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DESIGN CONTROL DOCUMENT TIER 2

CHAPTER 12
RADIATION PROTECTION

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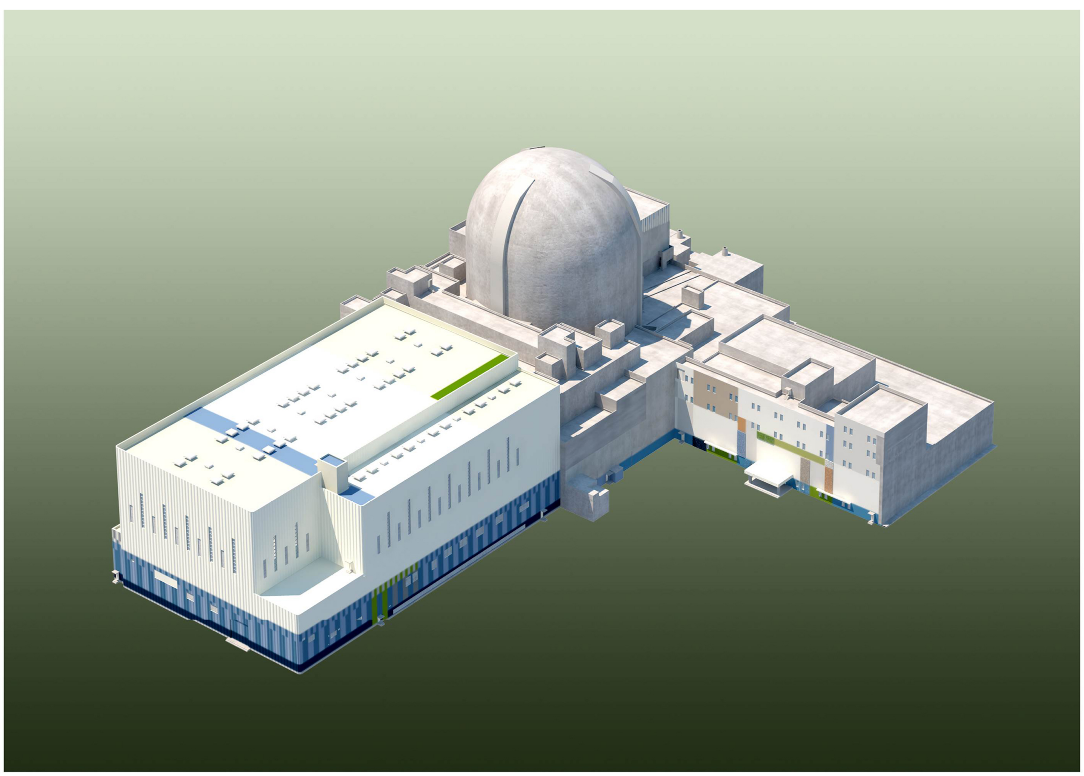

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KOREA HYDRO & NUCLEAR POWER CO., LTD

CHAPTER 12 – RADIATION PROTECTION

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ACRONYM AND ABBREVIATION LIST

AB	auxiliary building
AC	alternation current
AFW	auxiliary feedwater
APR1400	Advanced Power Reactor 1400
AST	alternative source term
BOC	beginning of cycle
CCW	component cooling water
CEACP	control element assembly change platform
CEA	control element assembly
CEDE	committed effective dose equivalent
CLVPS	containment low-volume purge system
COL	combined license
CPIAS	containment purge isolation actuation signal
CS	containment spray
CSS	containment spray system
CVCS	chemical and volume control system
DBA	design basis accident
DCF	dose conversion factor
DDE	deep dose equivalent
DE	dose equivalent
DF	decontamination factor
EAB	exclusion area boundary
EDE	effective dose equivalent
EFPD	effective full-power day
EOC	end of cycle
ESF	engineered safety features
ESFAS	engineered safety features actuation system
FF	flash fraction
GDC	general design criteria

HEPA	high-efficiency particulate air
HI	hydrogen igniter
HVAC	heating, ventilating, and air conditioning
ICRP	International Commission on Radiological Protection
IRWST	in-containment refueling water storage tank
LOCA	loss-of-coolant accident
MCR	main control room
MSIV	main steam isolation valve
NRC	United States Nuclear Regulatory Commission
ORE	occupational radiation exposure
PA	postulated accident
POSRV	pilot operated safety and relief valve
PTS	primary-to-secondary
PWR	pressurized water reactor
RADTRAD	radionuclide transport, removal, and dose
RCFC	reactor containment fan cooler
RCGVS	reactor coolant gas vent system
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RMS	radiation monitoring system
RV	reactor vessel
SCS	shutdown cooling system
SFP	spent fuel pool
SG	steam generator
SGBS	steam generator blowdown system
SI	safety injection
SIS	safety injection system
TEDE	total effective dose equivalent
TSC	technical support center

CHAPTER 12 – RADIATION PROTECTION

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

This chapter describes the radiation protection measures and operating policies for the APR1400 that meet the radiation protection standards in 10 CFR Part 20 and U.S. Nuclear Regulatory Commission (NRC) guidance related to protection from radiation, including the internal and external radiation that plant operating and construction personnel and the public are potentially exposed to during normal operation, including anticipated operational occurrences (AOOs). The combined license (COL) applicant is to supplement the radiation protection measures and operating policies by developing site-specific procedures and programs related to radiation protection.

"Radiation protection" and "health physics" are used interchangeably in this chapter.

The radiation protection measures in the design include the separation of radioactive components into individual shielded compartments, shielding that is designed to adequately attenuate radiation emitted from the components and piping that are sources of significant ionizing radiation, remote operation of equipment, ventilation of equipment areas with the potential to have airborne radiation, permanent radiation monitoring systems (RMSs), training of personnel in radiation protection, and administrative policies and procedures to maintain exposures ALARA.

The radiation protection program for the APR1400 associated with maintaining occupational exposure to radiation as low as reasonably achievable (ALARA) is described below, and conforms with Nuclear Energy Institute (NEI) 07-08A (Reference 1).

12.1.1 Policy Considerations

The design policy of the APR1400 is to maintain occupational radiation exposure ALARA to operating and construction personnel and to the public. The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and regulatory guides including NRC Regulatory Guides (RGs) 1.8 (Reference 2), 1.33 (Reference 3), 8.8 (Reference 4), and 8.10 (Reference 5) (COL 12.1(1)). The ALARA policy is applied to the total exposures (person-Sv) received by all operation and construction personnel, and to individual exposures. The proper approach and awareness of potential problems related to radiation protection are addressed through the proper training of all plant personnel as provided in NRC RG 1.8.

12.1.1.1 Design Policies

The design policy of the APR1400 is to implement the ALARA philosophy during the early stages of the design by reviewing and documenting the ALARA design to provide reasonable assurance of consistency with the applicable ALARA design criteria. The design policy is based on experience, including lessons learned, which indicates that the most effective design approach is to implement ALARA criteria during the early design stages to minimize the need for design changes in later stages.

During the early development of the design, the system engineers and designers are trained in the ALARA design principles and in radiation protection measures through procedure review assignments and periodic group training. ALARA design reviews are periodically conducted throughout the design and construction phases to update and modify the design based on the ongoing development of the plant’s detailed design and lessons learned from operating plants. The reviews are conducted to integrate the layout, shielding, ventilation, and radiation monitoring design with security, access control, maintenance, and inservice inspection (ISI) to provide reasonable assurance that occupational exposures are ALARA.

Onsite inspections are conducted during construction to provide reasonable assurance that the shielding, components, and piping layout are built in accordance with the established design. During construction, ALARA-experienced personnel conduct visual inspections to provide reasonable assurance that there are no instances of uncontrolled radiation, such as scattering and unforeseen streaming paths due to the locations of piping penetrations in the shield walls as they are installed.

Subsections 12.1.2 and 12.1.3 describe the criteria and methods used in the design and operation of shielding, ventilation, and radiation monitoring systems (including equipment and plant arrangements and access control provisions) to keep occupational exposures ALARA.

The design of the radiation protection measures is reviewed by qualified engineers as a normal part of the plant design process. The personnel responsible for ALARA design and review are engineers who are competent in the application of radiation protection principles, including radiological dose assessment, shielding design, and radwaste system design.

The radiation protection team reviews operating plant exposures and designs to establish criteria and guidelines for radiation protection. The team reviews various aspects of the plant design to ascertain conformance with the established criteria of maintaining exposures ALARA and recommends design modifications as necessary. During construction, field changes submitted to the plant designer are reviewed by radiation protection engineers to provide reasonable assurance that occupational exposures are ALARA. The reviews meet the intent of NRC RG 8.8.

12.1.1.2 Operation Policies

The APR1400 design is supplemented by operation policies and programs that are intended to keep occupational exposure to radiation ALARA while satisfying the applicable regulations. The program consists of written procedures for executing maintenance, testing, and inspection activities in radiation areas and training for plant personnel in radiation protection fundamentals that is conducted periodically and reviewed by plant management.

The general manager of the radiation protection team, who is responsible for the initial establishment of the health physics program, reports to the plant manager and exercises supervisory control over the radiation protection team. Radiation protection technicians are responsible for radiation surveys, contamination surveys, and environmental sampling.

The functions of the radiation protection team include the following:

a. Prepare and approve detailed procedures for radiation protection before plant operation begins and provide reasonable assurance that dose limits are established in accordance with the approved health physics program and all applicable regulations

b. Provide reasonable assurance that radiation exposure to plant personnel is maintained ALARA by:

    1) Evaluating radiological conditions in areas to be accessed by personnel and taking precautionary measures to provide reasonable assurance of minimal exposure

2) Controlling personnel and equipment movement into and out of radiation control areas

3) Providing instruction and supervision for the proper use and care of special protective clothing and equipment

4) Posting each radiation control area with conspicuous and appropriate caution signs, and access control means (locking devices, rope) in accordance with the applicable regulations

5) Administering and controlling conditions of radiation work permits for work in areas with high radiation or contamination levels in accordance with the approved procedures and the radiation protection program

c. Determine procedural requirements for the use of personnel monitoring devices and maintain records of personnel exposure in accordance with the applicable regulations

d. Control and document all radioactive material entering or leaving the plant site in accordance with the applicable regulations

e. Establish procedures for dealing with emergency conditions

f. Train plant staff and visitors in the radiation protection policies and procedures, as required

g. Include a dosimetry program as part of the site health physics program. The program includes whole-body counting to measure the uptake of radiation by personnel

h. Conduct periodic calibration for process radiation, area radiation, portable radiation, and airborne radioactivity monitors

i. Monitor and maintain records of all radioactive effluents released from the station

j. Collect data required in the evaluation of radiological effects from plant operation

12.1.2 Design Considerations

This section describes the general design features included in the plant and system design to implement the policies presented in Subsection 12.1.1. Provisions and design features for maintaining personnel exposures ALARA are presented in Section 12.3.

12.1.2.1 General Design Considerations for Maintaining Occupational Exposures ALARA

Design guidance to maintain occupational exposures ALARA is provided in the APR1400 ALARA Design Guide, which describes the design approach and operational implementation to reduce radiation exposure to plant personnel, including maintenance personnel. The APR1400 ALARA Design Guide is consistent with the recommendations in NRC RG 8.8 and NRC RG 8.10. It also incorporates lessons learned from earlier nuclear plant designs and operating experience that are obtained by reviewing and evaluating applicable operating experience, updated industry practices, and implementation of improvement through internal lessons-learned programs.

The APR1400 ALARA Design Guide also provides the procedure for training the responsible design engineers in each design discipline including the process system; plant layout; piping; heating, ventilating, and air conditioning (HVAC); instrumentation and control (I&C); and civil/structure during the early stage of the plant design through periodic procedure review and documented group training sessions. The independent ALARA review process is performed by experienced radiation protection engineers to provide reasonable assurance that the ALARA principles are implemented.

Experience from existing nuclear power plant designs and operating pressurized water reactor (PWR) plants in Korea and the United States is incorporated into the APR1400 design process to provide reasonable assurance that occupational radiation exposure (ORE) is maintained ALARA. Because Korea Hydro & Nuclear Power Co., Ltd. (KHNP) has constructed or is constructing and operating all of the Korean nuclear power plants, the feedback and lessons learned from past construction and operation are documented and incorporated in the design of the new plant. A lessons-learned program is used so that NRC generic communications and operating and construction experience are reviewed, assessed, and implemented as appropriate.

The ALARA design criteria in the APR1400 ALARA Design Guide are established to accomplish the following design objectives:

a. To reduce the duration and/or frequency of required equipment maintenance through the use of industry-proven equipment and construction materials that are commensurate with the intended plant processes and services

b. To reduce radiation levels through the use of operational controls such as frequent flushing and cleaning of equipment and the use of localized shielding where needed

c. To reduce the time spent in radiation areas where maintenance and other operational activities are necessary by providing easy access to components that require frequent maintenance and by moving components to low-radiation zones to perform maintenance when possible

The APR1400 design features that meet the above design criteria are described in Subsections 12.3.1.1 and 12.3.1.2.

The APR1400 is also designed to minimize the production, distribution, and retention of activated corrosion products throughout the reactor coolant system (RCS). The production of corrosion products is minimized by the proper selection of materials and the implementation of chemistry control of the primary system, as described in Subsection 12.3.1.3. Reactor coolant leakage is controlled to minimize the distribution and retention of corrosion products as described in Subsection 12.3.1.2(i).

In accordance with the requirements in 10 CFR 20.1406 (Reference 6), the APR1400 design implements the design approaches to minimize, to the extent practicable, contamination of the facility and the environment; facilitates eventual decommissioning; and minimizes, to the extent practicable, the generation of radioactive waste. The design considerations that conform with these requirements are described in Subsection 12.4.2.

12.1.2.2 Equipment Design Considerations for Maintaining Occupational Exposures ALARA

NRC RG 8.8 provides guidance for the selection of equipment that has an integral part in maintaining occupational exposures ALARA. In accordance with NRC RG 8.8, the selection of APR1400 equipment is based on (1) the effectiveness of contamination removal through the use of improved technologies, (2) a reduced need for equipment maintenance through the use of industry-proven equipment with demonstrated reliability and the use of preventive maintenance chemistry control, and (3) material properties that are suitable for minimizing corrosion. The use of materials containing cobalt is minimized to the greatest extent possible.

Components with the potential for crud accumulation are designed to prevent settlement of solids through the use of ellipsoidal bottoms with internal spargers and flushing using either demineralized water or the chemical decontamination system. The criteria used for equipment selection are provided in the APR1400 ALARA Design Guide.

The APR1400 ALARA design considerations with respect to the equipment design are provided in Subsection 12.3.1.2 and summarized as follows:

a. Enhanced reliability of equipment that reduces the frequency of maintenance and thus the occupational exposures to workers during maintenance

   The use of reliable lamps with extended service time in high-radiation areas reduces the need for frequent lamp replacement.

   Ion exchange and reverse osmosis are used to remove radioactive contamination. Both methods have resulted in higher contaminant removal and more reliable operations than processes using radwaste evaporators.

b. Enhanced environmental qualification of equipment

   Equipment qualification for a variety of environmental conditions, such as radiation, humidity, and temperature, is addressed in Section 3.11. Electrical components containing radiation-sensitive materials are shielded or located in low-radiation areas.

c. Material selection of piping and components

   Material with low cobalt and antimony impurities is selected to minimize the production of corrosion products that are a significant contributor to occupational exposures during maintenance and operational activities.

12.1.2.3 Facility Layout Design Considerations for Maintaining Occupational Exposures ALARA

The facility layout is designed so that the duration of time spent by workers within radiation areas for the performance of testing and maintenance activities is minimized.

Also, the separation of more highly radioactive equipment from less radioactive equipment is implemented to the greatest extent practicable in the layout design.

The plant layout is designed to maintain occupational exposures ALARA during normal operation including AOOs, post-accident conditions, and decommissioning as follows:

a. Adequate spacing to facilitate the accessibility of equipment during maintenance activities and to enable the installation of temporary shielding, where necessary, during the performance of testing and maintenance activities in radiation areas.

b. Separation of nonradioactive systems from radioactive systems to the extent practicable to minimize the spread of contamination.

c. Installation of system components into separate cubicles or compartments based on their respective frequency of maintenance, operational characteristics, level of radioactivity, and inclusion of passive versus active components.

d. Design of the ventilation systems so that airflows from clean areas to potentially contaminated areas to minimize the potential for the spread of airborne contamination. For areas with the potential for high airborne radioactivity, a slightly negative pressure is maintained.

Equipment and piping containing radioactive fluids are arranged in accordance with their contamination levels and the ALARA design criteria to provide reasonable assurance that equipment and piping expected to contain radiation sources are adequately shielded and properly routed to minimize occupational exposures. The APR1400 ALARA design features with respect to plant layout are described in Subsection 12.3.1.1 and are summarized as follows:

a. Tanks are located in individual cubicles to the extent practicable and based on their levels of contamination; shield walls are sized adequately and conservatively to attenuate radiation and maintain low dose rates in areas occupied by plant personnel.

b. Pumps and valve galleries are provided outside the cubicles containing the radioactive equipment to minimize operational exposures during operation and maintenance activities.

c. Instruments are located in low-radiation areas, and remote operation is implemented to the maximum extent practicable.

d. Piping that contains contaminated fluids is routed inside the pipe chases to the maximum extent practicable.

e. Shield wall penetrations are positioned to minimize radiation streaming from radioactive equipment to low-radiation areas.

12.1.3 Operational Considerations

The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational and public radiation exposures are ALARA (COL 12.1(2)). The APR1400 radiation protection design is implemented in conjunction with operational procedures and programs. Radiological health and safety procedures are developed and are continually reviewed and updated as necessary to provide reasonable assurance that occupational radiation exposures are ALARA. These procedures incorporate experience gained from plant operations. Improvements suggested during design and operation are incorporated and implemented to continually update the program to maximize protection of the plant personnel, the public, the environment, and the equipment.

The general manager of the radiation protection team is responsible for developing and updating radiological health and safety procedures. The plant manager supervises activities governed by these procedures so that workers are properly protected at all times. All personnel receive the training required to perform their assigned tasks safely.

12.1.3.1 General Operational Considerations for Maintaining Exposures ALARA

The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2 (Reference 7), 8.4 (Reference 25), 8.7 (Reference 8), 8.9 (Reference 9), 8.13 (Reference 10), 8.15 (Reference 11), 8.20 (Reference 12), 8.25 (Reference 13), 8.26 (Reference 14), 8.27 (Reference 15), 8.28 (Reference 16), 8.29 (Reference 17), 8.34 (Reference 18), 8.35 (Reference 19), 8.36 (Reference 20), and 8.38 (Reference 21) (COL 12.1(3)).

12.1.3.2 Design Features for ALARA During Maintenance and Inspection

The APR1400 design incorporates lessons learned from past designs, as well as the recommendations from NRC RGs 8.8 and 8.10. Design features to provide reasonable assurance that personnel exposure is maintained ALARA during maintenance and inspection activities are as follows:

a. Platforms are provided around the steam generator (SG) to facilitate the handling of the reactor coolant pump (RCP) seal cartridges.

b. Sufficient spacing is provided for ingress and egress, equipment laydown/pull areas, and removal paths for major equipment that is expected to be replaced during the plant lifetime.

c. Removable insulation is used.

d. The length of welds is reduced through the use of seamless piping as much as possible.

e. An integrated head assembly (IHA) is used.

High-radiation areas that house systems and components requiring ISI are designed to facilitate worker ingress and egress. The APR1400 is also designed with adequate spacing, including laydown areas, for ingress and egress between equipment (to facilitate maintenance), and spacing for inspections. Radiation protection technicians survey areas that will be accessed for maintenance and inspection prior to worker entry and conduct periodic inspections during work activities.

The APR1400 also provides the following design features to maintain personnel exposure ALARA during inservice inspection:

a. Pipe stops, snubbers, and pipe hangers near welds that require ISI or periodic maintenance are carefully positioned to facilitate weld accessibility.

b. Integrally forged components and seamless piping are selected whenever possible to minimize the need for inservice weld inspections.

c. Blanket-type thermal insulation with hook-and-loop fasteners is selected for components and piping containing radioactive material. This type of insulation can be removed easily by one worker for inservice weld inspection.

d. Adequate containment or ventilation is provided to minimize the potential for the spread of airborne contamination. Subsection 12.2.2 specifies the methodology for determining the in-plant concentration of airborne contamination. In Subsection 12.3.1.4, more information on airborne contamination control is described.

Following the radiation protection program, based on the guidance of NEI 07-03A (Reference 23), maintenance activities that could involve significant radiation exposure to personnel are carefully planned following the plant procedures to provide reasonable assurance of worker safety. Radiation work permits (RWPs) for routine and non-routine operations are issued for each activity to be performed in a high-radiation area. RWPs contain a list of the radiation protection requirements that are followed by all personnel working in a radiation area. Where practicable, radiation exposure reduction techniques, such as those in NRC RG 8.8, are evaluated and used.

12.1.3.3 Post-Accident Conditions

The layout and system design includes considerations for post-accident conditions. Separation and partitioning of components in individual shielded cubicles, provisions of adequate spacing, and ingress and egress design facilitate personnel access for surveillance and inspection, sample collection, cleanup, and repairs with minimum radiation exposure. The design also facilitates the use of temporary shielding, such as the use of lead blankets, to further minimize radiation exposure.

12.1.3.4 Decommissioning

The ALARA considerations for decommissioning are also incorporated into the system design and component layout through the provisions of individual shielded cubicles, adequate spacing, and ingress and egress design, and the partitioning and separation of low-contamination versus high-contamination areas. Other considerations for decommissioning include the provisions of decontamination capability to reduce residual contaminations, components designed with adequate capacities to reduce processing time, and the approach to maximize the use of integrated component assemblies to reduce stay time for dismantlement. In addition, decommissioning considerations for conformance with NRC RG 4.21 (Reference 22) and for applying to NEI 08-08A (Reference 24) are built into the design for systems and components that handle contaminated or potentially contaminated fluids. Decommissioning considerations are provided in Subsection 12.4.2.

