The PCUS components are located in a separate shielded cubicle inside the RXB to ensure leakage won't spread to adjacent areas. The sloped floors in each cubicle direct leakage to the nearest RWDS drain hub.
	The PCUS components (demineralizers, filters, and resin traps) have vents and drains that are hard piped to the RWDS.
Objective 2: Provide leak detection capability	An area radiation monitor is provided in each equipment room to assist in detecting spills and leaks.
	The PCUS is provided with multiple instruments (pressure, level, flow, pressure differential across filters and demineralizers) to monitor system performance and detect pressure boundary failure.
Objective 3: Reduce contamination to minimize releases, cross-contamination & waste generation	Components are designed with materials that are compatible with the operating environment. The wetted parts of piping, valves, demineralizers, filter housing, instrumentation, and controls are made of stainless steel material to reduce corrosion to reduce the potential for equipment failures.
	The filters and demineralizers remove impurities and minimize pool contamination. The PCUS has the capability to recycle the pool water after being filtered and demineralized. This feature reduces waste generation.
	Temperature transmitters upstream of the PCUS demineralizers protect the resin from being exposed to high temperature, preventing the damage of resins and reducing waste generation.
	Connections with the DWS are designed with at least two barriers to prevent cross-contamination of clean system.
	Relief valves are provided on the demineralizer vessels to prevent over pressurizations. The relief valves discharge lines are routed close to the floor to reduce the spread of contamination.
	The PCUS demineralizers and resin traps are provided with flushing capability to reduce the contamination levels within the system. This also applies to Objective 4.
Objective 4: Facilitate decommissioning	The PCUS components are designed for full service life and easily removable elements during decommissioning.
	The PCUS is designed with minimum embedded or buried piping. The contaminated lines are routed through pipe chases as much as practicable.
	Demineralized water break tank water is provided to flush and decontaminate equipment in the PCUS prior to maintenance or preparation for 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 Leak Detection System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The pool leakage detection system (PLDS) has leak channels behind the pool wall liner and under the pool floor liner 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 to prevent corrosion. This feature reduces the potential for leaks. This 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 liner.
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.
	The PLDS drain lines from the leak channels are gravity drained to the RWDS through enclosed pipes. This design feature prevents spread of contamination.
Objective 4: Facilitate decommissioning	The PLDS is designed for full service life of the plant. With the exception of the leak channels embedded into the concrete, the individual drain lines and the main header are above the ground. This design feature facilitates decommissioning.
Objective 5: Operating programs and documentation	COL item
Objective 6: Site radiological environmental monitoring	COL item

Table 12.3-29: Regulatory Guide 4.21 Design Features for Pool Surge Control System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The PSCS is designed with a catch basin to contain the total volume of the PSCS storage 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 welded stainless steel collection tank and piping system reduces the potential for leaks.
	The pool water from a surge event is processed through the PCUS and stored in the pool 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 also applies to Objective 3.
	The floor of the catch basin collects leakage and directs it to the drain sump within the catch basin. The sump transfers this fluid to the LRWS, minimizing the spread of contamination. This also applies to Objective 3.
	The use of underground, or buried piping, is minimized to the extent practicable. The PSCS is designed with double-walled pipe for lines between the RXB, RWB, and the yard (PSCS storage tank), reducing the potential for contamination of the environment. This also applies to Objective 4.
Objective 2: Provide leak detection capability	The sump located in the catch basin is equipped with a level transmitter to detect tank leakage.
	The PSCS storage tank located within the catch basin is equipped with the level transmitter to detect the tank leakage.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The PSCS storage tank bottom and the catch basin floor are sloped to reduce the potential for contamination buildup. The catch basin drain sump sends the water to the LRWS. Any leakage from the double-walled pipe annulus is directed to the RWDS. This feature minimizes the potential for environmental contamination.
Objective 4: Facilitate decommissioning	The PSCS components are designed for the 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 12.3-30: 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. This includes 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 sample. 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 & 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, instrumentations, and controls are made of stainless steel material to reduce corrosion.
	The automatic samples are returned back to the originating system as much as practical 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 sample 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 at least two barriers to prevent cross-contamination.
	The PSS welding techniques and material finishes are designed to minimize leaks and provide smooth internal surfaces for easy decontamination.
	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 full service life 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.
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 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 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 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.
	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 air handling units and fan coil units are provided with 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 piping and ductwork 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 12.3-32: Regulatory Guide 4.21 Design Features for Reactor Component Cooling Water System