12.1.4 Combined License Information

COL 12.1(1) The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and Regulatory Guides including NRC RGs 1.8, 1.33, 8.8, and 8.10.

COL 12.1(2) The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational and public radiation exposures are ALARA.

COL 12.1(3) The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2, 8.4, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.

12.1.5 References

1. NEI 07-08A, "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)," Nuclear Energy Institute, October 2009.

2. Regulatory Guide 1.8, "Qualification & Training of Personnel for Nuclear Power Plants," Rev. 3, U.S. Nuclear Regulatory Commission, May 2000.

3. Regulatory Guide 1.33, "Quality Assurance Program Requirements Operation," Rev. 3, U.S. Nuclear Regulatory Commission, June 2013.

4. Regulatory Guide 8.8, "Information Relevant to Ensuring the Occupational Radiation Exposures at Nuclear Power Stations will be ALARA," Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

5. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," Rev. 1-R, U.S. Nuclear Regulatory Commission, May 1977.

6. 10 CFR 20.1406, "Minimization of Contamination," U.S. Nuclear Regulatory Commission.

7. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring," Rev. 1, U.S. Nuclear Regulatory Commission, May 2011.

8. Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," Rev. 2, U.S. Nuclear Regulatory Commission, November 2005.

9. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," Rev. 1, U.S. Nuclear Regulatory Commission, July 1993.

10. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," Rev. 3, U.S. Nuclear Regulatory Commission, June 1999.

11. Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," Rev. 1, U.S. Nuclear Regulatory Commission, October 1999.

12. Regulatory Guide 8.20, "Applications of Bioassay for I-125 and I-131," Rev. 1, U.S. Nuclear Regulatory Commission, September 1979.

13. Regulatory Guide 8.25, "Air Sampling in the Workplace," Rev. 1, U.S. Nuclear Regulatory Commission, June 1992.

14. Regulatory Guide 8.26, "Applications of Bioassay for Fission and Activation Products," U.S. Nuclear Regulatory Commission, September 1980.

15. Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1981.

16. Regulatory Guide 8.28, "Audible-Alarm Dosimeters," U.S. Nuclear Regulatory Commission, August 1981.

17. Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure," Rev. 1, U.S. Nuclear Regulatory Commission, February 1996.

18. Regulatory Guide 8.34, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses," U.S. Nuclear Regulatory Commission, July 1992.

19. Regulatory Guide 8.35, "Planned Special Exposures," Rev. 1, U.S. Nuclear Regulatory Commission, August 2010.

20. Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus," U.S. Nuclear Regulatory Commission, July 1992.

21. Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants," Rev. 1, U.S. Nuclear Regulatory Commission, May 2006.

22. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

23. NEI 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program," Nuclear Energy Institute, May 2009.

24. NEI 08-08A, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," Nuclear Energy Institute, October 2009.

25. Regulatory Guide 8.4, "Personal Monitoring Device-Direct Reading Pocket Dosimeters," Rev.1, U.S. Nuclear Regulatory Commission, June 2011.

12.2 Radiation Sources

This section identifies the sources of radiation within the plant that form the basis for shielding design calculations and the sources of airborne radioactivity used for the design of personnel protection measures and dose assessment.

12.2.1 Contained Sources

The shielding design source terms are based on full-power operation with 0.25 percent fuel cladding defects and no credit for gas stripping. Sources in the primary coolant include fission products released from fuel cladding defects, activation products, and corrosion products. Throughout most of the RCS, activation products (principally nitrogen-16 [N-16]) are the primary radiation sources for shielding design during power operation.

12.2.1.1 Reactor Containment Building

12.2.1.1.1 Reactor Core

The primary radiation emanating from the reactor core during normal operation consists of neutrons and gamma rays. Tables 12.2-1 and 12.2-2 list neutron and gamma fluxes in the reactor cavity at the side of the reactor vessel (RV), respectively. These tables are consistent with the parameters presented in Chapter 4. Table 12.2-3 lists gamma fluxes outside the RV after shutdown. The flux data represent the typical values, not the design values.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the RCS are fission products released from the fuel and activation and corrosion products. The parameters used to calculate the reactor coolant fission product activities are listed in Table 12.2-4. The design basis reactor coolant activities are presented in Table 12.2-5. The reactor coolant fission product activities are calculated using Equations 11.1-1 and 11.1-2 (refer to Subsection 11.1.1.1).

Table 12.2-6 lists the average expected activities due to crud deposits on SG tubing and reactor system piping. The deposited crud activity is calculated using measured data from various operating PWRs and Equation 11.1-11 (refer to Subsection 11.1.3.1).

The activation product, N-16, is the predominant activity source in the RCPs, SGs, and reactor coolant piping. The N-16 activity in each of the components depends on the transit time from the reactor core outlet to the component.

N-16 is produced by the $O^{16}(n, p)N^{16}$ reaction. The threshold energy for the reaction is 10.2 MeV. N-16 decays by beta emission with subsequent high-energy gamma emission 74 percent of the time. The gamma energies are 6.13 MeV, 69 percent of the time and 7.12 MeV, 5 percent of the time. The half-life of N-16 is 7.13 seconds.

The N-16 activity at the RV outlet nozzle is 5.53×10^6 disintegrations/cm^3-sec. This activity, after considering the volume expansion of coolant at the vessel outlet nozzle, is calculated by the following expression and the reactor parameters. Table 12.2-7 lists typical N-16 activities at various locations in the primary coolant loop.

$$\text{Activity (disintegrations/cm}^3\text{-sec)} = \frac{\Sigma\Phi(1-e^{-\lambda t_c})e^{-\lambda t_r}}{(1-e^{-\lambda t_t})}$$

Where:

$\Sigma\Phi$ = reaction rate (5.02×10^7 reactions/cm^3-sec)

t_c = core transit time (0.81 sec)

t_t = total primary loop time (10.6 sec)

t_r = time from the active core outlet to the point of interest

λ = decay constant (0.097 sec^{-1})

The parameters and vapor activities in the pressurizer are listed in Table 12.2-29. The liquid activities in the pressurizer are assumed to be equal to the design basis reactor coolant activities in Table 12.2-5. The vapor activities of the pressurizer steam space are determined from the noble gases in Table 12.2-5. The vapor activities in the pressurizer are calculated using the following equation:

$$a_g = \frac{V_l}{V_g} \times \frac{m_g}{m_l} \times \left[\frac{M_{water} \times v_{steam}}{R \times T} \times H_{Xe\ or\ Kr}\right]^2 \times a_l$$

Where:

a_g, a_l = vapor and liquid activities, respectively, Bq

V_g, V_l = vapor and liquid volume, respectively, cm^3

m_g, m_l = vapor and liquid mass, respectively, gram

M_{water} = molecular weight of water, 18 gram/mol

ν_{steam} = specific volume of steam, cm^3/gram

R = ideal gas constant, 82.06 cm^3-atm/°K-mol

T = temperature, °K

$H_{Xe\ or\ Kr}$ = Henry's law constant for Xenon or Krypton, atm

12.2.1.1.3 Secondary-Side Systems

The rate of SG tube leakage is assumed to be 3,270 L/day (0.6 gal/min). This is assumed to be concurrent with 0.25 percent fuel cladding defects for the shielding design basis source term calculations. A blowdown rate of 0.2 percent of maximum steaming rate is assumed. No credit is taken for the operation of the condensate polisher demineralizers. Assumptions for calculation of main steam system (MSS) radiation sources are presented in Table 11.1-5, and results of the calculation are given in Table 12.2-19.

12.2.1.1.4 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant source of radiation in the reactor containment building (RCB) after plant shutdown for refueling. A reactor operating time to reach a near-equilibrium buildup level of fission products based on the rated power operation is used in determining the source strengths. The parameters used in the spent fuel decay gamma source calculation, such as fuel enrichment, specific core power, and discharge burnup, are given in Table 12.2-8. The spent fuel decay gamma source is given in Table 12.2-9. The neutron source term is not included because the dose rate contribution of neutron source term is negligible. Fuel assembly dose rates as a function of distance in the refueling pool water and time after shutdown are shown in Figures 12.2-1 and 12.2-2. The dose rates in the refueling pool are calculated by using neutron transport code, DORT (Reference 7), and flux-to-dose factors in ANSI/ANS 6.1.1 (Reference 8).

These figures are based on the gamma source terms in Table 12.2-9. These figures provide the information of dose rates to an operator during refueling operation.

12.2.1.1.5 Processing Systems

12.2.1.1.5.1 Chemical and Volume Control System

The shielding design for chemical and volume control system (CVCS) components is based on the maximum expected radioactivity in each of the components. The source term calculation is performed using SHIELD-APR (Reference 9) computer program. Parameters of the source term calculation in CVCS are listed in Table 12.2-30. The component source terms are given in Tables 12.2-10 through 12.2-14.

a. Heat exchangers

The radioactivity inventory for the CVCS heat exchanger is calculated using the following equations:

$A_{HX,i}=\frac{Q_i a_i}{\lambda_i}(1-e^{-\lambda_i t_R})$, for N-16

$A_{HX,i}=a_i \times V_{HX}$, for the other nuclides

Where:

$A_{HX,i}$ = activity in heat exchanger, Bq

Q_i = influent flowrate, cm^3/sec

a_i = influent specific activity, Bq/cm^3

λ_i = decay constant of a nuclide, sec^{-1}

t_R = transit time through heat exchanger, sec

V_{HX} = volume of heat exchanger, cm^3

The maximum values for the CVCS heat exchanger (HX) radionuclide inventories are presented in Table 12.2-10.

The total radioactivity inventory for the CVCS heat exchangers is based on the tube-side water volume (including the shell-side water volume for regenerative heat exchangers).

b. Ion exchangers

The radioactivity inventory for the CVCS ion exchangers is calculated using the following equations:

$A_{IX,i}=\frac{\eta Q_i a_i}{\lambda_i}(1-e^{-\lambda_i t})$, for DF≠1.0

$A_{IX,i}=a_i \times V_{IX,water}$, for DF=1.0

$A_{IX,i}=\frac{Q_i a_i}{\lambda_i}(1-e^{-\lambda_i t_R})$, for N-16

Where:

$A_{IX,i}$	=	activity in ion exchanger, Bq
Q_i	=	influent flowrate, cm^3/sec
a_i	=	influent specific activity, Bq/cm^3
η	=	removal efficiency of resin bed (=1-1/DF)
DF	=	decontamination factor
λ_i	=	decay constant of a nuclide, sec^{-1}
t	=	replacement period of ion exchanger resin, sec
t_R	=	transit time through ion exchanger, sec
$V_{IX,water}$	=	water volume in ion exchanger, cm^3

It is assumed that the porosity of resin is 0.5.

The maximum values for CVCS ion exchanger (IX) radionuclide inventories are presented in Table 12.2-11.

1) Purification ion exchanger

The total radioactivity inventory is based on resin buildup for the duration of 120 percent of one cycle of effective full-power days (EFPDs). This ion exchanger is used for removing lithium and purifying coolant water from RCS letdown. It is used for lithium removal for the first part of one-cycle EFPDs, which spans an average of 20 percent, and then it is put into service as a purification ion exchanger. The decontamination factor (DF) for anions is 100 with an efficiency of 99 percent. The DF for crud is 50 with an efficiency of 98 percent. The DF for all remaining nuclides except Xe, Kr, Y, Rb, and Cs is 50 with an efficiency of 98 percent. The DF for Xe, Kr, and Y is 1 with an efficiency of 0 percent. The DF for Rb and Cs is 2 with an efficiency of 50 percent. The radioactivity inventory in processed liquid is based on a normal letdown flow rate.

2) Deborating ion exchanger

The total radioactivity inventory is based on resin buildup during the end-of-cycle (EOC) deborating operation. This ion exchanger is used to reduce reactor coolant boron concentration at the EOC. Boron control in the CVCS is described in detail in Subsection 9.3.4. The DF for nuclides except anions and crud is 1 with an efficiency of 0 percent. The DF for anions and cruds is 10 with an efficiency of 90 percent. The radioactivity inventory in processed liquid is based on a normal letdown flow rate.

3) Pre-holdup ion exchanger

The total radioactivity inventory is based on resin buildup for the duration of one-cycle EFPDs. The DF for all nuclides except Xe, Kr, Y, Rb, and Cs is 10 with an efficiency of 90 percent. The DF for Rb and Cs is 100 with an efficiency of 99 percent; a 3-to-1 cation to anion ratio provides a minimum DF of 100.. The DF for Xe, Kr, and Y is 1 with an efficiency of 0 percent. Liquid inputs processed by the pre-holdup ion exchanger include the letdown processed through the purification ion exchanger and the purification filter, and the inflow from the reactor drain tank (RDT) and the equipment drain tank (EDT).

4) Boric acid condensate ion exchanger

The total radioactivity inventory is based on resin buildup for the duration of one-cycle EFPDs. The DF for anions is 10 with an efficiency of 90 percent. The DF for other nuclides including crud is 1 with an efficiency of 0 percent. The radioactivity inventory is based on input from the letdown processed through the purification ion exchanger and the purification filter, the inflow from the RDT and the EDT, and the inflow from the boric acid concentrator (BAC) in consideration of the average condensation rate.

c. Filters

The radioactivity inventory for the CVCS filters is calculated using the following equations:

$$A_{FL,i}=\frac{\eta Q_i a_i}{\lambda_i}(1-e^{-\lambda_i t}), \qquad \text{for DF}\neq 1.0$$

$$A_{FL,i}=a_i \times V_{FL}, \qquad \text{for DF}=1.0$$

$$A_{FL,i}=\frac{Q_i a_i}{\lambda_i}(1-e^{-\lambda_i t_R}), \qquad \text{for N-16}$$

Where:

$A_{FL,i}$ = activity in filter element, Bq

Q_i = influent flowrate, cm^3/sec

a_i = influent specific activity, Bq/cm^3

η = removal efficiency of filter element (=1-1/DF)

DF = decontamination factor of filter element

λ_i = decay constant of a nuclide, sec^{-1}

t = replacement period of filter element, sec

t_R = transit time through filter, sec

V_{FL} = filter volume, cm^3

The maximum values for CVCS filter radionuclide inventories are presented in Table 12.2-12.

The total radioactivity inventories in the CVCS filters are based on crud buildup for the duration of one-cycle EFPDs. The DF assumed for crud buildup in the filter is 10 with an efficiency of 90 percent.

d. Tanks

The radioactivity inventory for the CVCS tanks is calculated using the following equations:

For Liquid activity,

$$A_{L,i}=\frac{Q_i a_i}{\lambda_i+\frac{Q_i}{V_L}}\left(1-e^{-\left(\lambda_i+\frac{Q_i}{V_L}\right)t}\right)$$

For Noble Gases at equilibrium condition,

$$A_V=\frac{(18)(H)(V_V)(A_L)}{(V_L)(R)(T)}$$

For Noble Gases at non-equilibrium condition,

$$A_{V,i}=\left(\frac{PF}{PF+1}\right)\left(\frac{Q_i a_i}{\lambda_i}\right)\left(1-e^{-\lambda_i t}\right)$$

For Particulates,

$$A_V=\left(\frac{PF}{PF+1}\right)A_L$$

For new liquid activity (particulates and noble gases at non-equilibrium condition)

$$A_L=\left(\frac{1}{PF+1}\right)A_L$$

Where:

A_L, A_V = Liquid and Vapor activity, respectively, Bq

V_L, V_V = Liquid and Vapor volume, respectively, cm^3

Q_i	=	influent flowrate, cm^3/sec
a_i	=	influent specific activity, Bq/ cm^3
λ_i	=	decay constant of a nuclide, sec^{-1}
PF	=	partition factor
t	=	transit time through tank, sec
18	=	molecular weight of water, gram/mol
H	=	Henry's constant, atm
R	=	ideal gas constant, 82.06 cm^3-atm/°K-mol
T	=	temperature, °K

The liquid and vapor volumes are obtained from the low water level to maximize the VCT vapor source terms. The liquid source terms in VCT are additionally considered for the water volume from low water level to high water level. This maximizes the vapor and the liquid source terms in VCT.

The activity of the vapor space is determined initially by using Henry's law. For non-equilibrium condition, partition factors are used to re-evaluate the vapor space activity. In the case of particulates and non-equilibrium noble gases, a new liquid activity is calculated after accounting for partitioning.

The maximum values for CVCS tank radionuclide inventories are presented in Table 12.2-13.

The total radioactivity inventories in the CVCS tanks are based on buildup for the duration of one-cycle EFPDs.

e. Boric acid concentrator

The radioactivity inventory for the CVCS boric acid concentrator is calculated using the following equations. The influent specific activities of concentrate heater are assumed to be equal to the specific activities of holdup pump.

1) Concentrate Heater

$a_{heater,i}=a_i\left(\frac{1}{1+PF}\right)$, for noble gas

$a_{heater,i}=a_i\times CF_{BAC}$, for all nuclides except noble gas and N-16

$A_{heater,i}=a_{heater,i}\times V_{heater}$

Where:

$a_{heater,i}$ = specific activity of concentrate heater, Bq/cm^3

PF = partition factor for BAC heater (=10)

CF_{BAC} = BAC concentration factor (=100)

V_{heater} = BAC heater volume, cm^3

$A_{heater,i}$ = concentrate heater activity, Bq

2) Concentrate Cooler

$A_{cooler,i}=a_{heater,i}\times V_{cooler}$

Where:

V_{cooler} = concentrate cooler volume, cm^3

$A_{cooler,i}$ = concentrate heater activity, Bq

3) Flash Tank

The influent vapor specific activities of flash tank are assumed to be equal to the specific activities of holdup pump. The specific activities of BAC heater used to the influent liquid specific activities of BAC flash tank.

$A_{flash\ tank,i}=a_i\times V_{V,flash\ tank}$, for noble gas

$A_{flash\ tank,i}=a_{heater,i}\times V_{L,flash\ tank}$, for all nuclides except noble gas

Where:

$V_{V\ flash\ tank}$, $V_{L\ flash\ tank}$ = vapor and liquid volume in flash tank, resp ectively, cm^3

$A_{flash\ tank,I}$ = flash tank activity, Bq

4) Vapor Separator

$a_{separator,i}=a_i\left(\frac{PF}{1+PF}\right)$, for noble gas

$a_{separator,i}=a_i$, for all nuclides except noble gas

$A_{separator,i}=a_{separator,i}\times V_{V,separator}$, for noble gas

$A_{separator,i}=a_{separator,i}\times V_{L,separator}$, for all nuclides except noble gas

Where:

$a_{separator,i}$ = specific activity of vapor separator, Bq/cm^3

PF = partition factor for BAC vapor separator (=10)

$V_{V\ separator}$, $V_{L\ separator}$ = vapor and liquid volume in vapor separator, respectively, cm^3

$A_{separator,i}$ = vapor separator activity, Bq

5) Concentrate Pump,

$A_{conc\ pump,i}=a_{heater,i}\times V_{conc\ pump}$

Where:

$V_{conc\ pump}$ = concentrate pump volume, cm^3

$A_{conc\ pump,i}$ = concentrate pump activity, Bq

6) Concentrate Transfer Pump

$A_{trans\ pump,i}=a_{heater,i}\times V_{trans\ pump}$

Where:

$V_{trans\ pump,}$ = concentrate transfer pump volume, cm^3

$A_{trans\ pump,i}$ = concentrate transfer pump activity, Bq

The maximum values for BAC radionuclide inventories are presented in Table 12.2-14. The components of the boric acid concentrator (BAC) are conservatively modeled with the combined total source term based on the maximum design basis source terms in order to determine the minimum required shield wall thicknesses for the BAC room to maintain the dose rates in the adjacent rooms and areas below their respective radiation zone limits. The shielding analyses are performed for the vapor phase and liquid phase for the BAC components using their corresponding source terms as listed in Table 12.2-14. The dose rate results calculated for the two phases are summed for the determination of the minimum shield wall thicknesses and corresponding radiation zone for the BAC room.

The total radioactivity inventories in the BAC package are based on a concentration factor of 100.

12.2.1.1.5.2 Steam Generator Blowdown System

Radiation sources in the steam generator blowdown system (SGBS) are shown in Table 12.2-19. The sources are based on the assumed design basis primary-to-secondary (PTS) leakage rate and the assumed fuel defect percentage described in Subsection 12.2.1.1.3. The blowdown rate is assumed to be 1 percent of the maximum steaming rate. The nuclide accumulation in the SGBS pre-filter, post-filters, and the mixed beds are calculated based on radioactive crud and nuclide buildup at the end of 6 months of processing.

12.2.1.1.5.3 Condensate Polishing System

Radiation sources in the condensate polishing system (CPS) are shown in Table 12.2-19. The sources are based on the design basis PTS leakage and the assumed fuel defect percentage described in Subsection 12.2.1.1.3. It is assumed that 65 percent of the total steam flow rate and the SGBS blowdown flow through the CPS and that one pair (16-2/3 percent of the condensate flow) out of total seven pairs of CPS demineralizers, of which one pair of demineralizers is in standby service, is used to process the condensate during normal operation. The nuclide accumulation in the CPS cation bed and the mixed bed are based on radioactivity buildup at the end of their corresponding processing cycle. The nuclide accumulation times for the cation bed and the mixed bed ion exchangers are about 3 days and 30 days, respectively. The cation and mixed bed ion exchange resins are then replaced at the end of the processing times accordingly.