Objective	Design Features
Objective 1: Minimize the potential for leaks/ spills and provide containment areas	The reactor component cooling water system (RCCWS) is normally a clean closed loop (non-radioactive) system, and consists of piping and equipment that are made of stainless steel that is compatible with the RCCWS operating environment. This minimizes corrosion and potential leaks. This also applies to Objective 3.
	The RCCWS is located in the RXB with sloped floors that direct drainage to the nearest drain hub. This also applies to Objective 3.
Objective 2: Provide leak detection capability	The RCCWS is provided with process radiation monitors downstream of interfacing heat exchangers to detect cross contamination due to heat exchanger leakage.
	The RWDS RCCW 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 & waste generation	The RCCWS components are selected for reliable service for the life of the plant.
	The RCCWS is a closed loop intermediate system that removes heat from radioactive systems and transfers that heat to SCWS that releases the heat to the environment. The use of RCCWS as an intermediate cooling loop reduces the potential for releasing contamination.
	The RCCWS interface with the DWS is designed with a minimum of two barriers to prevent cross-contamination of the DWS.
Objective 4: Facilitate decommissioning	The RCCWS piping and equipment are made of stainless steel material with smooth surfaces 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 12.3-33: 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 reactor coolant system (RCS) is entirely contained within the reactor pressure vessel (RPV), which is an ASME BPVC Section III, Class 1 vessel, and is periodically inspected according to the applicable portions of ASME Section XI (See Section 5.2.4).
	Penetrations on the upper vessel head are welded such that potential leakage from these penetrations will be confined within the containment vessel.
	The RCS components are designed for the 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	RCS leaks into the SG will be detected by radiation monitors associated with the MSS or the condenser air removal system. Leakage monitoring for the SGs is in accordance with EPRI 97-06 (See 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 (see Section 5.2.5).
	The CVCS system 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. The RCS piping and equipment have smooth surfaces. This design feature facilitates decommissioning.
	The RCS is designed as part of a module that will facilitate 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 Reactor Pool Cooling System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The six-inch curb around the reactor pool cooling system (RPCS) heat exchangers in the RXB is designed to contain the leaks or spills. This also applies to Objective 3.
	The RPCS piping is designed according to ASME B31.1 using corrosion resistant materials, like stainless steel, for piping, valves, pumps, and wetted parts of the heat exchangers.
Objective 2: Provide leak detection capability	The RPCS has temperature, pressure, conductivity, and flow instruments to monitor system performance and to assist in identifying system leaks.
	The RWD sumps located in the RXB are equipped with level transmitters to detect RPCS leaks.
	Although the SCWS is normally at a higher pressure than RPCS, radiation detectors are provided in the site cooling water system to detect RPCS heat exchanger tube leaks into the SCWS.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The RPCS interfaces with the PCUS for the cleanup of the pool water to reduce the contamination levels.
	The RPCS equipment vents, drains, and relief valve discharges are hard piped to a drain hub to minimize the spread of contamination. Isolation arrangements are provided for major components to allow flushing using the vent and drain connections.
Objective 4: Facilitate decommissioning	The RPCS components are designed for the life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning.
	An epoxy coating is applied for the areas surrounding the suction strainers, pumps and heat exchangers to minimize contamination of concrete and facilitate decommissioning. This also applies to Objective 3.
	The RPCS components are above ground and embedded piping is minimized. This design prevents ground contamination and facilitates decommissioning.
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 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 RWBVS is performed in accordance with RG 1.140 and ASME N510.
	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 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 are equipped with 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 air handling units and fan coil units are provided with filters to reduce the potential contamination levels of the air.
	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 12.3-36: Regulatory Guide 4.21 Design Features for Radioactive Waste Building

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The RWBS is designed with the shielded tank cubicles to contain the total volume of each tank. Each tank room in the RWB that contains radioactive fluid has a welded stainless steel liner. Wall penetrations below the liquid containment level are minimized.
	The RWB floors are sloped to collect floor surface drainage and direct it to the appropriate RWDS drain hub to minimize the spread of contamination.
	Embedded floor drains in the RWB are minimized to extent practicable.
	The RWB is designed with a minimum number of structural joints to contain water and provide protection against ground water intrusion. The penetrations below grade are minimized as much as practicable.
Objective 2: Provide leak detection capability	See Objective 6.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The equipment rooms that contain radioactive fluids are designed with curbs or walls to prevent leaks from spreading into the other areas.
	The RWB is designed with pipe chase and a concrete tunnel between the RWB and RXB to provide a shielded pathway for contaminated lines, to minimize the spread of contamination. The concrete tunnel will collect leaks to RWDS, is inspectable, and prevents ground water intrusion.
	The RWB is designed to minimize use of structural joints and wall penetrations below the liquid containment level to prevent leaks in one area from cross-contaminating adjacent areas.
Objective 4: Facilitate decommissioning	Surfaces with the potential for contamination are protected with an epoxy coating to minimize contamination of concrete and facilitate decommissioning.
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 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, 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	The CVCS demineralizer cubicles are equipped with a moisture detector for early leak detection.
	The RWDS sump tanks have leak detection capability (moisture sensor) 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 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.
	Pipe connections with the clean utility system are designed with at least two barriers to prevent cross-contamination.
	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.
	Welding techniques and material finishes are designed to provide smooth internal surfaces to facilitate decontamination. This also applies to Objective 4.
	The RWDS sump tanks are strategically located to collect floor and equipment drains from various sources. The drains are segregated (floor, equipment, chemical, RCCW, and detergent drains) for different handling and processing requirements to minimize cross-contamination.
	Sump tanks vent directly into the HVAC system (ductwork) to reduce airborne contamination.
	The floor drains on upper elevation of the RXB and the RWB contain loop seals to prevent spread of airborne contamination between the building elevations or the rooms that are connected to the same drain header.
Objective 4: Facilitate decommissioning	The RWDS is designed with minimum embedded piping, as much as practicable. Drain lines are routed through pipe chases as much as practicable. Piping between buildings (RXB, RWB, and ANB) is routed through the pipe chases provided between the buildings.
	The instruments downstream of the sump pumps have diaphragm seal design to prevent instruments from becoming contaminated. This feature reduces the quantities of contaminated equipment requiring disposal during the decommissioning period.
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 Reactor Building