12.2.1.1.6 Gamma Sources of Irradiated Components

The components in the reactor vessel are irradiated by the fission neutrons during the core power operation and are activated. The in-core instrument (ICI) assembly, which consists of five rhodium detectors, one background detector, one core-exit thermocouple, and a central member assembly, is enclosed in a protective sheath. Activated gamma sources of the irradiated ICI assembly are estimated assuming 6 years of irradiation. The activated gamma sources of the irradiated control element assembly (CEA) and the irradiated neutron source assembly (NSA) are estimated assuming 10 years of irradiation. In CEA, the neutron absorbing material is B_4C and the cladding material is Inconel 625. The NSA contains the primary neutron source of Cf^{252} and the secondary neutron source of Sb-Be. The activated gamma source of the irradiated surveillance capsule assembly (SCA) is estimated assuming 60 years of irradiation. The irradiation calculation is performed using the ORIGEN-S computer program, and the irradiation time of each component is based on the design life time of each component. The activated gamma sources of ICI, CEA, NSA, and SCA are given in Table 12.2-15.

The handling and storage of the irradiated ICI and CEAs during refueling operation are described in Subsection 9.1.4.

Dimensions and parameters of the other radiation sources in reactor containment building used in the shielding analyses are listed in Table 12.2-27.

12.2.1.2 Auxiliary Building

12.2.1.2.1 Shutdown Cooling System

The shutdown cooling system (SCS) pumps, heat exchangers, and associated piping are considered to be contaminated with radioactive materials. During plant shutdown, radioactive isotopes carried in the reactor coolant after 4 hours of decay following shutdown are transported into the SCS pumps and heat exchangers. The maximum radioactive source strengths in the SCS are provided in Table 12.2-16.

12.2.1.2.2 Component Cooling Water System

The component cooling water system (CCWS) is a closed-loop demineralized water system that can potentially become contaminated by heat exchanger leakage from other radioactive systems that are cooled by the CCWS. The CCWS is provided with radiation monitors to detect inleakage of radioactive contaminants as described in Subsection 11.5.2.3. The detection of leakage is sensitive to a level of 3.7×10^{-2} Bq/cm^3 gross gamma activity in the CCWS cooling water.

In the event of a major leak in a CCWS train, the train is removed from service, a nonleaking train is used, and the contaminated water is treated through the liquid waste management system (LWMS). This design prevents the spread of contamination through the CCWS and connecting systems.

12.2.1.2.3 Spent Fuel Storage and Transfer

Spent fuel assemblies and associated crud are the primary sources of radiation in the spent fuel storage and transfer area. The shielding design assumes the maximum number of spent fuel assemblies in storage.

12.2.1.2.4 Spent Fuel Pool Cooling and Cleanup System

Activity levels in the spent fuel pool cooling and cleanup system (SFPCCS) are determined based on the activities present in the spent fuel pool (SFP). Design basis SFP activities are presented in Table 12.2-17.

SFP fission and corrosion product specific activities are evaluated in the initial stage of refueling. It is assumed that the reactor coolant is cooled down for approximately 2 days during a shutdown for refueling. During this period, the primary coolant is let down through the purification filter, purification ion exchanger, gas stripper, and volume control tank (VCT).

The purification process serves two purposes:

a. Removing the noble gases in the gas stripper avoids large activity releases to the reactor containment building following reactor vessel head removal.

b. The ion exchange and filtration reduces dissolved fission and corrosion product concentrations in the coolant that would otherwise enter the SFP and the refueling pool.

At the end of the 2-day period, the coolant above the reactor vessel flange is partially drained. The RV head is unbolted, and the refueling pool is filled with the water from the in-containment refueling water storage tank (IRWST). The remaining reactor coolant volume is then mixed with water in the refueling pool and the SFP.

After refueling, the SFP is isolated and the water in the refueling pool is returned to the IRWST. This series of events determines the total activity in the SFP. The fission and corrosion product activities in the SFP, listed in Table 12.2-17, are calculated using the reactor coolant equilibrium concentrations presented in Table 12.2-5. The SFP activities are subsequently reduced by decay during refueling as well as by operation of the SFPCCS.

The source terms for the SFP demineralizers and filters are provided in Table 12.2-18. The activities are integrated over the cleanup time for normal (expected) operation and are determined to be at maximum at about 265 hours and 290 hours for the SFP filter and demineralizer respectively, after which time the source terms decrease due to decay of short half-life nuclides.

There is no contribution from defective fuel elements because of low power and temperature during plant shutdown operations.

Dimensions and parameters of the radiation sources in auxiliary building used in the shielding analyses are listed in Table 12.2-27.

12.2.1.3 Turbine Generator Building

Radiation sources in the turbine generator building occur in the condensate polishing system (CPS) due to the design basis PTS leakage rate in the steam generator. Activity levels for all turbine generator building related sources are summarized in Table 12.2-19. The activities provided in Table 12.2-19 are based on normal operation reactor coolant activity levels and PTS leakage conditions. Radionuclide removal efficiencies of demineralizers in the CPS are assumed to be consistent with the guidance in NUREG-0017 (Reference 1).

12.2.1.4 Compound Building

Radioactive sources in the radwaste system components include fission and activation radionuclides produced in the core and in the reactor coolant. The level of radioactivity is dependent on the components and operating parameters of the particular radwaste system.

Gaseous radwaste system (GRS) source terms are provided in Table 12.2-20. Radiation sources for each component of the GRS are calculated using the shielding basis equilibrium reactor coolant radionuclide concentrations provided in Table 12.2-5, which are based on an assumed 0.25 percent fuel defect. Activity buildup on the process gas charcoal beds is calculated assuming maximum design basis holdup times for noble gases in accordance with NUREG-0017.

The source terms for LWMS tanks are provided in Table 12.2-21 and for the other LWMS processing equipment in Table 12.2-22. Source terms for the equipment waste tank (EWT) and floor drain tank (FDT) are calculated using reactor coolant equilibrium radionuclide concentrations presented in Table 12.2-5 and the activity fractions in Table 11.2-2. Radionuclide concentrations in the LWMS are determined using the DIJESTER Code (Reference 2). The accumulation and decay of radionuclides in the LWMS can be modeled using this code.

The activities of LWMS demineralizers are calculated using 44% of the RCS source term and an activity buildup and decay model. The calculation applies the process flow rates provided in Table 11.2-2, and the process fluid activity levels provided in Table 12.2-21. Source terms for the monitor tank are calculated using reactor coolant equilibrium radionuclide concentrations with 0.25% failed fuel and processed through the LWMS. The demineralizer resin is assumed to have a service life of 1 year. Although the service life of filters and resins in the LWMS may vary according to operating conditions, for radiation protection purposes, they are replaced based on the source term strength to provide reasonable assurance that occupational exposures associated with radwaste system operations remain ALARA.

Solid waste management system (SWMS) source terms are provided in Table 12.2-23. Source terms for the spent resin long-term storage tank are calculated based on 10 years of cumulative radioactive resin batches from the CVCS demineralizers with decay. Source terms for the LWMS demineralizers are presented in Table 12.2-22, based on a processing time of 1 year. The source terms for the low-activity spent resin tank (LASRT) in Table 12.2-23 are calculated using the source term for the LWMS spent resin and multiplying by a factor of 3 for conservatism in the shielding design. Since the LASRT is filled with resins from the LWMS, SGBD, and SFPCCS for 1 year and the source terms for the LWMS are highest, it is conservative that the LASRT is only filled with LWMS resin. Additionally, the source term in the waste drum storage area is presented in Table 12.2-24.

Dimensions and parameters of the radiation sources in compound building used in the shielding analyses are listed in Table 12.2-27.

12.2.1.5 Sources Resulting from Design Basis Accidents

Design parameters and source terms for design basis accidents (DBAs) are addressed in Chapter 15.

12.2.1.6 Stored Radioactivity

The holdup tank, reactor makeup water tank (RMWT), and boric acid storage tank (BAST) are the principal sources of activity outside the plant buildings. The surface dose rate of these tanks is designed so that it does not exceed 2.5 μSv/hr, except the immediate areas outside the side manways for these outdoor tanks which may exceed the Zone 1 criterion. A physical barrier and administrative controls are in place to prevent personnel from occupying the immediate vicinity of the outside tanks.

Spent fuel is stored in the SFP until it is placed in the spent fuel shipping cask for transport to an onsite interim storage facility or to an offsite storage facility.

Storage space is allocated in the compound building for the storage of spent filter cartridges and dewatered spent resin in the spent filter drum and HIC storage area as well as for solidified R/O concentrates and dry active waste (DAW) in drums in the waste drum storage area. The shielding design for the spent filter drum and HIC storage area is based on using the expected stored waste volumes for normal operation and the design basis source term (0.25% fuel failure) for the activity of wastes. For the source term of HICs, the volume and the associated source term of spent resin (not decayed) is increased by a factor of 1.656 (=Volume of 16 HICs / 1-cycle volume of spent resin) for conservatism. The zoning for this area is determined by summing the dose rates from the HIC and the spent filter drums. However, in determining the minimum shield wall thicknesses, the two dose rates are calculated individually since the impact of shield wall thicknesses is dominated by the close proximity of the individual sources (HIC or spent filter drum) to the walls around the designated storage areas. The shielding design for the waste drum area is based on the use of the design basis source term for the solidified R/O concentrate and the source term for the DAW is based on a waste drum with the highest activity at Korean domestic nuclear power plants to ensure that the shielding design is sufficiently conservative. Some of the drummed waste, such as R/O concentrate, are expected to have higher activity, and are stored at the interior of the drum layers within a stacking configuration, with lower activity waste drums at the exterior of each layer, to provide additional shielding.

The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1 (COL 12.2(1)).

12.2.1.7 Pipe Routing

Piping carrying radioactive materials is routed in pipe chases to the extent practicable when routed through low-radiation and low-contamination areas to maintain radiation exposure to plant personnel ALARA and reduce the spread of contamination.

Criteria for routing radioactive piping include:

a. Piping containing radioactive material is routed through shielded pipe chases to the extent practicable.

b. Systems containing radioactive liquids, gases, or slurries are physically located close to interfacing systems to reduce pipe length and minimize the need for routing radioactive piping through personnel access corridors.

c. Stagnant runs of piping are avoided to minimize the potential for crud traps. Flushing and decontamination capabilities are provided as necessary.

12.2.2 Airborne Radioactive Material Sources

Airborne radioactive material is introduced into the plant atmosphere through the leakage of radioactive fluids from equipment (e.g., valve stems, pump seals) and the evaporation of tritiated water.

The APR1400 is designed to minimize the potential for leakage of radioactive fluids. However, leakage of radioactive fluids through pump seals, valve stems, and other piping connections may not be eliminated entirely. According to the recommendations and guidance provided by NUREG-0017, airborne source terms are estimated based on the type of equipment, number of valves, pumps, and flanges, and level of radioactivity in the fluid stream.

The evaporation of tritiated water from unenclosed sources of water, such as the SFP, is a significant source of airborne tritium. The rate of evaporation is dependent on the pool temperature, air velocity across the pool, and relative humidity.

The airborne radioactivity concentrations in rooms or cubicles of occupied plant areas are maintained within the in-plant concentration limits prescribed in the applicable regulations. The plant design criteria that are implemented to provide reasonable assurance of conformance with the applicable regulations are as follows:

a. Maintenance of in-plant airborne concentrations of radioisotopes in normally occupied areas within the derived air concentration (DAC) prescribed in Table 1 of 10 CFR Part 20, Appendix B (Reference 3).

b. Sufficient confinement and ventilation capability to prevent the spread of airborne contamination.

c. Airborne radiation monitors in normally occupied areas where the potential for airborne contamination exists. The airborne radiation monitoring system is described in Section 11.5.

12.2.2.1 Production of Airborne Radioactive Material

Radioactive materials become airborne through the natural evaporation of contaminated fluids. Major contributors to airborne radioactivity during normal operation are (1) CVCS leaks, (2) evaporation from the SFP, (3) radwaste system leaks, (4) venting of radwaste tanks, and (5) HVAC system leaks. Minor contributions are from (1) cleaning and decontamination tools and equipment, (2) contaminated clothing, and (3) sample preparation and analysis.

Abnormal occurrences that can cause airborne radioactivity include (1) spills such as overflows and splashing, (2) failure of a ventilation system, (3) cracks and breaks in piping, (4) failures of pump and valve seals, and (5) equipment malfunctions.

12.2.2.2 Airborne Sources in Normally Accessible Areas

Airborne radioactive material is expected to affect general access areas and may become worse in the event of ventilation system failure or spillage of radioactive material in areas that are not sealed from general access areas. The ventilation pathway is designed to flow from areas of low airborne radioactivity to areas of potentially greater airborne radioactivity. The ventilation system is designed to control the airborne radioactivity in the laboratories, maintenance areas, and the reactor containment building.

Radiation exposure due to maintenance activities accounts for the majority of the exposure received by plant personnel because many maintenance activities are performed in areas with relatively high radiation and airborne radioactivity. The airborne radionuclide concentrations are calculated for the most common contributors, such as leaks and venting.

12.2.2.3 Airborne Concentrations

The determination of concentrations of airborne radioactive nuclides in cubicles is based on the model described below.

Concentrations of noble gases, tritium, and iodine are calculated as a fraction of the DAC for air in the reactor containment building and normally accessible cubicles within the auxiliary building and the compound building. Except under abnormal conditions, general access areas normally have little, if any, airborne contaminants during normal operation. Table 12.2-25 provides the airborne radioactivity concentrations and DAC fractions for the main cubicles in the reactor containment, auxiliary building, and compound building.

Equilibrium airborne radioactivity concentrations in rooms, cubicles, and other areas during normal operation are calculated based on the following equation:

$$C_A = \frac{LCP}{(\lambda V + F)}$$

Where:

C_A = airborne concentration in each cubicle (Bq/cm^3)

L = leak rate (cm^3/min)

C = radioactive concentration of liquid or gas (Bq/cm^3)

P = fraction of activity released to air

λ = decay constant (min^{-1})

V = enclosed room volume (cm^3)

F = air exhaust flow rate (cm^3/min)

The equation above represents the solution of a differential equation in equilibrium conditions in which there is no airborne radioactivity in the ventilation airflow(s) entering the area under consideration. To accommodate situations in which airborne radioactivity is in one or more ventilation airflow streams entering the area of concern, additional term(s) are added to the basic differential equation. Minimum HVAC flow rates and parameters used in the airborne source term calculations are provided in Table 12.2-28.

12.2.2.4 Basis and Assumptions for Partition Factors

a. Reactor Containment Building

The partition factors for airborne activity in the reactor containment building are calculated as follows:

1) 1.0 for noble gases

2) All other nuclides: use flashing fraction (ff), based on the enthalpy difference in accordance with NRC RG 1.183.

b. Auxiliary Building and Compound Building

The partition factors for airborne activity in the auxiliary building are calculated as follows:

1) 1.0 for noble gases

2) 1×10^{-3} for halogens in cold liquid (<120 °F) and 0.1 for halogens in a hot liquid (>120 °F)

3) 0.53 for H^3 in primary coolant and 0.1 for H^3 in cold liquids, and

4) 0.005 for all other nuclides

12.2.3 Sources Used in NUREG-0737 Post-Accident Shielding Analysis

Item II.B.2.3 of NUREG-0737 (Reference 4) clarifies the requirement for providing reasonable assurance that areas that require post-accident personnel access or contain safety-related equipment are adequately shielded in the vicinity of systems that may contain highly radioactive materials as a result of a DBA.

Plant areas requiring post-accident occupation (vital areas), their locations and the expected doses are identified in Subsections 12.3.1.8, 12.3.1.9 and 12.4.1.2, respectively.

The review of the shielding for systems that, as a result of an accident, contain highly radioactive materials is performed using the methodology described in Subsection 12.3.2.

Initial core releases that are equivalent to those recommended in NRC RG 1.183 (Reference 5) are used. The source terms are presented in Table 12.2-26.

These source terms are used to evaluate the adequacy of shielding in post-accident conditions using shielding computer codes, which are addressed in Section 12.3.2.2, to verify that:

a. Vital areas are accessible for operators to take the required mitigative actions during post-accident conditions.

b. Safety-related equipment is qualified for the environmental radiation condition in the area where the equipment is located.

In vital areas that are not occupied continuously, sufficient shielding is provided for reasonable assurance that radiation exposure during post-accident conditions does not exceed the total effective dose equivalent (TEDE) criterion (50 mSv). In vital areas that are occupied continuously, such as the main control room (MCR), the local radiation levels are limited to 0.15 mSv/hr averaged over 30 days per NUREG-0737, Subsection II.B.2.3 (Reference 4).

12.2.4 Combined License Information

COL 12.2(1) The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1.

12.2.5 References

1. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," Rev. 1, U.S. Nuclear Regulatory Commission, April 1985.

2. DIJESTER, "A Program to Compute Radioactive Decay in Fluid Flow Systems," Program No. 9.8. 060-1.0, D. J. Pichurski, April 1976.

3. 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," U.S. Nuclear Regulatory Commission.

4. NUREG-0737, Item II.B.2.3, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations," U.S. Nuclear Regulatory Commission, 1980.

5. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, July 2000.

6. ANSI/ANS 18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society, 1999.

7. CCC-650/DOORS3.2, "One-, Two- and Three- Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Safety Information Computational Center, ORNL, 1998.

8. ANSI/ANS 6.1.1, "American National Standard for Neutron and Gamma-Ray Fluence-to-Dose Factors," American Nuclear Society, August 1991.

9. 00000-SP-VV-118, Rev.00, "Software Verification and Validation Report for SHIELD-APR Version 2.0," KEPCO-E&C, March 2017.

Table 12.2-1

Normal Operation Neutron Spectra Outside the Reactor Vessel

Average Neutron Energy (eV)	Neutron Spectra[1] (neutrons/cm^2-s)
1.58E+07	3.84E+05
1.32E+07	1.04E+06
1.11E+07	3.78E+06
9.30E+06	6.69E+06
8.01E+06	9.76E+06
6.74E+06	2.03E+07
5.52E+06	2.79E+07
4.32E+06	5.00E+07
3.35E+06	4.00E+07
2.87E+06	3.17E+07
2.60E+06	3.94E+07
2.42E+06	2.00E+07
2.36E+06	6.20E+06
2.29E+06	3.13E+07
2.08E+06	8.73E+07
1.79E+06	1.22E+08
1.50E+06	2.07E+08
1.18E+06	4.96E+08
9.12E+05	5.06E+08
7.82E+05	2.31E+08
6.75E+05	1.04E+09
5.53E+05	1.01E+09
4.33E+05	9.84E+08
3.33E+05	1.53E+09

Average Neutron Energy (eV)	Neutron Spectra[1] (neutrons/cm^2-s)
2.40E+05	1.88E+09
1.47E+05	1.90E+09
8.92E+04	1.23E+09
5.41E+04	9.14E+08
3.64E+04	3.04E+08
2.89E+04	1.82E+08
2.51E+04	5.46E+08
2.30E+04	3.50E+08
1.85E+04	5.96E+08
1.11E+04	7.74E+08
5.23E+03	7.96E+08
2.47E+03	6.63E+08
1.02E+03	1.03E+09
3.34E+02	5.40E+08
1.58E+02	5.33E+08
6.93E+01	6.54E+08
2.40E+01	7.57E+08
7.86E+00	4.19E+08
3.45E+00	5.34E+08
1.37E+00	3.89E+08
6.45E-01	3.55E+08
2.57E-01	7.95E+08
5.00E-02	3.06E+09

(1) At core midplane, 15.24 cm (0.5 ft) from vessel surface

Table 12.2-2

Normal Operation Gamma Spectra Outside the Reactor Vessel

Average Gamma Energy (eV)	Gamma Spectra (1) (Gamma/cm^2-s)
1.20E+07	3.75E+05
9.00E+06	1.29E+08
7.50E+06	5.21E+08
6.50E+06	3.45E+08
5.50E+06	2.79E+08
4.50E+06	4.19E+08
3.50E+06	5.14E+08
2.50E+06	9.83E+08
1.75E+06	7.36E+08
1.25E+06	7.71E+08
9.00E+05	5.04E+08
7.50E+05	2.48E+08
6.50E+05	2.97E+08
5.00E+05	1.32E+09
3.00E+05	2.08E+09
1.50E+05	1.81E+09
8.00E+04	4.80E+08
4.50E+04	3.05E+07
2.50E+04	3.45E+05
1.50E+04	1.42E+05

(1) At core midplane, 15.24 cm (0.5 ft) from vessel surface

Table 12.2-3

Shutdown Gamma Spectra Outside the Reactor Vessel

Average Gamma Energy (eV)	Decay Gamma[1] (Gamma/cm^2-sec)	Material Activation (Gamma/cm^2-sec)
4.50E+06	0.00E+00	4.20E+02
3.50E+06	9.11E+02	4.45E+01
2.50E+06	9.70E+04	8.59E+05
1.75E+06	1.95E+05	1.33E+05
1.25E+06	3.03E+05	9.02E+06
9.00E+05	1.67E+05	1.60E+06
7.50E+05	9.73E+04	6.65E+05
6.50E+05	1.11E+05	7.38E+05
5.00E+05	2.89E+05	2.06E+06
3.00E+05	5.03E+05	4.98E+06
1.50E+05	4.21E+05	3.99E+06
8.00E+04	1.05E+05	1.03E+06
4.50E+04	5.25E+03	5.31E+04
2.50E+04	1.09E+00	1.10E+01
1.50E+04	0.00E+00	0.00E+00

(1) At core midplane, 15.24 cm (0.5 ft) from vessel surface, and 48 hours after shutdown

Table 12.2-4

Basis for Reactor Coolant Fission Product Activities[(1)]