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The RXBS equipment rooms are designed with curbs and shielded cubicles. This feature minimizes the spread of contamination.
	The welded stainless steel liner for the pool and dry dock minimize the potential of leakage.
	RXB floors are sloped to facilitate the collection of floor surface drainage and direct it to the nearest RWDS drain hub.
	The RXB is designed with a minimum number of structural joints to prevent contamination of the environment and provide protection against ground water intrusion. The RXB basemat is placed monolithically.
Objective 2: Provide leak detection capability	See Objective 6.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The equipment rooms that contain radioactive fluids are designed with curbs or walls to prevent leaks from spreading into other areas.
	The RXB is designed with a concrete tunnel between the RXB and RWB to provide a pathway for contaminated lines, to minimize the spread contamination. The concrete tunnel collects leaks to RWDS, is inspectable, and prevents ground water intrusion.
	There are no subgrade penetrations through the RXB exterior walls.
Objective 4: Facilitate decommissioning	Surfaces with the potential for contamination are protected with an epoxy coating to minimize contamination of concrete and facilitate 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 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 Reactor Building, Central Utility Building, North and South Turbine Generator Buildings, and the Auxiliary Boiler Building. The SCWS piping that is underground is reinforced or pre-stressed concrete pressure pipe designed to American Water Works Association standards. The SCWS piping that is above ground is made of carbon steel designed to ASME B31.1. These designs are compatible with the operating environment of the SCWS. This minimizes corrosion and reduces the potential for leaks. This also applies to Objective 3.
	The SCWS is equipped with a chemical feed system to add biocides, despersants, corrosion and scale inhibitors to minimize system degradation and potential leakage.
Objective 2: Provide leak detection capability	The SCW system is provided with process radiation monitors downstream of heat exchangers for the RCCWS, spent fuel pool cooling system (SFPCS), and RPCS, and on the cooling tower overflow and blowdown lines to the utility water system (UWS) discharge basin. These radiation monitors are provided to detect cross contamination due to leaks from heat exchangers.
	Grab sampling capability is also provided to measure radiation contamination levels in the system.
Objective 3: Reduce contamination to minimize releases, cross-contamination & waste generation	The SCWS components are selected for reliable service for the life of the plant.
Objective 4: Facilitate decommissioning	The SCWS is provided with both process radiation monitors and sampling provisions to monitor for water quality and contamination and alert operators to take corrective measures. This approach will minimize the contamination levels of the SCWS and facilitate decommissioning. This also applies to Objective 3.
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 Spent Fuel Pool Cooling System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The six-inch curb around the SFPCS pumps, strainers, and heat exchangers in the RXB is designed to contain leaks and spills. This also applies to Objective 3.
	The SFPCS piping is designed according to ASME B31.1 and uses stainless steel piping, valves, pumps, and wetted parts of the heat exchangers to resist corrosion and reduce the potential for leaks.
	The RXB floors are sloped and coated with epoxy to collect surface drainage and direct it to the RWDS drain sump. This also applies to Objective 3.
Objective 2: Provide leak detection capability	The area radiation monitor in proximity of the SFPCS heat exchanger area detects high radiation conditions, warning operators of abnormal condition. This feature minimizes personnel exposure.
	Although the site cooling water system is normally at a higher pressure than SFPCS, radiation detectors are provided in the SCWS to detect RPCS heat exchanger tube leaks into the SCWS.
	The RWDS sumps located in the RXB are equipped with the level transmitters to detect SFPCS leaks.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The concrete floors will have special coating (epoxy) to prevent spills penetrating the concrete and creating permanent fixed contamination.
	The SFPCS is designed with a minimum of two barriers between the SFPCS and the clean systems to prevent cross-contamination.
	The SFPCS equipment vents and drains are connected to the equipment drain hub to minimize contamination of floors.
	The SFPCS interfaces with the PCUS for the cleanup of the pool water to reduce contamination levels.
Objective 4: Facilitate decommissioning	The SFPCS components are designed for the life of the plant and are designed, to the extent practicable, as discreet assemblies to facilitate decommissioning.
	The SFPCS components are above ground and embedded pipe is minimized. This design prevents ground contamination and facilitates decommissioning.
Objective 5: Operating programs and documentation	COL item
Objective 6: Site radiological environmental monitoring	COL item

Table 12.3-41: 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 SRWS tanks, piping, and valves are designed with a corrosion resistance material in accordance with code and standards from RG 1.143 to minimize degradation over the life of the plant.
	The welded construction of components and piping minimize of the potential for failures and leaks as compared with flanged joints.
	Relief valves on each tank protect the tanks from over pressurization. The RV discharge lines are routed close to the floor to prevent contaminating the tank exterior surfaces, cubicle walls and personnel who work in the area.
	The SRWS equipment (tanks and pumps) are placed in a separate cubicles to limit the spread of contamination from leaks. Each SRWS storage tank cubicle has stainless steel lined walls to contain the contents of the tank.
	The spent cartridge filter transport cask is equipped with a drip pan to capture drips from the spent filter during transport.
Objective 2: Provide leak detection capability	The resin storage tanks are equipped with two level transmitters to detect level changes in the tanks associated with the tank leakages.
	The dewatering fill head is equipped with a level transmitter, level control valve, and a CCTV camera on the fill head for monitoring high integrity container (HIC) level during the filling process.
	The area radiation monitors near the SRWS components can assist in detecting potential leaks and spills.
	The waste storage areas and the radioactive waste building crane are designed with remote cameras, which provide monitoring capability.
	RWDS sumps are equipped with level transmitters to detect liquid accumulation.
Objective 3: Reduce contamination to minimize releases, cross-contamination and waste generation	The slurry lines are sized to ensure sufficient velocity to prevent resins from settling in the piping system during resin transfer. The piping is designed with five-diameter bend elbows to reduce fluid or slurry traps, thus reducing contamination and waste generation.
	The SRWS resin storage tanks are designed with conical bottoms to reduce the collection of resins or sludge.
	Connections from normally clean utility systems are designed with a minimum of two barriers to prevent cross-contamination from potentially radioactive systems.
	The HEPA filter downstream of the dewatering fillhead and the compactor capture radioactive particles prior to being discharged to the RWBVS.
	The internal resin screens on the vent line, pump suction, and break pot tank prevent contaminated resin from entering the RWBVS or pump suction.
	Resin sluice operations using resin storage tank decant water minimizes waste generation as compared to feed and bleed method using clean water.
	Three pump suction connections on the side of each resin storage tank minimize use of demineralized water needed for sluicing operations.
	The slurry lines are provided with full port ball valves to reduce crud buildup and resin traps.
	The level or pressure transmitters are designed with diaphragm type instrument lines.
	The top sluicing tank design reduces line blockage or hot spots in the resin transfer lines during resin transfers to a HIC. The line drains back into the storage tank by gravity when the transfer is terminated.