Parameter	Maximum
Core power level, MWt	4,063
Duration of reactor operation, fuel cycles	5
Equilibrium fuel cycle, equivalent full-power days	480
Thermal neutron flux, n/cm^2-sec	6.32E+13
Average thermal fission rate, fission/MW-sec	3.12E+16
Fraction of fuel defect	0.0025
Reactor coolant mass including pressurizer, kg (lb)	2.92E+05 (6.43E+05)
Core coolant volume to reactor coolant volume ratio	0.073
Purification flow, kg/sec (lb/sec)	5.02 (11.07)
Boron concentration reduction rate, ppm/second	2.60E-05
Fuel cycle boron concentration at BOC (minimum), ppm	1,110
Ion exchanger and gas stripper removal efficiency	
CVCS purification ion exchanger	
Noble gas, tritium	0.0
Cs, Rb	0.5
Anion	0.99
All others	0.98
CVCS gas stripper removal efficiency	
Noble gas	0.999
All others	0.0
CVCS gas stripper operation	None
Fission product escape rate coefficients (sec^{-1})	
Xe, Kr	6.5E-08
I, Br, Rb, Cs	1.3E-08
Mo	2.0E-09
Te	1.0E-09
Sr, Ba	1.0E-11
All others	1.6E-12

(1) Shielding design source term (0.25% fuel defect, DAMSAM code input)

Table 12.2-5 (1 of 2)

Reactor Coolant Equilibrium Concentration
(Core Power: 4,063 MWt, 0.25 % Fuel Defect, No Gas Stripping)

Nuclide	Activity (Bq/g)
Kr-85m	1.04E+04
Kr-85	4.44E+04
Kr-87	8.14E+03
Kr-88	2.26E+04
Xe-131m	4.44E+04
Xe-133m	2.70E+03
Xe-133	2.89E+06
Xe-135m	5.92E+03
Xe-135	5.92E+04
Xe-137	1.37E+03
Xe-138	5.18E+03
Br-84	1.96E+02
I-131	2.48E+04
I-132	6.66E+03
I-133	3.52E+04
I-134	4.07E+03
I-135	2.00E+04
Rb-88	2.29E+04
Cs-134	3.52E+03
Cs-136	4.81E+02
Cs-137	4.07E+03
N-16	8.22E+06(2)
H-3	1.30E+05
Na-24	1.81E+03(1)

Nuclide	Activity (Bq/g)
Co-60	2.10E+01(1)
Zn-65	2.02E+01(1)
Sr-89	3.26E+01
Sr-90	2.18E+00
Sr-91	4.81E+01
Y-91m	2.81E+01
Y-91	4.81E+00
Y-93	1.15E+00
Zr-95	1.54E+01(1)
Nb-95	5.18E+00
Mo-99	2.81E+03
Tc-99m	1.63E+03
Ru-103	1.74E+00
Ru-106	7.40E-01
Ag-110m	5.15E+01(1)
Te-129m	5.92E+01
Te-129	6.29E+01
Te-131m	2.81E+02
Te-131	1.11E+02
Te-132	1.96E+03
Ba-137m	3.70E+03
Ba-140	4.07E+01
La-140	1.37E+01
Ce-141	1.52E+00

Table 12.2-5 (2 of 2)

Nuclide	Activity (Bq/g)
Cr-51	5.48E+02
Mn-54	6.34E+01(1)
Fe-55	4.75E+01(1)
Fe-59	1.19E+01(1)
Co-58	1.82E+02(1)

Nuclide	Activity (Bq/g)
Ce-143	4.07E+00
Ce-144	4.44E+00
W-187	9.70E+01(1)
Np-239	8.62E+01(1)

(1) Expected source terms based on ANSI/ANS 18.1 (Reference 6 in Subsection 12.2.5) are used when these values are higher than the design basis source terms, for added conservatism.

(2) At the reactor vessel outlet nozzle

Table 12.2-6

Deposited Crud Activities
(SG Tubing and Reactor System Piping)

Isotope	Activity (Bq/g-crud)
Cr-51	7.31E+09
Mn-54	1.99E+07
Fe-59	4.18E+07
Co-58	1.77E+09
Co-60	6.22E+07
Zr-95	4.90E+07

Table 12.2-7

N-16 Activity

Location	Activity (Bq/g)	Accumulated Transit Time (sec)
Vessel outlet nozzle	8.22E+06	0.66
Vessel outlet line (midpoint)	8.13E+06	0.78
Steam generator (midpoint)	5.68E+06	3.85
Reactor coolant pump (midpoint)	3.78E+06	7.56
Vessel inlet line (midpoint)	3.61E+06	8.02
Containment penetration area	1.94E+04	62.43[(1)]

(1) The N-16 concentration at the containment penetration is estimated to be 1.94E+04 Bq/g based on the transit time from RV to SG and the residence times in the regenerative and the letdown heat exchangers before the containment penetration. For shielding analysis purposes, this concentration is assumed to be the same as that at the outlet of the letdown heat exchanger, and no credit is taken for further decay along the 200 ft of piping from the letdown heat exchanger outlet to the containment isolation valve.

Table 12.2-8

Parameters Used in Spent Fuel Decay Gamma Source Calculation

Parameter	Value
Core power level, MWt	4,063
No. of fuel assemblies in core	241
Uranium loading, kg U/assembly	430.8
Fuel initial enrichment, U^{235} w/o	4.20
Irradiation cycle length, EFPD	480
Burnup per cycle, GWD/MTU	18.8
No. of irradiation cycles	3
Discharge burnup, GWD/MTU	56.4

Table 12.2-9

Spent Fuel Gamma Source

Mean Energy (MeV)	Time after Shutdown				
	50 hr	100 hr	200 hr	500 hr	1,000 hr
	MeV/W-s				
1.00E-02	2.04E+08	1.31E+08	6.94E+07	3.50E+07	2.60E+07
3.00E-02	1.62E+08	1.25E+08	8.86E+07	5.23E+07	3.60E+07
5.50E-02	1.37E+08	1.02E+08	7.08E+07	4.36E+07	3.16E+07
8.50E-02	2.69E+08	1.93E+08	1.15E+08	5.05E+07	3.07E+07
1.20E-01	1.11E+09	6.38E+08	2.37E+08	5.87E+07	4.38E+07
1.70E-01	3.10E+08	2.51E+08	1.91E+08	1.30E+08	8.84E+07
3.00E-01	2.21E+09	1.39E+09	6.79E+08	2.22E+08	1.04E+08
6.50E-01	8.52E+09	7.02E+09	5.77E+09	4.40E+09	3.46E+09
1.13E+00	1.28E+09	9.12E+08	5.90E+08	2.91E+08	1.44E+08
1.58E+00	3.16E+09	2.81E+09	2.22E+09	1.12E+09	3.84E+08
2.00E+00	3.20E+08	2.72E+08	2.12E+08	1.23E+08	6.18E+07
2.40E+00	2.38E+08	2.13E+08	1.72E+08	8.80E+07	3.04E+07
2.80E+00	4.13E+06	3.62E+06	2.95E+06	1.69E+06	8.20E+05
3.25E+00	1.87E+06	1.71E+06	1.40E+06	7.50E+05	2.94E+05
3.75E+00	6.66E+01	5.27E+01	5.24E+01	5.12E+01	4.94E+01
4.25E+00	9.58E+00	6.75E+00	6.72E+00	6.63E+00	6.49E+00
4.75E+00	2.77E+01	4.38E+00	4.36E+00	4.30E+00	4.20E+00
5.50E+00	4.77E+00	4.59E+00	4.57E+00	4.50E+00	4.40E+00

Table 12.2-10 (1 of 2)

CVCS Heat Exchanger Inventories, Maximum Values (Bq)

Nuclide	Letdown	Regenerative	CCP Miniflow
H-3	3.5E+10	2.6E+10	1.9E+10
N-16	1.4E+11	7.5E+10	0.0E+00
Kr-85m	2.8E+09	2.0E+09	1.5E+09
Kr-85	1.2E+10	8.9E+09	6.5E+09
Kr-87	2.2E+09	1.5E+09	1.1E+09
Kr-88	6.1E+09	4.3E+09	3.2E+09
Xe-131m	1.2E+10	8.9E+09	6.5E+09
Xe-133m	7.3E+08	5.4E+08	3.9E+08
Xe-133	7.8E+11	5.8E+11	4.2E+11
Xe-135m	1.6E+09	9.3E+08	6.5E+08
Xe-135	1.6E+10	1.2E+10	8.5E+09
Xe-137	3.7E+08	1.8E+08	1.2E+08
Xe-138	1.4E+09	8.0E+08	5.6E+08
Br-84	5.3E+07	5.2E+06	2.4E+05
Rb-88	6.2E+09	2.1E+09	1.3E+09
Sr-89	8.8E+06	9.3E+05	9.5E+04
Sr-90	5.9E+05	6.2E+04	6.4E+03
Sr-91	1.3E+07	1.4E+06	1.4E+05
Y-91m	7.6E+06	5.0E+06	3.6E+06
Y-91	1.3E+06	9.6E+05	7.0E+05
Y-93	3.1E+05	2.3E+05	1.7E+05
Zr-95	4.2E+06	4.4E+05	4.5E+04
Nb-95	1.4E+06	1.5E+05	1.5E+04
Tc-99m	4.4E+08	4.6E+07	4.7E+06
Mo-99	7.6E+08	8.0E+07	8.2E+06
Ru-103	4.7E+05	5.0E+04	5.1E+03
Ru-106	2.0E+05	2.1E+04	2.2E+03
Ag-110m	1.4E+07	1.5E+06	1.5E+05

Table 12.2-10 (2 of 2)

Nuclide	Letdown	Regenerative	CCP Miniflow
Te-129m	1.6E+07	1.6E+06	8.6E+04
Te-129	1.7E+07	1.7E+06	8.3E+04
I-131	6.7E+09	6.6E+08	3.6E+07
Te-131m	7.6E+07	7.5E+06	4.1E+05
Te-131	3.0E+07	2.9E+06	1.3E+05
Te-132	5.3E+08	5.2E+07	2.9E+06
I-132	1.8E+09	1.8E+08	9.2E+06
I-133	9.5E+09	9.4E+08	5.1E+07
I-134	1.1E+09	1.1E+08	5.3E+06
Cs-134	9.5E+08	4.0E+08	2.6E+08
I-135	5.4E+09	5.3E+08	2.9E+07
Cs-136	1.3E+08	5.4E+07	3.5E+07
Cs-137	1.1E+09	4.6E+08	3.0E+08
Ba-140	1.1E+07	1.2E+06	1.2E+05
La-140	3.7E+06	3.9E+05	4.0E+04
Ce-141	4.1E+05	4.3E+04	4.4E+03
Ce-143	1.1E+06	1.2E+05	1.2E+04
Ce-144	1.2E+06	1.3E+05	1.3E+04
Na-24	4.9E+08	5.1E+07	5.2E+06
Cr-51	1.5E+08	1.6E+07	1.6E+06
Mn-54	1.7E+07	1.8E+06	1.8E+05
Fe-55	1.3E+07	1.4E+06	1.4E+05
Fe-59	3.2E+06	3.4E+05	3.5E+04
Co-58	4.9E+07	5.2E+06	5.3E+05
Co-60	5.7E+06	6.0E+05	6.1E+04
Zn-65	5.5E+06	5.8E+05	5.9E+04
Ba-137m(1)	1.1E+09	4.6E+08	3.0E+08
W-187	2.6E+07	2.8E+06	2.8E+05
Np-239	2.3E+07	2.5E+06	2.5E+05

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

Table 12.2-11 (1 of 2)

CVCS Ion Exchanger Inventories, Maximum Values (Bq)

Nuclide	Purification	Deborating	Pre-Holdup	Boric Acid Condensate
H-3	1.5E+11	1.5E+11	1.4E+11	1.3E+11
N-16	5.4E+08	0.0E+00	0.0E+00	0.0E+00
Kr-85m	1.2E+10	1.2E+10	1.1E+10	1.4E+02
Kr-85	5.0E+10	5.0E+10	4.9E+10	3.1E+04
Kr-87	9.2E+09	9.2E+09	8.6E+09	3.2E+01
Kr-88	2.5E+10	2.5E+10	2.4E+10	2.0E+02
Xe-131m	5.0E+10	5.0E+10	4.8E+10	1.6E+04
Xe-133m	3.0E+09	3.0E+09	2.9E+09	4.1E+02
Xe-133	3.3E+12	3.3E+12	3.1E+12	6.0E+05
Xe-135m	6.7E+09	6.7E+09	6.2E+09	5.0E+00
Xe-135	6.7E+10	6.7E+10	6.2E+10	1.7E+03
Xe-137	1.5E+09	1.5E+09	1.4E+09	3.0E-01
Xe-138	5.8E+09	5.8E+09	5.4E+09	3.9E+00
Br-84	2.7E+09	2.5E+07	3.1E+06	5.5E-01
Rb-88	1.0E+11	1.3E+10	1.6E+09	1.0E+02
Sr-89	1.0E+12	7.3E+05	2.3E+09	3.1E+02
Sr-90	4.4E+11	4.9E+04	1.0E+09	2.5E+01
Sr-91	1.2E+10	1.1E+06	1.7E+07	1.1E+01
Y-91m	3.2E+07	3.2E+07	3.0E+07	8.1E+01
Y-91	5.4E+06	5.4E+06	5.3E+06	4.2E+03
Y-93	1.3E+06	1.3E+06	1.2E+06	3.9E+01
Zr-95	6.2E+11	3.5E+05	1.4E+09	1.5E+02
Nb-95	1.1E+11	1.2E+05	2.5E+08	4.5E+01
Tc-99m	2.5E+11	3.7E+07	3.4E+08	2.4E+02
Mo-99	4.8E+12	6.3E+07	8.2E+09	4.8E+03
Ru-103	4.3E+10	3.9E+04	9.5E+07	1.6E+01
Ru-106	1.0E+11	1.7E+04	2.3E+08	8.2E+00
Ag-110m	5.9E+12	1.2E+06	1.4E+10	5.6E+02

Table 12.2-11 (2 of 2)

Nuclide	Purification	Deborating	Pre-Holdup	Boric Acid Condensate
Te-129m	1.3E+12	4.1E+09	2.6E+09	1.9E+05
Te-129	1.9E+09	1.7E+07	2.2E+06	1.2E+05 (2)
I-131	1.2E+14	9.6E+11	2.2E+11	9.3E+06
Te-131m	2.1E+11	1.9E+09	2.9E+08	2.5E+03
Te-131	1.2E+09	1.1E+07	1.4E+06	5.6E+02 (2)
Te-132	4.0E+12	3.6E+10	6.3E+09	1.3E+05
I-132	4.0E+11	3.6E+09	4.6E+08	1.3E+05 (2)
I-133	1.9E+13	1.7E+11	2.5E+10	1.6E+05
I-134	9.2E+10	8.4E+08	1.1E+08	3.1E+01
Cs-134	3.5E+14	2.0E+09	6.1E+12	1.6E+04
I-135	3.5E+12	3.2E+10	4.2E+09	9.2E+03
Cs-136	2.3E+12	2.7E+08	3.9E+10	1.3E+03
Cs-137	5.0E+14	2.3E+09	8.6E+12	1.9E+04
Ba-140	3.2E+11	9.2E+05	6.6E+08	2.4E+02
La-140	1.4E+10	3.1E+05	2.3E+07	1.4E+01
Ce-141	3.1E+10	3.4E+04	6.7E+07	1.3E+01
Ce-143	3.5E+09	9.2E+04	5.4E+06	3.4E+00
Ce-144	5.4E+11	1.0E+05	1.2E+09	4.9E+01
Na-24	7.0E+11	4.1E+07	1.0E+09	6.7E+02
Cr-51	9.4E+12	7.2E+10	2.1E+10	4.5E+03
Mn-54	8.0E+12	1.1E+10	1.9E+10	7.0E+02
Fe-55	8.3E+12	8.0E+09	1.9E+10	5.3E+02
Fe-59	3.3E+11	1.7E+09	7.4E+08	1.1E+02
Co-58	8.0E+12	2.8E+10	1.8E+10	1.8E+03
Co-60	4.0E+12	3.6E+09	9.2E+09	2.4E+02
Zn-65	2.3E+12	4.5E+05	5.2E+09	2.2E+02
Ba-137m(1)	5.0E+14	2.3E+09	8.6E+12	1.9E+04
W-187	6.0E+10	2.2E+06	8.9E+07	5.8E+01
Np-239	1.3E+11	1.9E+06	2.1E+08	1.2E+02

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

(2) The inventories include the contributions from the decay of parents to cover the potential non-conservatism of not including the buildup of daughters due to the decay of Telluriums.

Table 12.2-12 (1 of 2)

CVCS Filter Inventories, Maximum Values (Bq)

Nuclide	Seal Injection	Reactor Drain	Boric Acid	Purification	Reactor Makeup Water
H-3	9.8E+08	1.7E+09	1.6E+09	2.0E+09	1.9E+09
N-16	0.0E+00	0.0E+00	0.0E+00	1.8E+08	0.0E+00
Kr-85m	7.7E+07	6.9E+07	0.0E+00	1.6E+08	7.1E-03
Kr-85	3.4E+08	5.7E+08	2.7E+04	6.7E+08	2.0E+02
Kr-87	5.6E+07	5.3E+07	0.0E+00	1.2E+08	4.6E-04
Kr-88	1.6E+08	1.5E+08	0.0E+00	3.4E+08	6.2E-03
Xe-131m	3.4E+08	4.9E+08	3.3E+02	6.7E+08	4.8E+01
Xe-133m	2.0E+07	2.3E+07	2.2E-02	4.1E+07	2.5E-01
Xe-133	2.2E+10	2.9E+10	1.5E+03	4.4E+10	8.2E+02
Xe-135m	3.4E+07	3.8E+07	0.0E+00	9.0E+07	1.5E-05
Xe-135	4.4E+08	4.1E+08	0.0E+00	9.0E+08	1.7E-01
Xe-137	6.3E+06	8.8E+06	0.0E+00	2.1E+07	2.4E-07
Xe-138	2.9E+07	3.3E+07	0.0E+00	7.8E+07	1.0E-05
Br-84	1.2E+04	1.3E+06	0.0E+00	3.0E+06	1.3E-06
Rb-88	6.6E+07	1.5E+08	0.0E+00	3.5E+08	3.4E-04
Sr-89	4.9E+03	4.0E+05	5.4E+04	4.9E+05	2.4E+00
Sr-90	3.3E+02	2.8E+04	3.2E+04	3.3E+04	3.7E-01
Sr-91	7.2E+03	3.3E+05	1.6E-12	7.3E+05	1.2E-03
Y-91m	1.9E+05	1.8E+05	0.0E+00	4.3E+05	7.6E-04
Y-91	3.6E+04	6.0E+04	8.6E+05	7.3E+04	3.5E+01
Y-93	8.6E+03	8.0E+03	1.5E-11	1.7E+04	4.4E-03
Zr-95	2.3E+03	1.9E+05	3.5E+04	2.3E+05	1.3E+00
Nb-95	7.8E+02	6.3E+04	5.0E+03	7.8E+04	2.9E-01
Tc-99m	2.4E+05	1.1E+07	0.0E+00	2.5E+07	1.6E-02
Mo-99	4.2E+05	2.5E+07	7.8E+02	4.3E+07	3.5E+00
Ru-103	2.6E+02	2.1E+04	2.0E+03	2.6E+04	1.1E-01
Ru-106	1.1E+02	9.5E+03	6.9E+03	1.1E+04	1.1E-01
Ag-110m	7.8E+03	6.6E+05	4.0E+05	7.8E+05	7.1E+00

Table 12.2-12 (2 of 2)

Nuclide	Seal Injection	Reactor Drain	Boric Acid	Purification	Reactor Makeup Water
Te-129m	4.5E+03	7.2E+05	5.1E+04	9.0E+05	3.1E-01
Te-129	4.3E+03	4.1E+05	0.0E+00	9.5E+05	1.9E-06
I-131	1.9E+06	2.6E+08	8.4E+05	3.8E+08	2.1E+01
Te-131m	2.1E+04	2.2E+06	4.1E-02	4.3E+06	5.9E-03
Te-131	6.8E+03	7.2E+05	0.0E+00	1.7E+06	4.6E-07
Te-132	1.5E+05	1.8E+07	1.3E+03	3.0E+07	3.1E-01
I-132	4.8E+05	4.4E+07	0.0E+00	1.0E+08	8.3E-04
I-133	2.6E+06	2.6E+08	5.3E-02	5.3E+08	3.8E-01
I-134	2.7E+05	2.6E+07	0.0E+00	6.2E+07	7.3E-05
Cs-134	1.3E+07	4.5E+07	1.7E+07	5.3E+07	2.3E+02
I-135	1.5E+06	1.4E+08	0.0E+00	3.0E+08	2.2E-02
Cs-136	1.8E+06	5.4E+06	3.0E+04	7.3E+06	4.1E+00
Cs-137	1.5E+07	5.2E+07	2.5E+07	6.2E+07	2.8E+02
Ba-140	6.2E+03	4.6E+05	5.5E+03	6.2E+05	7.6E-01
La-140	2.1E+03	1.1E+05	7.8E-02	2.1E+05	6.3E-03
Ce-141	2.3E+02	1.8E+04	1.3E+03	2.3E+04	8.1E-02
Ce-143	6.1E+02	3.2E+04	3.1E-03	6.2E+04	1.2E-03
Ce-144	6.7E+02	5.7E+04	3.7E+04	6.7E+04	6.3E-01
Na-24	2.7E+05	1.3E+07	6.5E-06	2.7E+07	1.1E-01
Cr-51	5.7E+10	1.8E+10	8.5E+08	8.6E+12	5.3E+05
Mn-54	4.9E+10	1.6E+10	9.5E+09	7.4E+12	1.4E+06
Fe-55	5.0E+10	1.7E+10	1.4E+10	7.6E+12	1.6E+06
Fe-59	2.0E+09	6.4E+08	6.3E+07	3.0E+11	2.8E+04
Co-58	4.8E+10	1.6E+10	2.7E+09	7.3E+12	8.8E+05
Co-60	2.4E+10	8.0E+09	7.4E+09	3.6E+12	8.0E+05
Zn-65	3.1E+03	2.6E+05	1.5E+05	3.1E+05	2.8E+00
Ba-137m(1)	1.5E+07	5.2E+07	2.5E+07	6.2E+07	2.8E+02
W-187	1.5E+04	7.4E+05	1.3E-03	1.5E+06	1.6E-02
Np-239	1.3E+04	7.5E+05	7.8E+00	1.3E+06	7.8E-02

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

Table 12.2-13 (1 of 5)

CVCS Tank Inventories, Maximum Values (Bq)

Nuclide	Reactor Drain		Equipment Drain		Volume Control	
	Liquid	Vapor	Liquid	Vapor	Liquid	Vapor
H-3	1.6E+12	7.9E+08	1.7E+11	3.7E+06	2.0E+12	1.0E+09
N-16	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Kr-85m	6.4E+10	7.9E+10	1.1E+10	1.2E+10	1.6E+11	3.6E+12
Kr-85	5.3E+11	8.5E+12	4.7E+10	3.5E+11	7.0E+11	1.6E+13
Kr-87	4.9E+10	1.8E+10	8.5E+09	2.8E+09	1.2E+11	2.5E+12
Kr-88	1.4E+11	1.1E+11	2.4E+10	1.7E+10	3.4E+11	7.7E+12
Xe-131m	4.6E+11	3.3E+12	4.7E+10	1.3E+10	7.0E+11	8.8E+12
Xe-133m	2.2E+10	9.5E+10	3.0E+09	1.8E+10	4.2E+10	5.4E+11
Xe-133	2.7E+13	1.6E+14	3.0E+12	3.8E+11	4.5E+13	5.7E+14
Xe-135m	3.5E+10	1.5E+09	6.2E+09	2.3E+08	7.0E+10	6.0E+11
Xe-135	3.8E+11	4.9E+11	6.2E+10	7.8E+10	9.2E+11	1.1E+13
Xe-137	8.1E+09	9.3E+07	1.4E+09	1.4E+07	1.3E+10	5.8E+10
Xe-138	3.1E+10	1.2E+09	5.4E+09	1.8E+08	6.0E+10	4.9E+11
Br-84	1.2E+09	5.9E+03	2.0E+08	1.2E+01	2.6E+07	1.0E+04
Rb-88	1.4E+11	3.8E+05	2.4E+10	7.7E+02	1.4E+11	4.8E+07
Sr-89	3.7E+08	1.8E+05	4.2E+07	7.8E+02	1.0E+07	5.1E+03
Sr-90	2.6E+07	1.3E+04	2.9E+06	6.3E+01	6.9E+05	3.4E+02
Sr-91	3.1E+08	2.4E+04	5.1E+07	5.1E+01	1.5E+07	7.4E+03
Y-91m	1.7E+08	1.3E+03	2.9E+07	2.7E+00	3.9E+08	1.7E+05
Y-91	5.5E+07	2.7E+04	6.2E+06	1.2E+02	7.6E+07	3.8E+04
Y-93	7.4E+06	6.0E+02	1.2E+06	1.3E+00	1.8E+07	8.8E+03
Zr-95	1.8E+08	8.7E+04	2.0E+07	3.8E+02	4.8E+06	2.4E+03
Nb-95	5.8E+07	2.8E+04	6.5E+06	1.1E+02	1.6E+06	8.1E+02
Tc-99m	1.0E+10	5.3E+05	1.7E+09	1.1E+03	5.0E+08	2.5E+05
Mo-99	2.3E+10	6.5E+06	3.1E+09	1.7E+04	8.8E+08	4.4E+05
Ru-103	2.0E+07	9.5E+03	2.2E+06	4.0E+01	5.5E+05	2.7E+02
Ru-106	8.8E+06	4.4E+03	9.8E+05	2.1E+01	2.3E+05	1.2E+02
Ag-110m	6.1E+08	3.1E+05	6.8E+07	1.4E+03	1.6E+07	8.1E+03

Table 12.2-13 (2 of 5)

Nuclide	Reactor Drain		Equipment Drain		Volume Control	
	Liquid	Vapor	Liquid	Vapor	Liquid	Vapor
Te-129m	6.7E+08	3.2E+05	7.5E+07	1.3E+03	9.3E+06	4.6E+03
Te-129	3.8E+08	4.0E+03	6.6E+07	8.1E+00	9.0E+06	4.0E+03
I-131	2.4E+11	9.7E+07	2.9E+10	3.1E+05	3.9E+09	1.9E+06
Te-131m	2.0E+09	3.6E+05	3.0E+08	8.3E+02	4.4E+07	2.2E+04
Te-131	6.6E+08	2.6E+03	1.2E+08	5.3E+00	1.4E+07	5.4E+03
Te-132	1.7E+10	5.0E+06	2.2E+09	1.3E+04	3.1E+08	1.5E+05
I-132	4.0E+10	8.5E+05	7.0E+09	1.7E+03	9.9E+08	4.7E+05
I-133	2.4E+11	3.5E+07	3.7E+10	7.7E+04	5.5E+09	2.7E+06
I-134	2.4E+10	2.0E+05	4.2E+09	4.0E+02	5.7E+08	2.5E+05
Cs-134	4.2E+10	2.1E+07	4.7E+09	1.0E+05	2.8E+10	1.4E+07
I-135	1.3E+11	7.2E+06	2.1E+10	1.5E+04	3.1E+09	1.5E+06
Cs-136	5.0E+09	2.2E+06	5.8E+08	7.6E+03	3.8E+09	1.9E+06
Cs-137	4.9E+10	2.5E+07	5.4E+09	1.2E+05	3.2E+10	1.6E+07
Ba-140	4.2E+08	1.8E+05	4.9E+07	6.4E+02	1.3E+07	6.4E+03
La-140	1.0E+08	2.3E+04	1.5E+07	5.4E+01	4.3E+06	2.1E+03
Ce-141	1.7E+07	8.1E+03	1.9E+06	3.3E+01	4.8E+05	2.4E+02
Ce-143	3.0E+07	5.8E+03	4.4E+06	1.4E+01	1.3E+06	6.3E+02
Ce-144	5.3E+07	2.6E+04	5.9E+06	1.2E+02	1.4E+06	7.0E+02
Na-24	1.2E+10	1.4E+06	1.9E+09	2.9E+03	5.6E+08	2.8E+05
Cr-51	6.1E+09	2.9E+06	6.9E+08	1.1E+04	1.7E+08	8.6E+04
Mn-54	7.5E+08	3.8E+05	8.4E+07	1.8E+03	2.0E+07	1.0E+04
Fe-55	5.7E+08	2.9E+05	6.3E+07	1.4E+03	1.5E+07	7.5E+03
Fe-59	1.4E+08	6.6E+04	1.5E+07	2.8E+02	3.7E+06	1.9E+03
Co-58	2.1E+09	1.0E+06	2.4E+08	4.6E+03	5.7E+07	2.9E+04
Co-60	2.5E+08	1.3E+05	2.8E+07	6.0E+02	6.6E+06	3.3E+03
Zn-65	2.4E+08	1.2E+05	2.7E+07	5.6E+02	6.3E+06	3.2E+03
Ba-137m(1)	4.9E+10	2.5E+07	5.4E+09	1.2E+05	3.2E+10	1.6E+07
W-187	6.8E+08	1.1E+05	1.0E+08	2.4E+02	3.0E+07	1.5E+04
Np-239	6.9E+08	1.8E+05	9.4E+07	4.5E+02	2.7E+07	1.3E+04

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

Table 12.2-13 (3 of 5)

Nuclide	Holdup		Reactor Makeup Water		Boric Acid Storage	
	Liquid	Vapor	Liquid	Vapor	Liquid	Vapor
H-3	2.3E+15	1.0E+09	1.8E+14	1.1E+10	9.7E+13	3.3E+09
N-16	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Kr-85m	3.3E+08	8.1E+09	6.6E+02	4.2E+03	0.0E+00	0.0E+00
Kr-85	1.4E+10	1.8E+11	1.9E+07	1.1E+08	1.6E+09	1.3E+09
Kr-87	7.9E+07	1.9E+09	4.3E+01	2.7E+02	0.0E+00	0.0E+00
Kr-88	5.2E+08	1.1E+10	5.8E+02	3.7E+03	0.0E+00	0.0E+00
Xe-131m	8.6E+09	6.6E+09	4.5E+06	1.2E+07	2.0E+07	1.1E+08
Xe-133m	2.9E+08	8.7E+09	2.4E+04	6.4E+04	1.3E+03	7.6E+03
Xe-133	3.9E+11	1.9E+11	7.7E+07	2.1E+08	8.7E+07	5.0E+08
Xe-135m	4.7E+06	1.3E+08	1.4E+00	3.9E+00	0.0E+00	0.0E+00
Xe-135	2.6E+09	4.1E+10	1.6E+04	4.5E+04	2.5E-09	1.4E-08
Xe-137	1.6E+05	7.7E+06	2.2E-02	6.2E-02	0.0E+00	0.0E+00
Xe-138	3.4E+06	9.9E+07	9.7E-01	2.7E+00	0.0E+00	0.0E+00
Br-84	1.1E+06	3.5E+01	1.2E-01	6.1E-06	0.0E+00	0.0E+00
Rb-88	1.0E+08	1.6E+03	3.2E+01	1.6E-03	0.0E+00	0.0E+00
Sr-89	1.8E+08	2.5E+03	2.3E+05	1.1E+01	3.2E+09	1.5E+05
Sr-90	2.4E+07	2.0E+02	3.5E+04	2.1E+00	1.9E+09	6.5E+04
Sr-91	1.3E+07	1.7E+02	1.2E+02	5.8E-03	9.6E-08	0.0E+00
Y-91m	1.5E+08	1.3E+03	7.2E+01	3.6E-03	0.0E+00	0.0E+00
Y-91	9.6E+09	3.4E+04	3.3E+06	1.7E+02	5.1E+10	2.3E+06
Y-93	7.6E+07	5.9E+02	4.1E+02	2.0E-02	9.2E-07	0.0E+00
Zr-95	8.9E+07	1.2E+03	1.2E+05	6.4E+00	2.1E+09	9.3E+04
Nb-95	2.6E+07	3.8E+02	2.8E+04	1.3E+00	3.0E+08	1.4E+04
Tc-99m	2.3E+08	3.7E+03	1.5E+03	7.5E-02	0.0E+00	0.0E+00
Mo-99	5.0E+09	6.1E+04	3.3E+05	1.6E+01	4.6E+07	2.3E+03
Ru-103	8.9E+06	1.3E+02	1.0E+04	5.0E-01	1.2E+08	5.6E+03
Ru-106	6.6E+06	6.5E+01	1.0E+04	5.9E-01	4.1E+08	1.5E+04
Ag-110m	4.3E+08	4.5E+03	6.6E+05	3.8E+01	2.4E+10	9.0E+05

Table 12.2-13 (4 of 5)

Nuclide	Holdup		Reactor Makeup Water		Boric Acid Storage	
	Liquid	Vapor	Liquid	Vapor	Liquid	Vapor
Te-129m	2.7E+08	4.0E+03	2.9E+04	1.4E+00	3.0E+09	1.4E+05
Te-129	9.6E+05	2.4E+01	1.8E-01	9.0E-06	0.0E+00	0.0E+00
I-131	7.0E+10	1.0E+06	2.0E+06	9.3E+01	5.0E+10	2.5E+06
Te-131m	2.5E+08	2.6E+03	5.5E+02	2.7E-02	2.4E+03	1.2E-01
Te-131	4.4E+05	1.5E+01	4.3E-02	2.1E-06	0.0E+00	0.0E+00
Te-132	3.6E+09	4.4E+04	2.9E+04	1.4E+00	7.7E+07	3.8E+03
I-132	2.3E+08	5.1E+03	7.8E+01	3.9E-03	0.0E+00	0.0E+00
I-133	2.2E+10	2.4E+05	3.6E+04	1.8E+00	3.2E+03	1.6E-01
I-134	4.4E+07	1.2E+03	6.9E+00	3.4E-04	0.0E+00	0.0E+00
Cs-134	8.8E+09	1.3E+05	2.1E+07	1.3E+03	1.0E+12	3.6E+07
I-135	2.3E+09	4.5E+04	2.0E+03	1.0E-01	0.0E+00	0.0E+00
Cs-136	7.2E+08	1.2E+04	3.8E+05	1.8E+01	1.8E+09	8.8E+04
Cs-137	1.1E+10	1.5E+05	2.6E+07	1.6E+03	1.5E+12	5.0E+07
Ba-140	1.5E+08	2.3E+03	7.1E+04	3.3E+00	3.3E+08	1.6E+04
La-140	1.8E+07	1.9E+02	5.9E+02	2.9E-02	4.6E+03	2.3E-01
Ce-141	7.3E+06	1.1E+02	7.6E+03	3.6E-01	7.8E+07	3.7E+03
Ce-143	4.5E+06	4.8E+01	1.2E+02	5.8E-03	1.9E+02	9.3E-03
Ce-144	3.8E+07	3.9E+02	5.9E+04	3.4E+00	2.2E+09	8.2E+04
Na-24	8.8E+08	1.0E+04	1.1E+04	5.2E-01	3.9E-01	1.9E-05
Cr-51	2.5E+09	3.8E+04	2.4E+06	1.1E+02	2.2E+10	1.0E+06
Mn-54	5.5E+08	5.6E+03	8.5E+05	4.9E+01	3.3E+10	1.2E+06
Fe-55	4.9E+08	4.2E+03	7.2E+05	4.3E+01	3.5E+10	1.2E+06
Fe-59	6.3E+07	9.1E+02	7.7E+04	3.8E+00	9.9E+08	4.5E+04
Co-58	1.1E+09	1.5E+04	1.5E+06	8.0E+01	2.7E+10	1.2E+06
Co-60	2.3E+08	1.9E+03	3.3E+05	2.0E+01	1.7E+10	5.9E+05
Zn-65	1.7E+08	1.8E+03	2.6E+05	1.5E+01	9.1E+09	3.5E+05
Ba-137m[(1)]	1.1E+10	1.5E+05	2.6E+07	1.6E+03	1.5E+12	5.0E+07
W-187	8.1E+07	8.4E+02	1.5E+03	7.2E-02	7.9E+01	4.0E-03
Np-239	1.4E+08	1.6E+03	7.3E+03	3.6E-01	4.6E+05	2.3E+01

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

Table 12.2-13 (5 of 5)

Nuclide	In-containment Refueling Water Storage		Nuclide	In-containment Refueling Water Storage	
	Liquid	Vapor		Liquid	Vapor
H-3	1.7E+14	2.7E+10	Te-129m	1.6E+09	8.8E+04
N-16	0.0E+00	0.0E+00	Te-129	3.1E-03	0.0E+00
Kr-85m	5.1E+08	0.0E+00	I-131	3.6E+11	3.5E+05
Kr-85	1.3E+12	4.0E+09	Te-131m	1.6E+09	2.5E-03
Kr-87	2.2E+01	0.0E+00	Te-131	0.0E+00	0.0E+00
Kr-88	2.3E+07	0.0E+00	Te-132	2.2E+10	2.1E+02
Xe-131m	1.2E+12	3.6E+06	I-132	2.1E+05	0.0E+00
Xe-133m	5.0E+10	4.7E+01	I-133	1.4E+11	2.3E-03
Xe-133	7.0E+13	7.1E+06	I-134	7.0E-05	0.0E+00
Xe-135m	0.0E+00	0.0E+00	Cs-134	1.6E+12	2.3E+08
Xe-135	8.8E+10	0.0E+00	I-135	3.7E+09	0.0E+00
Xe-137	0.0E+00	0.0E+00	Cs-136	1.4E+10	2.0E+04
Xe-138	0.0E+00	0.0E+00	Cs-137	2.7E+12	4.1E+08
Br-84	0.0E+00	0.0E+00	Ba-140	6.6E+08	3.6E+03
Rb-88	0.0E+00	0.0E+00	La-140	1.1E+08	6.6E-03
Sr-89	1.5E+09	1.4E+05	Ce-141	4.1E+07	2.2E+03
Sr-90	3.3E+09	5.4E+05	Ce-143	2.7E+07	2.2E-04
Sr-91	3.6E+07	0.0E+00	Ce-144	2.4E+09	3.7E+05
Y-91m	4.0E-07	0.0E+00	Na-24	4.1E+09	2.1E-07
Y-91	2.0E+10	2.6E+06	Cr-51	1.3E+10	5.2E+05
Y-93	3.8E+06	0.0E+00	Mn-54	3.7E+10	5.8E+06
Zr-95	1.1E+09	1.2E+05	Fe-55	5.3E+10	8.5E+06
Nb-95	1.5E+08	9.0E+03	Fe-59	4.7E+08	3.8E+04
Tc-99m	1.8E+08	0.0E+00	Co-58	1.4E+10	1.7E+06
Mo-99	3.0E+10	1.1E+02	Co-60	2.8E+10	4.5E+06
Ru-103	5.9E+07	4.1E+03	Zn-65	9.2E+09	1.4E+06
Ru-106	4.9E+08	7.7E+04	Ba-137m	2.7E+12	4.1E+08
Ag-110m	2.4E+10	3.7E+06	W-187	4.6E+08	6.8E-05
			Np-239	8.5E+08	9.3E-01

Table 12.2-14 (1 of 2)

Boric Acid Concentrator Package Inventories, Maximum Values (Bq)

Nuclide	Concentrate Heater	Concentrate Cooler	Flash Tank	Vapor Separator	Concentrate Pump	Concentrate Transfer Pump
H-3	6.9E+10	5.3E+09	6.0E+11	1.3E+08	5.8E+09	5.8E+08
N-16	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Kr-85m	7.7E+03	5.9E+02	7.3E+05	2.0E+05	6.5E+02	6.5E+01
Kr-85	1.7E+06	1.3E+05	1.6E+08	4.2E+07	1.4E+05	1.4E+04
Kr-87	1.7E+03	1.3E+02	1.7E+05	4.4E+04	1.5E+02	1.5E+01
Kr-88	1.1E+04	8.2E+02	1.0E+06	2.7E+05	8.9E+02	8.9E+01
Xe-131m	8.8E+05	6.8E+04	8.4E+07	2.2E+07	7.4E+04	7.4E+03
Xe-133m	2.2E+04	1.7E+03	2.1E+06	5.6E+05	1.9E+03	1.9E+02
Xe-133	3.2E+07	2.5E+06	3.1E+09	8.2E+08	2.7E+06	2.7E+05
Xe-135m	2.7E+02	2.1E+01	2.6E+04	6.8E+03	2.3E+01	2.3E+00
Xe-135	9.0E+04	6.9E+03	8.5E+06	2.3E+06	7.5E+03	7.5E+02
Xe-137	1.6E+01	1.3E+00	1.6E+03	4.1E+02	1.4E+00	1.4E-01
Xe-138	2.1E+02	1.6E+01	2.0E+04	5.3E+03	1.8E+01	1.8E+00
Br-84	1.2E+05	9.1E+03	1.0E+06	2.3E+02	9.9E+03	9.9E+02
Rb-88	5.6E+06	4.3E+05	4.8E+07	1.1E+04	4.7E+05	4.7E+04
Sr-89	1.7E+07	1.3E+06	1.4E+08	3.2E+04	1.4E+06	1.4E+05
Sr-90	1.3E+06	1.0E+05	1.1E+07	2.6E+03	1.1E+05	1.1E+04
Sr-91	6.1E+05	4.7E+04	5.3E+06	1.2E+03	5.1E+04	5.1E+03
Y-91m	4.4E+06	3.4E+05	3.8E+07	8.4E+03	3.7E+05	3.7E+04
Y-91	2.2E+08	1.7E+07	1.9E+09	4.3E+05	1.9E+07	1.9E+06
Y-93	2.1E+06	1.6E+05	1.8E+07	4.0E+03	1.8E+05	1.8E+04
Zr-95	8.2E+06	6.3E+05	7.0E+07	1.6E+04	6.8E+05	6.8E+04
Nb-95	2.4E+06	1.9E+05	2.1E+07	4.7E+03	2.0E+05	2.0E+04
Tc-99m	1.3E+07	9.8E+05	1.1E+08	2.4E+04	1.1E+06	1.1E+05
Mo-99	2.6E+08	2.0E+07	2.2E+09	5.0E+05	2.2E+07	2.2E+06
Ru-103	8.4E+05	6.5E+04	7.3E+06	1.6E+03	7.1E+04	7.1E+03
Ru-106	4.4E+05	3.4E+04	3.8E+06	8.5E+02	3.7E+04	3.7E+03
Ag-110m	3.0E+07	2.3E+06	2.6E+08	5.8E+04	2.5E+06	2.5E+05

Table 12.2-14 (2 of 2)