Table 12.3-41: Regulatory Guide 4.21 Design Features for Solid Radioactive Waste System (Continued)

Objective	Design Features
Objective 4: Facilitate decommissioning	The SRWS equipment is designed for the life of the plant and designed, to the extent practicable, as discreet assemblies to facilitate decommissioning.
	The SSC are designed with decontamination capabilities using the CIP skids. Design features, such as welding techniques and surface finishes are included to facilitate decontamination and minimize waste generation.
	The instruments that interface with contaminated fluid or slurry are designed with diaphragm seals to reduce decontamination requirements during decommissioning.
	The SRWS is designed with above ground piping, to the extent practical.
Objective 5: Operating programs and documentation	COL item
Objective 6: Site radiological environmental monitoring	COL item

Table 12.3-42: Regulatory Guide 4.21 Design Features for Ultimate Heat Sink System

Objective	Design Features
Objective 1: Minimize the potential for leaks or 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 minimizes the potential for leaks. This applies to objective 3 for cross contamination.
	The UHS uses stainless steel piping, valve, and pool liner material to reduce the potential that corrosion will result in leaks. This feature also applies to Objective 3.
	The UHS components are welded to minimize of the potential for leakage and spreading contamination to the plant or environment. 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	The UHS components are designed for the life of the facility using nuclear industry-proven materials compatible with the operating environment.
	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 welded, stainless steel surfaces for easier decontamination.
Objective 5: Operating programs and documentation	COL item
Objective 6: Site radiological environmental monitoring	COL item

Table 12.3-43: 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	The UWS above ground piping is constructed of coated carbon steel or equivalent materials compatible with the operating conditions and designed in accordance with ASME B31.1. The UWS underground piping is reinforced, pre-stressed, or both, concrete pressure pipe, as applicable, designed to the American Water Works Association standards. Using proven material in accordance with applicable codes and standards minimizes the potential for leaks.
Objective 2: Provide leak detection capability	The UWS discharge basin outfall is the monitored single point liquid effluent release path to the environment and includes a radiation monitor, plus sampling provisions.
Objective 3: Reduce contamination to minimize releases, cross-contamination & waste generation	The UWS components are selected for reliable service for the life of the plant.
Objective 4: Facilitate decommissioning	The portion of the UWS that may become contaminated is the discharge basin, limiting the amount of equipment needed to be decontaminated during decommissioning.
Objective 5: Operating programs and documentation	COL item
Objective 6: Site radiological environmental monitoring	COL item

Table 12.3-44: Regulatory Guide 4.21 Design Features for Demineralized Water System

Objective	Design Features
Objective 1: Minimize the potential for leaks or spills and provide containment areas	The DWS tanks and piping system are designed and fabricated according to industry codes and standards. This 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 will result in leaks. This feature also applies to Objective 3.
	The DWS tanks are welded to minimize of the potential for leakage and spreading contamination to the plant or environment. This feature also applies to Objective 3.
Objective 2: Provide leak detection capability	Samples are routinely taken from the DWS to test for contaminants that may have leaked into the DWS.
	The DWS incorporates radiation monitors to detect the backflow of contamination into the DWS.
Objective 3: Reduce contamination to minimize releases, cross-contamination, and waste generation	The DWS includes backflow preventers at the connections to systems throughout the plant to minimize the probability of contaminating the DWS distribution system. The DWS supply headers are segregated based on supplying either clean or contaminated systems.
Objective 4: Facilitate decommissioning	The DWS components are designed for full service life 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

Figure 12.3-1a: Reactor Building Radiation Zone Map - 24' Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1b: Reactor Building Radiation Zone Map - 35'-8" Elevation

{{ Withheld - See Part 9 }}

Figure 12.3-1c: Reactor Building Radiation Zone Map - 50' Elevation

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Figure 12.3-1d: Reactor Building Radiation Zone Map - 62' Elevation

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Figure 12.3-1e: Reactor Building Radiation Zone Map - 75' Elevation

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Figure 12.3-1f: Reactor Building Radiation Zone Map - 86' Elevation

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

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

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Figure 12.3-1i: Reactor Building Radiation Zone Map - 146' Elevation

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Figure 12.3-2a: Radioactive Waste Building Radiation Zone Map - 71' Elevation

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

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Figure 12.3-3: NuScale Power Module Monte Carlo N-Particle Shielding Model

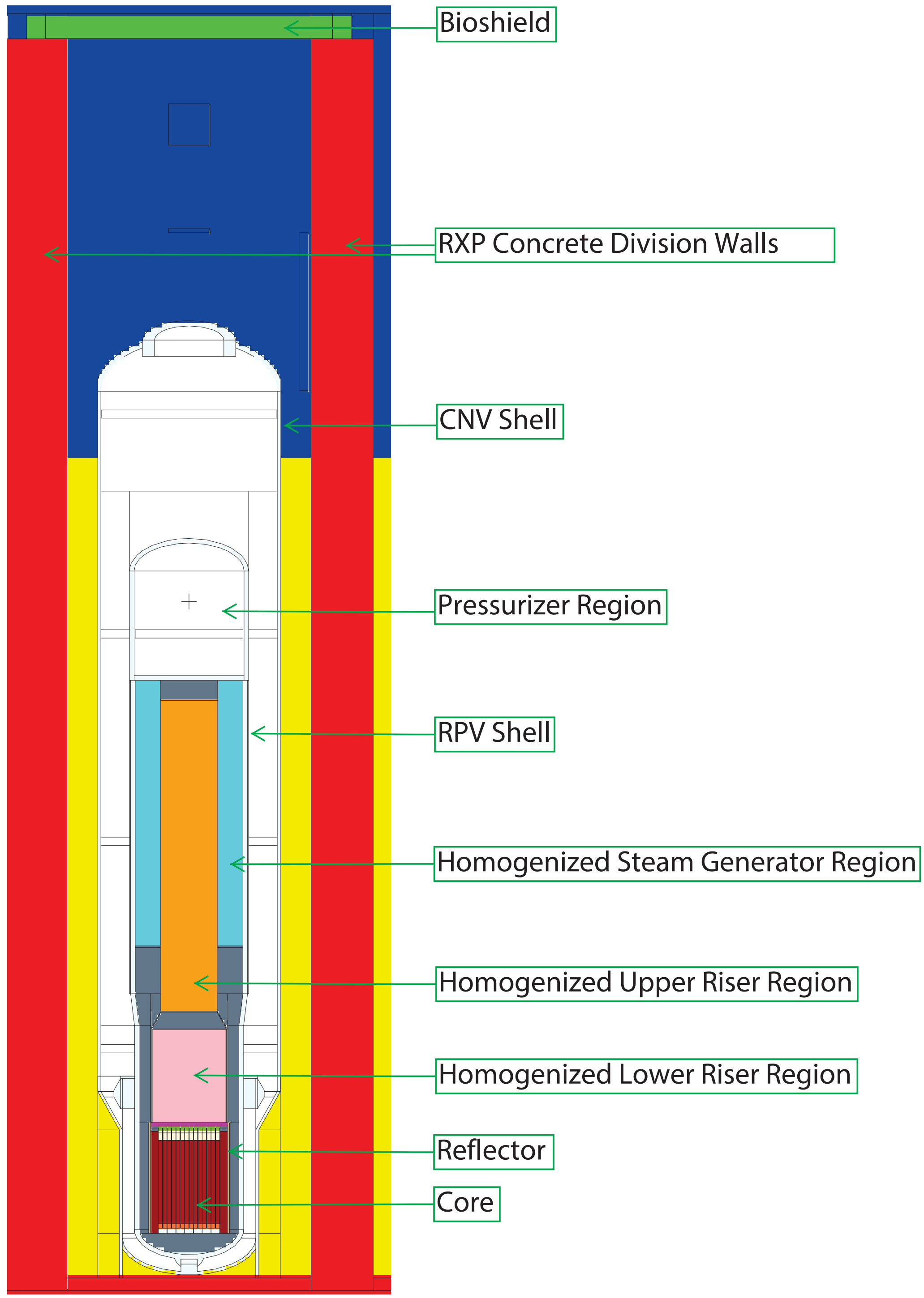

Figure 12.3-4a: Not Used

Figure 12.3-4b: Not Used

Figure 12.3-4c: Not Used

Figure 12.3-4d: Not Used

12.4 Dose Assessment

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

The dose assessment process is integrated into the overall ALARA program during the design of the NuScale Power Plant facility to help maintain occupational radiation exposures as low as reasonably achievable. To estimate the occupational radiation exposures for the NuScale facility, various work activities and work durations are compiled along with the expected significant (>0.1 mrem/hr) radiation fields that would be encountered.

12.4.1 Occupational Radiation Exposure

Radiation exposures to plant operating personnel are determined in accordance with the methods described in RG 8.19. The radiation protection features described in Section 12.3, together with the health physics program discussed in Section 12.5, maintain operator occupational radiation exposure as low as reasonably achievable. The airborne concentrations in various parts of the facility are provided in Section 12.2.

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

The estimated annual occupational radiation exposures are calculated from the following activity categories:

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

The total estimated annual occupational personnel doses are summarized in Table 12.4-1.

Details for each work activity assessed are discussed below. 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.

12.4.1.1 Reactor Operations and Surveillance

During plant operations, systems and components are monitored for performance and operating condition. This monitoring includes operator rounds during each shift. 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 the calculated values of the collective doses for reactor operations and surveillances.