Nuclide	Concentrate Heater	Concentrate Cooler	Flash Tank	Vapor Separator	Concentrate Pump	Concentrate Transfer Pump
Te-129m	2.6E+07	2.0E+06	2.2E+08	5.0E+04	2.2E+06	2.2E+05
Te-129	8.1E+04	6.2E+03	7.0E+05	1.6E+02	6.8E+03	6.8E+02
I-131	5.5E+09	4.2E+08	4.7E+10	1.0E+07	4.6E+08	4.6E+07
Te-131m	9.9E+06	7.6E+05	8.6E+07	1.9E+04	8.3E+05	8.3E+04
Te-131	5.2E+04	4.0E+03	4.5E+05	1.0E+02	4.4E+03	4.4E+02
Te-132	1.9E+08	1.5E+07	1.7E+09	3.7E+05	1.6E+07	1.6E+06
I-132	1.7E+07	1.3E+06	1.5E+08	3.3E+04	1.5E+06	1.5E+05
I-133	8.8E+08	6.8E+07	7.6E+09	1.7E+06	7.4E+07	7.4E+06
I-134	4.0E+06	3.1E+05	3.5E+07	7.8E+03	3.4E+05	3.4E+04
Cs-134	8.7E+08	6.7E+07	7.5E+09	1.7E+06	7.3E+07	7.3E+06
I-135	1.6E+08	1.2E+07	1.3E+09	3.0E+05	1.3E+07	1.3E+06
Cs-136	6.9E+07	5.3E+06	6.0E+08	1.3E+05	5.8E+06	5.8E+05
Cs-137	1.0E+09	7.8E+07	8.8E+09	2.0E+06	8.5E+07	8.5E+06
Ba-140	1.3E+07	1.0E+06	1.1E+08	2.5E+04	1.1E+06	1.1E+05
La-140	7.5E+05	5.8E+04	6.5E+06	1.5E+03	6.3E+04	6.3E+03
Ce-141	7.0E+05	5.4E+04	6.1E+06	1.4E+03	5.9E+04	5.9E+03
Ce-143	1.8E+05	1.4E+04	1.6E+06	3.5E+02	1.5E+04	1.5E+03
Ce-144	2.6E+06	2.0E+05	2.3E+07	5.0E+03	2.2E+05	2.2E+04
Na-24	3.6E+07	2.8E+06	3.1E+08	6.9E+04	3.0E+06	3.0E+05
Cr-51	2.4E+08	1.9E+07	2.1E+09	4.7E+05	2.0E+07	2.0E+06
Mn-54	3.8E+07	2.9E+06	3.2E+08	7.2E+04	3.2E+06	3.2E+05
Fe-55	2.9E+07	2.2E+06	2.5E+08	5.5E+04	2.4E+06	2.4E+05
Fe-59	5.9E+06	4.6E+05	5.1E+07	1.1E+04	5.0E+05	5.0E+04
Co-58	9.7E+07	7.5E+06	8.4E+08	1.9E+05	8.2E+06	8.2E+05
Co-60	1.3E+07	9.8E+05	1.1E+08	2.5E+04	1.1E+06	1.1E+05
Zn-65	1.2E+07	9.1E+05	1.0E+08	2.3E+04	1.0E+06	1.0E+05
Ba-137m(1)	1.0E+09	7.8E+07	8.8E+09	2.0E+06	8.5E+07	8.5E+06
W-187	3.1E+06	2.4E+05	2.7E+07	6.0E+03	2.6E+05	2.6E+04
Np-239	6.7E+06	5.2E+05	5.8E+07	1.3E+04	5.6E+05	5.6E+04

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

Table 12.2-15

Gamma Sources of Irradiated ICI, CEA, NSA, and SCA

Mean Energy (MeV)	In-Core Instrument [(1)] (γ/sec/cm)	Control Element Assembly (γ/sec/cm)	Neutron Source Assembly (γ/sec/assembly)	Surveillance Capsule Assembly (γ/sec/cm)
3.00E-01	3.72E+10	1.15E+10	6.90E+13	1.19E+09
6.50E-01	3.40E+11	2.43E+11	2.36E+15	1.04E+09
1.13E+00	1.08E+11	1.02E+11	2.46E+14	1.48E+10
1.58E+00	1.31E+09	6.61E+08	7.46E+14	2.26E+06
2.00E+00	5.76E+05	5.46E+05	8.02E+13	1.05E+05
2.40E+00	1.90E+03	1.74E+03	5.12E+11	1.24E+03
2.80E+00	7.20E+07	8.79E+00	5.14E+10	8.96E+02
3.25E+00	1.37E+01	1.61E+00	2.93E+06	1.64E+02
3.75E+00	4.68E+04	1.42E-08	3.25E+05	2.02E-08
4.25E+00	6.14E+02	0.00E+00	1.00E+05	2.64E-10
4.75E+00	0.00E+00	0.00E+00	5.70E+04	0.00E+00
5.50E+00	0.00E+00	0.00E+00	5.16E+04	0.00E+00

(1) Includes detector and associated cable.

Table 12.2-16

Shutdown Cooling System Specific Source Strength

Decay Time (hr)	Energy (MeV)							
	0.25	0.50	0.75	1.00	1.38	2.00	3.00	4.00
	Maximum Values (γ/g-sec)							
1	1.2E+06	4.9E+04	2.9E+04	1.2E+04	1.6E+04	1.6E+04	2.5E+03	5.4E+00
10	1.1E+06	3.1E+04	1.4E+04	3.6E+03	6.0E+03	2.3E+03	1.1E+03	4.2E-05
100	6.8E+05	6.1E+03	9.2E+03	5.3E+02	3.1E+02	8.6E-01	1.7E+01	0.0E+00

Table 12.2-17

Fission and Corrosion Product Activities in the Spent Fuel Pool

Nuclide	Activity (Bq/g)
H-3	4.0E+04
N-16	0.0E+00
Kr-85m	0.0E+00
Kr-85	0.0E+00
Kr-87	0.0E+00
Kr-88	0.0E+00
Xe-131m	0.0E+00
Xe-133m	0.0E+00
Xe-133	0.0E+00
Xe-135m	0.0E+00
Xe-135	0.0E+00
Xe-137	0.0E+00
Xe-138	0.0E+00
Br-84	0.0E+00
Rb-88	0.0E+00
Sr-89	2.6E-01
Sr-90	8.5E-02
Sr-91	1.6E-02
Y-91m	0.0E+00
Y-91	6.5E-01
Y-93	1.7E-03
Zr-95	1.3E-01
Nb-95	3.9E-02
Tc-99m	8.1E-02
Mo-99	1.3E+01
Ru-103	1.3E-02
Ru-106	1.6E-02
Ag-110m	9.1E-01
Te-129m	4.4E-01

Nuclide	Activity (Bq/g)
Te-129	0.0E+00
I-131	1.6E+02
Te-131m	7.3E-01
Te-131	0.0E+00
Te-132	9.9E+00
I-132	9.4E-05
I-133	6.1E+01
I-134	0.0E+00
Cs-134	8.1E+01
I-135	1.7E+00
Cs-136	6.2E+00
Cs-137	1.1E+02
Ba-140	2.8E-01
La-140	4.9E-02
Ce-141	1.1E-02
Ce-143	1.2E-02
Ce-144	8.4E-02
Na-24	1.8E+00
Cr-51	4.0E+00
Mn-54	1.3E+00
Fe-55	1.5E+00
Fe-59	9.4E-02
Co-58	1.6E+00
Co-60	7.4E-01
Zn-65	3.5E-01
Ba-137m(1)	1.1E+02
W-187	2.1E-01
Np-239	3.8E-01

(1) This nuclide is a daughter nuclide in secular equilibrium and the activity is that of the parent nuclide (Cs-137).

Table 12.2-18

SFP Demineralizer and Filter Source Terms

Nuclide	Demin. (Bq)	Filter (Bq)	Nuclide	Demin. (Bq)	Filter (Bq)
BR-84	0.00E+00	0.00E+00	RU-106	6.00E+07	0.00E+00
I-131	2.83E+11	0.00E+00	AG-110M	3.39E+09	0.00E+00
I-132	1.96E-23	0.00E+00	TE-129M	1.40E+09	0.00E+00
I-133	1.89E+08	0.00E+00	TE-129	0.00E+00	0.00E+00
I-134	0.00E+00	0.00E+00	TE-131M	1.89E+07	0.00E+00
I-135	1.33E+00	0.00E+00	TE-131	0.00E+00	0.00E+00
RB-88	0.00E+00	0.00E+00	TE-132	5.56E+09	0.00E+00
CS-134	2.78E+11	0.00E+00	BA-137M	3.81E+11	0.00E+00
CS-136	1.33E+10	0.00E+00	BA-140	6.56E+08	0.00E+00
CS-137	3.81E+11	0.00E+00	LA-140	4.51E+06	0.00E+00
NA-24	3.18E+05	0.00E+00	CE-141	3.48E+07	0.00E+00
SR-89	8.80E+08	0.00E+00	CE-143	4.92E+05	0.00E+00
SR-90	3.24E+08	0.00E+00	CE-144	3.14E+08	0.00E+00
SR-91	1.16E+01	0.00E+00	W-187	1.53E+06	0.00E+00
Y-91M	0.00E+00	0.00E+00	NP-239	1.02E+08	0.00E+00
Y-91	2.23E+09	0.00E+00	CR-51	1.22E+10	1.21E+10
Y-93	2.79E+00	0.00E+00	MN-54	4.86E+09	4.88E+09
ZR-95	4.51E+08	4.50E+08	FE-55	5.69E+09	5.71E+09
NB-95	1.25E+08	0.00E+00	FE-59	3.12E+08	3.11E+08
MO-99	5.31E+09	0.00E+00	CO-58	5.59E+09	5.59E+09
TC-99M	4.46E-03	0.00E+00	CO-60	2.82E+09	2.82E+09
RU-103	4.24E+07	0.00E+00	ZN-65	1.30E+09	1.30E+09

Table 12.2-19 (1 of 3)

Steam Generator Blowdown and Condensate Polishing System Source Terms (0.25 % Fuel Defect)

Isotope	SG Water (Bq/g)	SG Steam (Bq/g)	Blowdown Mixed Bed (Bq)	Blowdown Pre-Filter (Bq)	Blowdown Post-Filter (Bq)	Condensate (Bq/g)	CPS Cation Bed (Bq)	CPS Mixed Bed (Bq)	Flash Tank (Bq)
Kr-85m	0.00E+00	1.29E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	5.49E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	1.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	2.80E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	5.49E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	0.00E+00	3.34E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	3.58E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	7.32E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	7.32E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-137	0.00E+00	1.70E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	6.41E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	7.65E-02	7.65E-04	4.71E+06	0.00E+00	0.00E+00	2.43E-04	0.00E+00	1.28E+05	7.48E+05
I-131	2.99E+01	2.99E-01	6.73E+11	0.00E+00	0.00E+00	9.50E-02	0.00E+00	1.71E+10	2.92E+08
I-132	5.44E+00	5.44E-02	1.46E+09	0.00E+00	0.00E+00	1.73E-02	0.00E+00	3.97E+07	5.32E+07
I-133	4.05E+01	4.05E-01	9.90E+10	0.00E+00	0.00E+00	1.29E-01	0.00E+00	2.70E+09	3.96E+08
I-134	2.16E+00	2.16E-02	2.18E+08	0.00E+00	0.00E+00	6.87E-03	0.00E+00	5.94E+06	2.11E+07
I-135	2.08E+01	2.08E-01	1.63E+10	0.00E+00	0.00E+00	6.61E-02	0.00E+00	4.44E+08	2.03E+08
Rb-88	5.96E+00	2.98E-02	2.06E+08	0.00E+00	0.00E+00	1.36E-02	4.00E+06	4.00E+05	5.83E+07

Table 12.2-19 (2 of 3)

Isotope	SG Water (Bq/g)	SG Steam (Bq/g)	Blowdown Mixed Bed (Bq)	Blowdown Pre-Filter (Bq)	Blowdown Post-Filter (Bq)	Condensate (Bq/g)	CPS Cation Bed (Bq)	CPS Mixed Bed (Bq)	Flash Tank (Bq)
Cs-134	4.77E+00	2.39E-02	1.55E+12	0.00E+00	0.00E+00	1.08E-02	5.27E+08	5.63E+08	4.67E+07
Cs-136	6.50E-01	3.25E-03	2.36E+10	0.00E+00	0.00E+00	1.48E-03	6.65E+07	3.75E+07	6.35E+06
Cs-137	5.52E+00	2.76E-02	1.94E+12	0.00E+00	0.00E+00	1.25E-02	6.10E+08	6.61E+08	5.40E+07
Cr-51	6.78E-01	3.39E-03	5.20E+10	4.73E+10	4.73E+08	1.54E-03	7.23E+07	5.60E+07	9.87E+08
Mn-54	7.85E-02	3.93E-04	2.28E+10	2.07E+10	2.07E+08	7.14E-05	3.46E+06	3.63E+06	5.94E+10
Fe-55	5.88E-02	2.94E-04	1.95E+10	1.77E+10	1.77E+08	5.35E-05	2.60E+06	2.79E+06	6.80E+09
Fe-59	1.47E-02	7.36E-05	1.73E+09	1.57E+09	1.57E+07	1.34E-05	6.37E+05	5.57E+05	6.89E+10
Co-58	2.25E-01	1.13E-03	3.71E+10	3.37E+10	3.37E+08	2.05E-04	9.82E+06	9.28E+06	8.52E+09
Co-60	2.60E-02	1.30E-04	8.90E+09	8.09E+09	8.09E+07	2.36E-05	1.15E+06	1.24E+06	1.07E+09
Zr-95	1.91E-02	9.53E-05	2.96E+09	2.69E+09	2.69E+07	1.73E-05	8.30E+05	7.75E+05	8.03E+08
Zn-65	2.50E-02	1.25E-04	6.90E+09	6.27E+09	6.27E+07	2.27E-05	1.10E+06	1.15E+06	1.91E+08
N-16	7.38E-01	3.69E-03	7.09E+03	0.00E+00	0.00E+00	6.71E-04	5.52E+01	5.52E+00	7.22E+06
Na-24	2.08E+00	1.04E-02	3.64E+09	0.00E+00	0.00E+00	1.89E-03	2.72E+07	2.83E+06	2.04E+07
Sr-89	4.03E-02	2.02E-04	5.35E+09	0.00E+00	0.00E+00	3.67E-05	1.75E+06	1.58E+06	3.94E+05
Sr-90	2.70E-03	1.35E-05	9.48E+08	0.00E+00	0.00E+00	2.45E-06	1.19E+05	1.29E+05	2.64E+04
Sr-91	5.33E-02	2.67E-04	6.01E+07	0.00E+00	0.00E+00	4.85E-05	4.65E+05	4.68E+04	5.21E+05
Y-91m	1.47E-02	7.36E-05	1.43E+06	0.00E+00	0.00E+00	1.34E-05	1.11E+04	1.11E+03	1.44E+05
Y-91	5.95E-03	2.98E-05	8.64E+08	0.00E+00	0.00E+00	5.41E-06	2.59E+05	2.38E+05	5.82E+04
Y-93	1.28E-03	6.41E-06	1.52E+06	0.00E+00	0.00E+00	1.16E-06	1.18E+04	1.19E+03	1.25E+04
Nb-95	6.41E-03	3.20E-05	6.11E+08	0.00E+00	0.00E+00	5.82E-06	2.75E+05	2.28E+05	6.27E+04

Table 12.2-19 (3 of 3)

Isotope	SG Water (Bq/g)	SG Steam (Bq/g)	Blowdown Mixed Bed (Bq)	Blowdown Pre-Filter (Bq)	Blowdown Post-Filter (Bq)	Condensate (Bq/g)	CPS Cation Bed (Bq)	CPS Mixed Bed (Bq)	Flash Tank (Bq)
Mo-99	3.42E+00	1.71E-02	2.68E+10	0.00E+00	0.00E+00	3.11E-03	1.08E+08	2.09E+07	3.35E+07
Tc-99m	1.70E+00	8.48E-03	1.18E+09	0.00E+00	0.00E+00	1.54E-03	9.18E+06	9.18E+05	1.66E+07
Ru-103	2.15E-03	1.08E-05	2.28E+08	0.00E+00	0.00E+00	1.96E-06	9.28E+04	7.91E+04	2.10E+04
Ru-106	9.16E-04	4.58E-06	2.74E+08	0.00E+00	0.00E+00	8.33E-07	4.04E+04	4.26E+04	8.96E+03
Ag-110m	6.38E-02	3.19E-04	1.77E+10	0.00E+00	0.00E+00	5.80E-05	2.81E+06	2.93E+06	6.24E+05
Te-129m	7.32E-02	3.66E-04	6.78E+09	0.00E+00	0.00E+00	6.66E-05	3.14E+06	2.58E+06	7.16E+05
Te-129	3.91E-02	1.96E-04	5.24E+06	0.00E+00	0.00E+00	3.56E-05	4.08E+04	4.08E+03	3.83E+05
Te-131m	3.35E-01	1.68E-03	1.17E+09	0.00E+00	0.00E+00	3.05E-04	7.33E+06	9.12E+05	3.28E+06
Te-131	3.69E-02	1.85E-04	1.79E+06	0.00E+00	0.00E+00	3.35E-05	1.40E+04	1.40E+03	3.61E+05
Te-132	2.39E+00	1.20E-02	2.18E+10	0.00E+00	0.00E+00	2.18E-03	7.87E+07	1.69E+07	2.34E+07
Ba-137m	5.52E+00	2.76E-02	1.94E+12	0.00E+00	0.00E+00	1.25E-02	6.10E+08	6.61E+08	5.40E+07
Ba-140	5.02E-02	2.51E-04	1.79E+09	0.00E+00	0.00E+00	4.57E-05	2.05E+06	1.15E+06	4.91E+05
La-140	1.65E-02	8.25E-05	7.74E+07	0.00E+00	0.00E+00	1.50E-05	4.23E+05	6.03E+04	1.61E+05
Ce-141	1.88E-03	9.40E-06	1.70E+08	0.00E+00	0.00E+00	1.71E-06	8.07E+04	6.57E+04	1.84E+04
Ce-143	4.87E-03	2.44E-05	1.87E+07	0.00E+00	0.00E+00	4.43E-06	1.13E+05	1.46E+04	4.77E+04
Ce-144	5.50E-03	2.75E-05	1.57E+09	0.00E+00	0.00E+00	5.00E-06	2.42E+05	2.54E+05	5.38E+04
W-187	1.15E-01	5.73E-04	3.19E+08	0.00E+00	0.00E+00	1.04E-04	2.16E+06	2.48E+05	1.12E+06
Np-239	1.05E-01	5.23E-04	6.86E+08	0.00E+00	0.00E+00	9.51E-05	3.10E+06	5.34E+05	1.02E+06

Table 12.2-20 (1 of 2)

Gaseous Radwaste System Source Terms (0.25 % Fuel Defect)

Nuclide	At Inlet (1)		Buildup Activity on Charcoal Bed										At Outlet	
			Guard Bed		1st Delay Bed		2nd Delay Bed		3rd Delay Bed		4th Delay Bed			
	μCi//cm^3	Bq/cm^3	μCi	Bq	μCi	Bq	μCi	Bq	μCi	Bq	μCi	Bq	μCi//cm^3	Bq/cm^3
Kr-85m	8.70E+00	3.22E+05	2.24E+05	8.28E+09	9.92E+07	3.67E+12	3.85E+06	1.43E+11	1.50E+05	5.53E+09	5.81E+03	2.15E+08	1.98E-05	7.32E-01
Kr-85	3.73E+01	1.38E+06	9.60E+05	3.55E+10	1.44E+09	5.32E+13	1.44E+09	5.32E+13	1.44E+09	5.32E+13	1.44E+09	5.32E+13	3.73E+01	1.38E+06
Kr-87	6.71E+00	2.48E+05	1.72E+05	6.38E+09	2.26E+07	8.35E+11	2.41E+02	8.93E+06	2.58E-03	9.54E+01	2.76E-08	1.02E-03	8.75E-20	3.24E-15
Kr-88	1.89E+01	6.99E+05	4.86E+05	1.80E+10	1.41E+08	5.22E+12	8.37E+05	3.10E+10	4.97E+03	1.84E+08	2.95E+01	1.09E+06	2.34E-08	8.65E-04
Xe-131m	3.71E+01	1.37E+06	9.54E+05	3.53E+10	1.35E+10	4.99E+14	7.00E+09	2.59E+14	3.64E+09	1.35E+14	1.89E+09	6.98E+13	2.70E+00	9.98E+04
Xe-133m	2.23E+00	8.24E+04	5.73E+04	2.12E+09	3.01E+08	1.11E+13	8.55E+06	3.16E+11	2.43E+05	8.99E+09	6.90E+03	2.55E+08	1.45E-06	5.37E-02
Xe-133	2.40E+03	8.88E+07	6.17E+07	2.28E+12	6.19E+11	2.29E+16	1.40E+11	5.18E+15	3.16E+10	1.17E+15	7.16E+09	2.65E+14	6.27E+00	2.32E+05
Xe-135m	4.93E+00	1.82E+05	1.27E+05	4.69E+09	3.33E+06	1.23E+11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	4.94E+01	1.83E+06	1.27E+06	4.70E+10	1.19E+09	4.40E+13	1.36E+00	5.02E+04	1.55E-09	5.73E-05	1.77E-18	6.54E-14	8.39E-35	3.10E-30
Xe-137	1.13E+00	4.19E+04	2.91E+04	1.08E+09	1.91E+05	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	4.19E+00	1.55E+05	1.08E+05	3.99E+09	2.62E+06	9.69E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	1.99E-06	7.37E-02	2.80E+00	1.03E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	2.48E-04	9.16E+00	1.27E+05	4.68E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	6.59E-05	2.44E+00	4.01E+02	1.48E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	3.57E-04	1.32E+01	1.96E+04	7.27E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	4.17E-05	1.54E+00	9.67E+01	3.58E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.01E-04	7.44E+00	3.52E+03	1.30E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-88	1.89E+01	6.99E+05	4.86E+05	1.80E+10	1.41E+08	5.22E+12	8.37E+05	3.10E+10	4.97E+03	1.84E+08	2.95E+01	1.09E+06	2.34E-08	8.65E-04
Cs-137	1.13E+00	4.19E+04	2.91E+04	1.08E+09	1.91E+05	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	1.13E+00	4.19E+04	2.91E+04	1.08E+09	1.91E+05	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

(1) The continuous venting flowrates for the gas stripper, reactor drain tank, volume control tank and equipment drain tank are 0.32 scfm, 0.024 scfm, 0.004 scfm and 0.005 scfm, respectively.

Table 12.2-20 (2 of 2)

Nuclide	Header Drain Tank		Waste Gas Dryer		Vessel for HEPA Filter	
	μCi	Bq	μCi	Bq	μCi	Bq
Kr-85m	4.93E+06	1.82E+11	1.68E+05	6.20E+09	1.65E+00	6.10E+04
Kr-85	2.11E+07	7.82E+11	7.19E+05	2.66E+10	3.11E+06	1.15E+11
Kr-87	3.80E+06	1.41E+11	1.29E+05	4.78E+09	7.30E-15	2.70E-10
Kr-88	1.07E+07	3.96E+11	3.64E+05	1.35E+10	1.95E-03	7.21E+01
Xe-131m	2.10E+07	7.77E+11	7.13E+05	2.64E+10	2.25E+05	8.32E+09
Xe-133m	1.26E+06	4.67E+10	4.29E+04	1.59E+09	1.21E-01	4.48E+03
Xe-133	1.36E+09	5.03E+13	4.62E+07	1.71E+12	5.23E+05	1.93E+10
Xe-135m	2.79E+06	1.03E+11	9.48E+04	3.51E+09	6.99E-30	2.59E-25
Xe-135	2.80E+07	1.04E+12	9.53E+05	3.53E+10	0.00E+00	0.00E+00
Xe-137	6.41E+05	2.37E+10	2.18E+04	8.07E+08	0.00E+00	0.00E+00
Xe-138	2.37E+06	8.78E+10	8.07E+04	2.99E+09	0.00E+00	0.00E+00
Br-84	1.13E+00	4.17E+04	3.84E-02	1.42E+03	0.00E+00	0.00E+00
I-131	1.40E+02	5.19E+06	4.77E+00	1.76E+05	0.00E+00	0.00E+00
I-132	3.73E+01	1.38E+06	1.27E+00	4.70E+04	0.00E+00	0.00E+00
I-133	2.02E+02	7.48E+06	6.87E+00	2.54E+05	0.00E+00	0.00E+00
I-134	2.36E+01	8.73E+05	8.02E-01	2.97E+04	0.00E+00	0.00E+00
I-135	1.14E+02	4.21E+06	3.87E+00	1.43E+05	0.00E+00	0.00E+00
Rb-88	1.07E+07	3.96E+11	3.64E+05	1.35E+10	1.95E-03	7.21E+01
Cs-137	6.41E+05	2.37E+10	2.18E+04	8.07E+08	0.00E+00	0.00E+00
Ba-137m	6.41E+05	2.37E+10	2.18E+04	8.07E+08	0.00E+00	0.00E+00

Table 12.2-21 (1 of 3)

Liquid Radwaste System Tank Source Terms (Bq/cm^3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
Br-84	6.27E+01	8.62E+01	1.96E+00	2.00E-03
I-131	7.94E+03	1.09E+04	2.48E+02	2.17E+01
I-132	2.13E+03	2.93E+03	6.66E+01	4.33E-01
I-133	1.13E+04	1.55E+04	3.52E+02	1.30E+01
I-134	1.30E+03	1.79E+03	4.07E+01	6.78E-02
I-135	6.40E+03	8.80E+03	2.00E+02	2.58E+00
Rb-88	7.33E+03	1.01E+04	2.29E+02	6.53E-01
Cs-134	1.13E+03	1.55E+03	3.52E+01	1.76E+01
Cs-136	1.54E+02	2.12E+02	4.81E+00	2.21E+00
Cs-137	1.30E+03	1.79E+03	4.07E+01	2.03E+01
Na-24	5.79E+02	7.96E+02	1.81E+01	5.07E-02
Cr-51	1.75E+02	2.41E+02	5.48E+00	5.27E-02
Mn-54	2.03E+01	2.79E+01	6.34E-01	6.31E-03
Fe-55	1.52E+01	2.09E+01	4.75E-01	4.74E-03
Fe-59	3.81E+00	5.24E+00	1.19E-01	1.16E-03
Co-58	5.82E+01	8.01E+01	1.82E+00	1.79E-02
Co-60	6.72E+00	9.24E+00	2.10E-01	2.10E-03
Zn-65	6.46E+00	8.89E+00	2.02E-01	2.01E-03

Table 12.2-21 (2 of 3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
Sr-89	1.04E+01	1.43E+01	3.26E-01	3.19E-03
Sr-90	6.98E-01	9.59E-01	2.18E-02	2.18E-04
Sr-91	1.54E+01	2.12E+01	4.81E-01	8.91E-04
Y-91m	8.99E+00	1.24E+01	2.81E-01	5.67E-04
Y-91	1.54E+00	2.12E+00	4.81E-02	5.00E-04
Y-93	3.68E-01	5.06E-01	1.15E-02	2.24E-05
Zr-95	4.93E+00	6.78E+00	1.54E-01	1.51E-03
Nb-95	1.66E+00	2.28E+00	5.18E-02	5.47E-04
Mo-99	8.99E+02	1.24E+03	2.81E+01	1.95E-01
Tc-99m	5.22E+02	7.17E+02	1.63E+01	1.75E-01
Ru-103	5.57E-01	7.66E-01	1.74E-02	1.69E-04
Ru-106	2.37E-01	3.26E-01	7.40E-03	7.37E-04
Ag-110m	1.65E+01	2.27E+01	5.15E-01	5.12E-03
Te-129m	1.89E+01	2.60E+01	5.92E-01	5.73E-03
Te-129	2.01E+01	2.77E+01	6.29E-01	3.73E-03
Te-131m	8.99E+01	1.24E+02	2.81E+00	1.33E-02
Te-131	3.55E+01	4.88E+01	1.11E+00	2.48E-03
Te-132	6.27E+02	8.62E+02	1.96E+01	1.43E-01
Ba-137m	1.18E+03	1.63E+03	3.70E+01	1.90E+01
Ba-140	1.30E+01	1.79E+01	4.07E-01	3.74E-03

Table 12.2-21 (3 of 3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
La-140	4.38E+00	6.03E+00	1.37E-01	2.45E-03
Ce-141	4.86E-01	6.69E-01	1.52E-02	1.47E-04
Ce-143	1.30E+00	1.79E+00	4.07E-02	2.05E-04
Ce-144	1.42E+00	1.95E+00	4.44E-02	4.42E-04
W-187	3.10E+01	4.27E+01	9.70E-01	3.95E-03
Np-239	2.76E+01	3.79E+01	8.62E-01	5.63E-03

Table 12.2-22 (1 of 3)

Liquid Radwaste System Component Source Terms (Bq)

Nuclide	Reverse Osmosis	Cation Bed	Mixed Bed 1	Mixed Bed 2
Na-24	2.14E+11	2.15E+10	2.36E+09	2.15E+07
Cr-51	5.58E+11	5.58E+10	6.14E+09	5.58E+07
Mn-54	7.11E+10	7.11E+09	7.82E+08	7.11E+06
Fe-55	5.36E+10	5.36E+09	5.90E+08	5.36E+06
Co-58	1.98E+11	1.98E+10	2.17E+09	1.98E+07
Fe-59	1.26E+10	1.26E+09	1.39E+08	1.26E+06
Co-60	2.37E+10	2.37E+09	2.61E+08	2.37E+06
Zn-65	2.26E+10	2.26E+09	2.49E+08	2.26E+06
Br-84	8.79E+08	1.23E+07	1.02E+08	9.40E+05
Rb-88	6.06E+10	6.78E+09	4.41E+08	3.39E+08
Sr-89	3.49E+10	3.49E+09	3.84E+08	3.49E+06
Y-89m	3.49E+06	3.49E+05	3.83E+04	3.49E+02
Sr-90	2.47E+09	2.47E+08	2.71E+07	2.47E+05
Y-90m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	1.49E+09	1.49E+08	1.64E+07	1.49E+05
Sr-91	3.67E+09	3.69E+08	4.05E+07	3.69E+05
Y-91m	2.34E+09	2.35E+08	2.58E+07	2.35E+05
Y-91	5.52E+09	5.53E+08	6.08E+07	5.53E+05
Y-93	9.27E+07	9.30E+06	1.02E+06	9.30E+03
Zr-93	9.40E-01	9.40E-02	1.03E-02	9.40E-05
Nb-93m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.67E+10	1.67E+09	1.83E+08	1.67E+06
Nb-95m	1.69E+08	1.69E+07	1.86E+06	1.69E+04
Nb-95	6.74E+09	6.74E+08	7.42E+07	6.74E+05
Mo-99	1.31E+12	1.31E+11	1.44E+10	1.31E+08
Tc-99m	1.20E+12	1.21E+11	1.33E+10	1.21E+08
Tc-99	6.99E+04	6.99E+03	7.68E+02	6.99E+00
Ru-103	1.83E+09	1.83E+08	2.01E+07	1.83E+05

Table 12.2-22 (2 of 3)

Nuclide	Reverse Osmosis	Cation Bed	Mixed Bed 1	Mixed Bed 2
Rh-103m	1.82E+09	1.82E+08	2.00E+07	1.82E+05
Ru-106	8.31E+09	8.31E+08	9.14E+07	8.31E+05
Rh-106m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh-106	8.31E+09	8.31E+08	9.14E+07	8.31E+05
Ag-110m	5.76E+10	5.76E+09	6.34E+08	5.76E+06
Ag-110	7.49E+08	7.49E+07	8.24E+06	7.49E+04
Te-129m	6.15E+10	6.15E+09	6.76E+08	6.15E+06
Te-129	3.96E+10	3.96E+09	4.36E+08	3.96E+06
I-129	3.08E+01	3.08E+00	3.38E-01	3.08E-03
Te-131m	6.58E+10	6.59E+09	7.24E+08	6.59E+06
Te-131	1.22E+10	1.23E+09	1.35E+08	1.23E+06
I-131	1.98E+13	4.49E+09	2.18E+12	1.98E+10
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	1.01E+12	1.01E+11	1.12E+10	1.01E+08
I-132	1.13E+12	1.01E+11	2.47E+10	2.25E+08
I-133	5.82E+12	2.21E+09	6.41E+11	5.83E+09
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	2.90E+10	2.56E+08	3.31E+09	3.03E+07
Cs-134	3.97E+12	3.97E+11	2.21E+10	1.99E+10
I-135	1.06E+12	1.26E+09	1.17E+11	1.07E+09
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-136	4.36E+11	4.36E+10	2.42E+09	2.18E+09
Cs-137	4.61E+12	4.61E+11	2.56E+10	2.30E+10
Ba-137m	4.31E+12	4.31E+11	2.40E+10	2.15E+10
Ba-140	3.68E+10	3.68E+09	4.05E+08	3.68E+06

Table 12.2-22 (3 of 3)

Nuclide	Reverse Osmosis	Cation Bed	Mixed Bed 1	Mixed Bed 2
La-140	3.21E+10	3.21E+09	3.54E+08	3.21E+06
Ce-141	1.57E+09	1.57E+08	1.73E+07	1.57E+05
Ce-143	1.04E+09	1.05E+08	1.15E+07	1.05E+05
Pr-143	3.02E+08	3.02E+07	3.32E+06	3.02E+04
Ce-144	4.97E+09	4.98E+08	5.47E+07	4.98E+05
Pr-144	4.96E+09	4.96E+08	5.46E+07	4.96E+05
W-187	1.82E+10	1.82E+09	2.00E+08	1.82E+06
Np-239	3.53E+10	3.53E+09	3.89E+08	3.53E+06

Table 12.2-23

Solid Radwaste System Tank Source Terms (Bq)

Nuclide	Spent Resin Long-Term Storage Tank (1)	Low-Activity Spent Resin Tank (2)
Na-24	7.01E+11	7.16E+10
Cr-51	9.49E+12	1.86E+11
Mn-54	1.45E+13	2.37E+10
Fe-55	3.40E+13	1.79E+10
Co-58	8.28E+12	6.60E+10
Fe-59	3.34E+11	4.20E+09
Co-60	2.38E+13	7.90E+09
Zn-65	3.57E+12	7.53E+09
Br-84	2.73E+09	3.46E+08
Rb-88	1.15E+11	2.27E+10
Sr-89	1.01E+12	1.16E+10
Y-89m	0.00E+00	1.16E+06
Sr-90	3.96E+12	8.23E+08
Y-90	0.00E+00	4.97E+08
Sr-91	1.20E+10	1.23E+09
Y-91m	9.40E+07	7.83E+08
Y-91	1.63E+07	1.84E+09
Y-93	3.80E+06	3.10E+07
Zr-93	0.00E+00	3.13E-01
Zr-95	6.34E+11	5.56E+09
Nb-95m	0.00E+00	5.63E+07
Nb-95	1.10E+11	2.25E+09
Mo-99	4.81E+12	4.37E+11
Tc-99m	2.50E+11	4.03E+11
Tc-99	0.00E+00	2.33E+04
Ru-103	4.32E+10	6.10E+08
Rh-103m	0.00E+00	6.07E+08
Ru-106	2.01E+11	2.77E+09

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank
Rh-106	0.00E+00	2.77E+09
Ag-110m	9.29E+12	1.92E+10
Ag-110	0.00E+00	2.50E+08
Te-129m	1.31E+12	2.05E+10
Te-129	1.92E+09	1.32E+10
I-129	0.00E+00	1.03E+01
Te-131m	2.12E+11	2.20E+10
Te-131	1.21E+09	4.10E+09
I-131	4.04E+11	6.61E+12
Te-132	4.04E+12	3.37E+11
I-132	4.04E+11	3.78E+11
I-133	1.92E+13	1.95E+12
I-134	9.30E+10	1.08E+10
Cs-134	1.20E+15	1.32E+12
I-135	3.54E+12	3.58E+11
Cs-136	2.34E+12	1.45E+11
Cs-137	4.60E+15	1.53E+12
Ba-137m	4.60E+15	1.43E+12
Ba-140	3.21E+11	1.23E+10
La-140	1.40E+10	1.07E+10
Ce-141	3.11E+10	5.23E+08
Ce-143	3.51E+09	3.50E+08
Pr-143	0.00E+00	1.01E+08
Ce-144	9.18E+11	1.66E+09
Pr-144	0.00E+00	1.65E+09
W-187	6.01E+10	6.07E+09
Np-239	1.30E+11	1.18E+10

(1) Source terms for the spent resin long-term storage tank are calculated based on a maximum of 10 years of cumulative radioactive resin batches from the CVCS demineralizers with decay.

(2) Source terms for the low-activity spent resin storage tank are calculated based on the using the source term for LWMS spent resin and multiplying by a factor of 3 for conservative shielding design.

Table 12.2-24

Waste Drum Storage Area Source Terms (Bq)

Nuclide	Activity
I-131	2.96E+12
I-132	6.15E-82
I-133	2.12E+03
I-134	3.77E-235
I-135	5.15E-20
Cs-134	5.47E+12
Cs-136	1.57E+11
Cs-137	6.56E+12
Cr-51	4.11E+11
Mn-54	9.70E+10
Fe-55	7.38E+10
Fe-59	1.19E+10
Co-58	2.55E+11
Co-60	2.31E+11
Zr-95	2.05E+10
Zn-65	2.95E+10
Na-24	1.02E-02
Sr-89	3.43E+10
Sr-90	3.45E+09
Sr-91	1.17E-12
Y-91m	1.10E-251

Nuclide	Activity
Y-91	5.35E+09
Y-93	6.34E-13
Nb-95	1.42E+10
Mo-99	2.33E+09
Tc-99m	2.57E-24
Ru-103	1.63E+09
Ru-106	1.08E+10
Ag-110m	7.64E+10
Sb-125	5.30E+10
Te-129m	1.42E-176
Te-129	2.66E+04
Te-131m	0.00E+00
Te-132	5.26E+09
Ba-137m	5.86E+12
Ba-140	1.27E+10
La-140	8.88E+04
Ce-141	1.27E+09
Ce-143	1.75E+03
Ce-144	6.55E+09
W-187	1.24E+02
Np-239	2.00E+07

Table 12.2-25 (1 of 4)

Airborne Radioactivity Concentrations

Reactor Containment Building (Normal Operation)

Nuclide	Airborne Concentration (1) (Bq/cm^3)	10 CFR 20 Appendix B (Bq/cm^3)	DAC Fraction	Nuclide	Airborne Concentration (Bq/cm^3)	10 CFR 20 Appendix B (Bq/cm^3)	DAC Fraction
Kr-85m	8.60E-02	7.40E-01	1.16E-01	Zn-65	9.83E-02	3.70E-03	2.66E+01
Kr-85	7.13E+02	3.70E+00	1.93E+02	Sr-89	4.15E-02	2.22E-03	1.87E+01
Kr-87	1.92E-02	1.85E-01	1.04E-01	Sr-90	2.05E-02	7.40E-05	2.77E+02
Kr-88	1.19E-01	7.40E-02	1.61E+00	Sr-91	4.80E-04	3.70E-02	1.30E-02
Xe-131m	2.34E+01	1.48E+01	1.58E+00	Y-91m	2.45E-05	2.59E+00	9.45E-06
Xe-133m	2.62E-01	3.70E+00	7.09E-02	Y-91	7.09E-03	1.85E-03	3.83E+00
Xe-133	6.73E+02	3.70E+00	1.82E+02	Y-93	1.22E-05	3.70E-02	3.30E-04
Xe-135m	2.79E-03	3.33E-01	8.38E-03	Zr-95	2.48E-02	1.85E-03	1.34E+01
Xe-135	9.95E-01	3.70E-01	2.69E+00	Nb-95	4.60E-03	1.85E-02	2.48E-01
Xe-138	2.26E-03	1.48E-01	1.53E-02	Mo-99	1.95E-01	2.22E-02	8.79E+00
Br-84	1.09E-04	7.40E-01	1.47E-04	Tc-99m	1.03E-02	2.22E+00	4.66E-03
I-131	5.04E+00	7.40E-04	6.81E+03	Ru-103	1.73E-03	1.11E-02	1.56E-01
I-132	1.61E-02	1.11E-01	1.45E-01	Ru-106	4.43E-03	1.85E-04	2.39E+01
I-133	7.71E-01	3.70E-03	2.08E+02	Ag-110m	2.53E-01	1.48E-03	1.71E+02
I-134	3.75E-03	7.40E-01	5.07E-03	Te-129m	5.02E-02	3.70E-03	1.36E+01
I-135	1.39E-01	2.59E-02	5.36E+00	Te-129	7.67E-05	1.11E+00	6.91E-05
Rb-88	7.13E-03	1.11E+00	6.43E-03	Te-131m	8.87E-03	7.40E-03	1.20E+00
Cs-134	2.65E+01	1.48E-03	1.79E+04	Te-131	4.88E-05	7.40E-02	6.59E-04
Cs-136	1.59E-01	1.11E-02	1.44E+01	Te-132	1.61E-01	3.33E-03	4.84E+01
Cs-137	3.83E+01	2.22E-03	1.73E+04	Ba-140	1.31E-02	2.22E-02	5.92E-01
Na-24	2.86E-02	7.40E-02	3.86E-01	La-140	5.81E-04	1.85E-02	3.14E-02
Cr-51	3.84E-01	2.96E-01	1.30E+00	Ce-141	1.25E-03	7.40E-03	1.69E-01
Mn-54	3.52E-01	1.11E-02	3.17E+01	Ce-143	1.41E-04	2.59E-02	5.46E-03
Fe-55	3.75E-01	2.96E-02	1.27E+01	Ce-144	2.34E-02	2.22E-04	1.06E+02
Fe-59	1.35E-02	3.70E-03	3.65E+00	W-187	2.44E-03	1.48E-01	1.65E-02
Co-58	3.24E-01	1.11E-02	2.92E+01	Np-239	5.44E-03	3.33E-02	1.63E-01
Co-60	1.83E-01	3.70E-04	4.94E+02	H-3	7.92E-01	7.40E-01	1.07E+00
						Sum of DAC fractions	4.39E+04

(1) The airborne activity for other nuclides (primarily particulates) in the containment building is calculated based on the calculation of flashing fraction.