12.4.1.2 Routine Inspection and Maintenance

Routine inspection and maintenance activities are performed for plant components during plant operations. These routine 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 lists work activities and their respective doses for routine inspection and maintenance.

12.4.1.3 Inservice Inspection

Periodic inservice inspections are required to be performed on 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.

Detailed listings of the doses associated with major inservice inspection activities are provided in Table 12.4-4.

12.4.1.4 Special Maintenance

Special maintenance consists of activities that go beyond routine scheduled maintenance. This includes 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. Special maintenance activities for each NPM are assumed to occur once every two years, based on a two-year refueling cycle.

Special maintenance activities consist of the following:

- control rod drive mechanism electromagnetic coil replacement
- in-core instrumentation maintenance

- instrumentation calibration and maintenance
- valve maintenance or replacement
- valve position indicator calibration
- steam generator tube cleaning and plugging
- pressurizer heater replacement

Table 12.4-5 provides the estimated doses due to special maintenance operations. The estimated dose is for up to a 12-NPM site with six refueling outages per year.

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 operation, waste maintenance, and other activities.

Waste operation 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.

Estimated annual doses from waste processing operations are listed in Table 12.4-6.

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 major activities included in the dose assessment for refueling activities include:

- 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

Occupational doses are estimated for a single NPM refueling outage and for an entire year, assuming six NPM refueling outages. Table 12.4-7 provides dose estimates for the various refueling activities.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the activities discussed above are summarized in Table 12.4-1.

Occupational personnel dose estimates are calculated assuming a 12-NPM site and 24-month fuel cycle for NPM operation, which equates to six refueling outages per year.

12.4.1.8 Post-Accident Actions

There are no vital areas, as defined by NUREG-0737, Item II.B.2, other than the areas for initiating combustible gas monitoring (described in Section 9.3.2.2.3), the main control room, and the technical support center, which are in compliance with 10 CFR 50.34(f)(2)(vii). There are no credited post-accident operator actions outside of the main control room for design basis events, as described in Chapter 15. The operator dose assessments for the main control room and the technical support center are provided in Section 15.0.3.

12.4.1.9 Construction Activities

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

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

12.4.2 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	Percent of Total	Estimated Annual Dose (man-rem)
Reactor operations & surveillance	8%	2.6
Routine maintenance & inspections	7%	2.2
Inservice inspection	15%	5.0
Special maintenance	30%	10
Waste processing	4%	1.4
Refueling	35%	11.5
Total	**100%**	**33**

Note: Estimates assume a plant with 12 NPMs on a two year refueling cycle.

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

Activity Category	Average Dose Rate (mrem/hr)	Exposure Time (man-hr/year)	Estimated Annual Dose (man-rem)
Security patrols	0.08	380	0.03
Calibrations	0.31	4300	1.3
Health Physics surveys	0.19	3300	0.64
Chemistry	0.03	2400	0.08
Surveillance	0.19	1900	0.36
Leak identification	2.0	78	0.16
Operations	0.06	160	0.01
Total	**0.21**	**12,000**	**2.6**

Note: Estimates assume a plant with 12 NPMs on a two year refueling cycle.

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

Activity	Average Dose Rate (mrem/hr)	Exposure Time (man-hr/year)	Estimated Annual Dose (man-rem)
24′ Elevation	1.6	50	0.08
35′-8″ Elevation	1.6	380	0.61
50′ Elevation	0.96	780	0.75
62′ Elevation	1.8	35	0.06
75' Elevation	0.09	5	0.0004
100′ Elevation	0.18	2300	0.42
Hot machine shop	2	160	0.32
Total	**0.61**	**3700**	**2.2**

Note: Estimates assume a plant with 12 NPMs on a two year refueling cycle.

Table 12.4-4: Occupational Dose Estimates from Inservice Inspection

Activity	Average Dose Rate (mrem/hr)	Exposure Time (man-hr/year)	Estimated Annual Dose (man-rem)
NPM preparation work	0.69	3170	2.19
Steam generator	8.7	147	1.28
Containment vessel side	5.0	8.22	0.04
Reactor pressure vessel side	32	26.0	0.83
Containment vessel head	5.0	15.9	0.08
Reactor pressure vessel head	20	33.5	0.67
Total	**1.48**	**3400**	**5.0**

Note: Estimates assume a plant with 12 NPMs on a two year refueling cycle.

Table 12.4-5: Occupational Dose Estimates from Special Maintenance

Inspection Location	Average Dose Rate (mrem/hr)	Annual Time (man-hrs)	Estimated Annual Dose (man-rem)
Control rod drive mechanism	12	385	4.6
In-core instrumentation	20	108	2.2
Instrumentation	8	192	1.5
Valves	9	72	0.65
Valve position indicators	7.4	69	0.51
Steam generators	7.7	59	0.46
Pressurizer heaters	14	12	0.17
Totals	**11**	**897**	**10**

Note: Estimates assume a plant with 12 NPMs on a two year 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 Operation	0.39	1300	0.51
Waste Maintenance	0.54	370	0.20
Other Waste Activities	0.30	2200	0.66
Total	**0.35**	**3900**	**1.4**

Note: Estimates assume a plant with 12 NPMs on a two year refueling cycle.

Table 12.4-7: Occupational Dose Estimates from Refueling Activities

NPM Refueling Activity	Activity Time (man-hrs)	Estimated Annual Dose (man-rem)
Prepare NPM for movement	105	0.28
Disconnect and move NPM to containment vessel flange tool	46	0.10
Disassemble NPM	178	0.03
Complete lower containment vessel work	20	0.13
Refuel reactor	133.0	0.27
Reassemble and move module to operating bay	230	0.18
Reconnect NPM	196	0.83
Transition to operations	35	0.10
Occupational radiation exposure for one NPM refueling outage	**943**	**1.92**
Annual Refueling occupational radiation exposure (6 NPMs)	**5660**	**11.5**

Note: Estimates assume a plant with 12 NPMs on a two year refueling cycle.

Table 12.4-8: Not Used

12.5 Operational Radiation Protection Program

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