Table 12.2-25 (2 of 4)

Reactor Containment Building (48 hr after Shutdown)

Nuclide	Airborne Concentration (Bq/cm^3)	10 CFR 20 Appendix B (Bq/cm^3)	DAC Fraction	Nuclide	Airborne Concentration (Bq/cm^3)	10 CFR 20 Appendix B (Bq/cm^3)	DAC Fraction
Kr-85m	1.41E-06	7.40E-01	1.90E-06	Zn-65	4.05E-07	3.70E-03	1.10E-04
Kr-85	8.95E-03	3.70E+00	2.42E-03	Sr-89	6.40E-07	2.22E-03	2.88E-04
Kr-87	1.14E-14	1.85E-01	6.14E-14	Sr-90	4.40E-08	7.40E-05	5.94E-04
Kr-88	4.60E-08	7.40E-02	6.21E-07	Sr-91	3.10E-08	3.70E-02	8.39E-07
Xe-131m	7.98E-03	1.48E+01	5.40E-04	Y-91m	4.12E-24	2.59E+00	1.59E-24
Xe-133m	2.92E-04	3.70E+00	7.90E-05	Y-91	9.48E-08	1.85E-03	5.12E-05
Xe-133	4.50E-01	3.70E+00	1.22E-01	Y-93	9.14E-10	3.70E-02	2.47E-08
Xe-135m	0.00E+00	3.33E-01	0.00E+00	Zr-95	3.04E-07	1.85E-03	1.64E-04
Xe-135	3.27E-04	3.70E-01	8.84E-04	Nb-95	1.00E-07	1.85E-02	5.43E-06
Xe-138	0.00E+00	1.48E-01	0.00E+00	Mo-99	3.45E-05	2.22E-02	1.56E-03
Br-84	0.00E+00	7.40E-01	0.00E+00	Tc-99m	1.45E-07	2.22E+00	6.54E-08
I-131	3.43E-05	7.40E-04	4.63E-02	Ru-103	3.39E-08	1.11E-02	3.05E-06
I-132	7.49E-12	1.11E-01	6.75E-11	Ru-106	1.49E-08	1.85E-04	8.04E-05
I-133	1.20E-05	3.70E-03	3.24E-03	Ag-110m	1.03E-06	1.48E-03	6.98E-04
I-134	4.24E-22	7.40E-01	5.73E-22	Te-129m	1.15E-06	3.70E-03	3.10E-04
I-135	2.32E-07	2.59E-02	8.95E-06	Te-129	7.22E-19	1.11E+00	6.50E-19
Rb-88	0.00E+00	1.11E+00	0.00E+00	Te-131m	1.91E-06	7.40E-03	2.58E-04
Cs-134	7.09E-05	1.48E-03	4.79E-02	Te-131	0.00E+00	7.40E-02	0.00E+00
Cs-136	8.74E-06	1.11E-02	7.88E-04	Te-132	2.60E-05	3.33E-03	7.82E-03
Cs-137	8.21E-05	2.22E-03	3.70E-02	Ba-140	7.38E-07	2.22E-02	3.32E-05
Na-24	4.13E-06	7.40E-02	5.59E-05	La-140	1.23E-07	1.85E-02	6.64E-06
Cr-51	1.05E-05	2.96E-01	3.56E-05	Ce-141	2.94E-08	7.40E-03	3.97E-06
Mn-54	1.27E-06	1.11E-02	1.15E-04	Ce-143	3.05E-08	2.59E-02	1.18E-06
Fe-55	9.56E-07	2.96E-02	3.23E-05	Ce-144	8.91E-08	2.22E-04	4.01E-04
Fe-59	2.33E-07	3.70E-03	6.29E-05	W-187	4.98E-07	1.48E-01	3.37E-06
Co-58	3.60E-06	1.11E-02	3.24E-04	Np-239	1.04E-06	3.33E-02	3.11E-05
Co-60	4.23E-07	3.70E-04	1.14E-03	H-3	1.74E-06	7.40E-01	2.35E-06
						Sum of DAC fractions	2.75E-01

Table 12.2-25 (3 of 4)

Auxiliary Building Cubicles (Normal Operation)

Cubicle	Airborne Radioactivity Concentration (Bq/cm^3)				Derived Air Concentration (DAC) Fraction				
	Kr, Xe	Br, I	H-3	Others	Kr, Xe	Br, I	H-3	Others	Total
CS Pump and Miniflow HX Rm (050-A01C,D)	7.37E-02	4.92E-07	9.82E-03	2.86E-06	1.99E-02	5.18E-04	1.33E-02	1.40E-03	3.50E-02
SI Pump Rm (050-A02C,D)	1.16E-01	7.81E-07	1.55E-02	4.54E-06	3.13E-02	8.18E-04	2.09E-02	2.19E-03	5.52E-02
Floor Drain Sump Pump Rm (055-A34A,B,C,D)	1.84E+00	5.34E-05	4.10E-02	1.05E-04	7.99E-01	2.61E-02	5.54E-02	1.55E-02	8.96E-01
Pipe Chase and Valve Rm (055-A14C)	9.44E-02	6.20E-07	1.27E-02	3.75E-06	2.54E-02	6.64E-04	1.72E-02	1.82E-03	4.50E-02
Shutdown Cooling HX Rm (055-A30A,B)	3.63E-02	2.39E-07	4.88E-03	1.44E-06	9.77E-03	2.55E-04	6.59E-03	6.98E-04	1.73E-02
Charging Pump Rm (055-A55B)	1.58E+00	2.68E-07	3.56E-02	1.12E-05	6.44E-01	1.39E-04	4.81E-02	3.47E-03	6.95E-01
Charging Pump Miniflow HX Rm (055-A43 A)	4.49E-01	1.23E-05	1.01E-02	3.07E-06	1.81E-01	6.37E-03	1.37E-02	9.87E-04	2.02E-01
Equipment Drain Tank Rm (055-A51B)	8.48E-02	1.77E-06	3.59E-04	2.53E-06	3.62E-02	1.03E-03	4.85E-04	7.14E-04	3.84E-02
Reactor Drain Pump Rm (055-A52B,A53B)	2.70E+00	8.76E-06	2.11E-03	1.75E-05	7.95E-01	5.20E-03	2.85E-03	3.93E-03	8.07E-01
Gas Stripper Rm (068-A06A)	2.14E+00	6.11E-07	3.53E-07		9.02E-01	3.04E-04	4.76E-07		9.02E-01
Filter and Demin. Valve Area (068-A10A)	1.90E+00	4.85E-05	3.67E-02	9.76E-05	7.92E-01	2.37E-02	4.96E-02	1.42E-02	8.79E-01
SFP Cleanup Pump Rm (078-A38A)	4.16E-02	2.87E-07	5.17E-03	1.53E-06	1.12E-02	3.00E-04	6.99E-03	7.33E-04	1.92E-02
Reactor Makeup Water Pump Rm (078-A49A)	2.49E-07	1.36E-12	1.20E-02	1.95E-10	6.57E-08	1.81E-09	1.63E-02	9.88E-08	1.63E-02
Holdup Pump Rm (078-A50B)	5.29E-04	5.88E-08	1.19E-02	1.11E-07	2.95E-04	6.30E-04	1.61E-02	4.05E-05	1.65E-02
Volume Control Tank Rm (100-A25A)	1.06E+00	1.62E-07	2.41E-021	5.66E-06	3.88E-01	9.22E-05	3.26E-02	2.37E-03	4.23E-01
Valve Rm (120-A23A)	1.81E-01	5.27E-06	5.01E-03	1.06E-05	7.89E-02	2.58E-03	6.78E-03	1.73E-03	9.00E-02
Fuel Handling Area (Normal Operation)			2.40E-02				3.20E-02		3.20E-02
Fuel Handling Area (Refueling)			5.70E-02				7.70E-02		7.70E-02

Table 12.2-25 (4 of 4)

Compound Building Cubicles (Normal Operation)

Cubicle	Airborne Radioactivity Concentration (Bq/cm^3)				Derived Air Concentration (DAC) Fraction				
	Kr, Xe	Br, I	H-3	Others	Kr, Xe	Br, I	H-3	Others	Total
Valve Rm (063-P07)	1.62E+00	1.03E-04	8.29E-02	1.55E-04	6.66E-01	5.24E-02	1.12E-01	3.13E-02	8.62E-01
Equipment Waste Pump Rm (063-P21,P22)	1.30E+00	3.72E-05	2.92E-02	6.29E-05	5.52E-01	1.85E-02	3.94E-02	1.10E-02	6.21E-01
Equipment Waste Tank Rm (063-P23,P24)	1.73E-01	4.87E-06	7.33E-04	7.60E-06	7.20E-02	2.46E-03	9.91E-04	1.47E-03	7.69E-02
Floor Drain Pump Rm (063-P25)	2.17E+00	5.93E-05	4.89E-02	8.40E-05	8.71E-01	3.08E-02	6.61E-02	1.84E-02	9.87E-01
Normal Sump Pump Rm (063-P26)	1.90E-01	5.52E-06	4.23E-03	1.09E-05	8.26E-02	2.69E-03	5.72E-03	1.60E-03	9.26E-02
Chemical Waste Pump Rm (063-P27)	5.50E-02	1.46E-06	1.13E-03	2.73E-06	2.18E-02	7.16E-04	1.52E-03	4.26E-04	2.45E-02
Floor Drain Tank Rm (063-P28,P29)	4.74E-01	1.30E-05	1.07E-02	1.88E-05	1.92E-01	6.73E-03	1.44E-02	4.02E-03	2.17E-01
Chemical Waste Tank Rm (063-P30,P31)	1.07E-02	2.83E-07	2.42E-04	3.80E-07	4.15E-03	1.51E-04	3.27E-04	9.13E-05	4.72E-03
Detergent Waste Tank and Pump Rm (063-P32)	0.00E+00	5.85E-08	1.02E-05	3.21E-08	0.00E+00	3.33E-05	1.38E-05	8.59E-06	5.57E-05
Chemical Drain Sump Pump Rm (063-P36)	1.00E-01	2.92E-06	2.24E-03	5.79E-06	4.37E-02	1.42E-03	3.02E-03	8.44E-04	4.90E-02
Monitor Tank Rm (063-P37)	0.00E+00	1.08E-08	0.00E+00	1.44E-07	0.00E+00	1.00E-05	0.00E+00	7.51E-05	8.52E-05
Monitor Tank Pump Rm (063-P54)	0.00E+00	2.07E-07	0.00E+00	2.58E-06	0.00E+00	1.84E-04	0.00E+00	1.34E-03	1.53E-03
Valve Rm (085-P06)	0.00E+00	1.98E-05	1.62E-02	2.86E-05	0.00E+00	1.02E-02	2.19E-02	6.10E-03	3.81E-02
Valve Rm (085-P15)	0.00E+00	4.61E-05	3.64E-02	7.48E-05	0.00E+00	2.31E-02	4.92E-02	1.37E-02	8.61E-02
Valve Rm (085-P16)	2.06E-01	5.96E-06	4.59E-03	1.08E-05	8.91E-02	2.92E-03	6.21E-03	1.64E-03	9.45E-02

Table 12.2-26 (1 of 14)

Source Terms for Post-Accident Shielding Analysis

a. Initial Release Fraction from the Core

Source Term[1]	Gap Release		Early In-Vessel Release Plus Gap Release	
	Nuclide Group[2]	Percentage	Nuclide Group[2]	Percentage
Liquid-containing systems[3]	2 and 3	5 %	2	40 %
			3	30 %
			4	5 %
			5	2 %
			6	0.25 %
			7	0.02 %
			8	0.05 %
Airborne	1, 2, and 3	5 %	1	100 %
			2	40 %
			3	30 %
			4	5 %
			5	2 %
			6	0.25 %
			7	0.02 %
			8	0.05 %

(1) The source terms represent the initial releases from the core into the reactor containment building sump water and atmosphere.

(2) Nuclide Group 1: Xe, Kr
Nuclide Group 2: I, Br
Nuclide Group 3: Cs, Rb
Nuclide Group 4: Te, Sb, Se
Nuclide Group 5: Sr, Ba
Nuclide Group 6: Co, Ru, Rh, Pd, Mo, Tc
Nuclide Group 7: Am, Cm, La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
Nuclide Group 8: Ce, Pu, Np

(3) The following components of the engineered safety systems use water in the IRWST during post-accident conditions:

- Shutdown Cooling System : shutdown cooling pumps, shutdown cooling mini flow heat exchanger
- Safety Injection System : safety injection pump
- Containment Spray System : containment spray pump, containment spray heat exchanger, containment spray mini flow heat exchanger
- Piping : associated pipes of the above systems

Table 12.2-26 (2 of 14)

b. Radioactive Concentration of Post-Accident Recirculating Water (Bq/cc) (1 of 5)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Se-84	2.75E+07	9.25E+01	0.00E+00	0.00E+00	0.00E+00
Se-85	1.37E+07	2.24E-21	0.00E+00	0.00E+00	0.00E+00
Se-87	2.03E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	2.25E+08	6.08E+07	5.26E-06	8.54E-88	0.00E+00
Br-85	2.59E+08	1.30E+02	0.00E+00	0.00E+00	0.00E+00
Br-87	4.25E+08	1.61E-11	0.00E+00	0.00E+00	0.00E+00
Br-88	4.28E+08	1.40E-58	0.00E+00	0.00E+00	0.00E+00
Br-89	2.89E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	0.00E+00	3.93E+00	2.73E+01	2.79E+01	2.78E+01
Kr-85m	0.00E+00	2.39E+06	6.80E+04	1.43E-05	1.16E-42
Rb-86	1.59E+06	1.59E+06	1.53E+06	1.23E+06	5.22E+05
Rb-88	5.74E+08	5.55E+07	2.55E-16	0.00E+00	0.00E+00
Rb-89	7.63E+08	5.16E+07	6.41E-20	0.00E+00	0.00E+00
Rb-90	7.20E+08	9.01E+01	0.00E+00	0.00E+00	0.00E+00
Rb-91	9.04E+08	2.17E-10	0.00E+00	0.00E+00	0.00E+00
Sr-89	4.68E+07	4.69E+07	4.63E+07	4.27E+07	3.11E+07
Sr-90	4.21E+06	4.21E+06	4.21E+06	4.21E+06	4.20E+06
Sr-91	6.30E+07	5.86E+07	1.09E+07	2.99E+02	9.65E-16
Sr-92	6.35E+07	4.92E+07	1.37E+05	1.38E-11	6.67E-73
Sr-95	5.93E+07	1.24E-34	0.00E+00	0.00E+00	0.00E+00
Y-90	4.44E+04	8.92E+04	9.97E+05	3.53E+06	4.20E+06
Y-91	5.74E+05	5.92E+05	9.13E+05	9.24E+05	7.03E+05
Y-91m	3.64E+05	1.98E+07	6.88E+06	1.88E+02	6.07E-16
Y-92	6.41E+05	1.04E+07	1.45E+06	1.08E-06	1.23E-53
Y-93	4.60E+05	4.29E+05	8.86E+04	4.52E+00	1.60E-16
Y-95	7.08E+05	1.35E+04	3.68E-36	0.00E+00	0.00E+00
Zr-93	0.00E+00	2.30E-05	2.80E-04	3.46E-04	3.46E-04
Zr-95	6.71E+05	6.71E+05	6.64E+05	6.22E+05	4.85E+05
Zr-97	6.37E+05	6.11E+05	2.38E+05	6.48E+02	9.53E-08
Nb-93m	0.00E+00	6.29E-11	2.30E-08	2.88E-07	1.32E-06
Nb-95	6.68E+05	6.68E+05	6.68E+05	6.65E+05	6.21E+05

Table 12.2-26 (3 of 14)

b. Radioactive Concentration of Post-Accident Recirculating Water (Bq/cc) (2 of 5)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Nb-95m	0.00E+00	4.28E+01	9.33E+02	3.79E+03	4.09E+03
Nb-97	0.00E+00	2.69E+05	2.56E+05	6.98E+02	1.03E-07
Nb-97m	0.00E+00	5.80E+05	2.26E+05	6.14E+02	9.04E-08
Nb-99	4.14E+05	6.81E-71	0.00E+00	0.00E+00	0.00E+00
Mo-99	8.74E+06	8.65E+06	6.79E+06	1.50E+06	4.56E+03
Mo-103	7.79E+06	6.76E-12	0.00E+00	0.00E+00	0.00E+00
Tc-99	0.00E+00	3.23E-03	7.25E-02	2.76E-01	3.34E-01
Tc-99m	7.70E+06	7.70E+06	6.57E+06	1.46E+06	4.44E+03
Tc-103	7.89E+06	1.67E-15	0.00E+00	0.00E+00	0.00E+00
Tc-106	4.52E+06	2.32E-23	0.00E+00	0.00E+00	0.00E+00
Ru-103	7.94E+06	7.93E+06	7.80E+06	7.02E+06	4.68E+06
Ru-105	6.04E+06	5.17E+06	1.43E+05	2.46E-05	9.23E-43
Ru-106	3.57E+06	3.57E+06	3.56E+06	3.52E+06	3.37E+06
Rh-103m	0.00E+00	4.14E+06	7.79E+06	7.01E+06	4.67E+06
Rh-105	5.40E+06	5.40E+06	3.89E+06	2.33E+05	4.65E+00
Rh-105m	0.00E+00	1.27E+06	3.50E+04	6.04E-06	2.27E-43
Rh-106	0.00E+00	3.57E+06	3.56E+06	3.52E+06	3.37E+06
Sb-127	8.86E+06	8.79E+06	7.40E+06	2.51E+06	3.99E+04
Sb-129	3.03E+07	2.59E+07	6.91E+05	9.72E-05	1.67E-42
Sb-131	7.44E+07	1.22E+07	1.06E-11	0.00E+00	0.00E+00
Sb-132	4.33E+07	1.53E+01	0.00E+00	0.00E+00	0.00E+00
Sb-133	6.85E+07	2.04E+00	0.00E+00	0.00E+00	0.00E+00
Sb-134	1.32E+07	4.00E-92	0.00E+00	0.00E+00	0.00E+00
Te-127	8.77E+06	8.77E+06	8.14E+06	3.75E+06	1.29E+06
Te-127m	1.48E+06	1.48E+06	1.48E+06	1.45E+06	1.27E+06
Te-129	2.89E+07	2.80E+07	4.44E+06	3.23E+06	2.01E+06
Te-129m	5.90E+06	5.90E+06	5.81E+06	5.13E+06	3.19E+06
Te-131	7.56E+07	1.77E+07	2.50E+06	8.96E+04	2.59E-01
Te-131m	1.93E+07	1.89E+07	1.11E+07	3.98E+05	1.15E+00
Te-132	1.29E+08	1.28E+08	1.04E+08	2.91E+07	2.18E+05
Te-133	1.09E+08	1.05E+07	2.29E-01	2.58E-48	0.00E+00

Table 12.2-26 (4 of 14)

b. Radioactive Concentration of Post-Accident Recirculating Water (Bq/cc) (3 of 5)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Te-133m	9.10E+07	4.30E+07	1.36E+00	1.54E-47	0.00E+00
Te-134	1.92E+08	7.10E+07	8.18E-03	4.90E-65	0.00E+00
Te-135	9.49E+07	3.42E-49	0.00E+00	0.00E+00	0.00E+00
I-129	3.71E+01	3.71E+01	3.71E+01	3.71E+01	3.71E+01
I-131	7.26E+08	7.24E+08	6.67E+08	3.99E+08	5.49E+07
I-132	1.05E+09	8.10E+08	1.08E+08	3.00E+07	2.25E+05
I-133	1.54E+09	1.49E+09	6.95E+08	5.72E+06	5.87E-02
I-134	1.75E+09	8.56E+08	1.43E+01	5.12E-49	0.00E+00
I-135	1.45E+09	1.31E+09	1.17E+08	3.24E+01	2.35E-24
I-137	7.48E+08	6.62E-36	0.00E+00	0.00E+00	0.00E+00
I-138	3.76E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	0.00E+00	1.92E+04	4.30E+05	1.96E+06	1.63E+06
Xe-133	0.00E+00	8.09E+06	1.27E+08	1.20E+08	5.92E+06
Xe-133m	0.00E+00	5.72E+05	8.14E+06	3.07E+06	2.19E+03
Xe-135	0.00E+00	9.53E+07	3.09E+08	1.07E+04	6.22E-15
Xe-135m	0.00E+00	2.08E+08	2.01E+07	5.56E+00	4.04E-25
Cs-134	1.70E+08	1.70E+08	1.70E+08	1.69E+08	1.65E+08
Cs-135	3.65E+02	3.65E+02	3.65E+02	3.65E+02	3.65E+02
Cs-136	4.37E+07	4.36E+07	4.15E+07	3.02E+07	9.00E+06
Cs-137	9.41E+07	9.41E+07	9.41E+07	9.41E+07	9.39E+07
Cs-138	1.12E+09	3.08E+08	3.86E-05	6.51E-86	0.00E+00
Cs-140	9.69E+08	1.00E-08	0.00E+00	0.00E+00	0.00E+00
Ba-137m	5.99E+06	8.90E+07	8.90E+07	8.90E+07	8.89E+07
Ba-139	7.19E+07	4.36E+07	4.37E+02	2.20E-29	0.00E+00
Ba-140	7.12E+07	7.10E+07	6.74E+07	4.87E+07	1.40E+07
Ba-143	5.80E+07	1.20E-72	0.00E+00	0.00E+00	0.00E+00
Ba-144	4.84E+07	1.47E-91	0.00E+00	0.00E+00	0.00E+00
La-140	7.13E+05	1.92E+06	2.39E+07	5.16E+07	1. 61E+07
La-141	6.56E+05	5.50E+05	9.62E+03	9.57E-08	6.40E-50
La-142	6.57E+05	4.25E+05	1.88E+01	1.02E-26	0.00E+00
La-143	6.51E+05	3.34E+04	7.09E-26	0.00E+00	0.00E+00

Table 12.2-26 (5 of 14)

b. Radioactive Concentration of Post-Accident Recirculating Water (Bq/cc) (4 of 5)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
La-144	5.93E+05	4.79E-22	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.60E+06	1.60E+06	1.57E+06	1.38E+06	8.46E+05
Ce-143	1.64E+06	1.61E+06	9.91E+05	4.81E+04	4.44E-01
Ce-144	1.16E+06	1.16E+06	1.16E+06	1.14E+06	1.08E+06
Pr-143	6.40E+05	6.42E+05	6.72E+05	5.71E+05	1.78E+05
Pr-144	4.67E+05	1.10E+06	1.16E+06	1.14E+06	1.08E+06
Pr-144m	0.00E+00	1.65E+04	1.65E+04	1.63E+04	1.54E+04
Nd-147	2.57E+05	2.56E+05	2.41E+05	1.65E+05	3.87E+04
Pm-147	0.00E+00	7.74E+00	1.80E+02	1.05E+03	2.47E+03
Sm-147	0.00E+00	2.89E-15	1.63E-12	7.06E-11	8.65E-10
Tl-207	0.00E+00	1.24E-21	1.62E-21	5.24E-22	8.85E-22
Tl-208	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.29E-21
Tl-209	0.00E+00	0.00E+00	3.68E-20	3.20E-20	6.69E-21
Pb-209	0.00E+00	8.72E-21	7.73E-21	6.83E-19	4.82E-19
Pb-210	0.00E+00	0.00E+00	4.00E-16	4.63E-16	0.00E+00
Pb-211	0.00E+00	0.00E+00	1.41E-22	0.00E+00	1.61E-22
Pb-212	0.00E+00	3.13E-22	0.00E+00	0.00E+00	9.82E-21
Pb-214	0.00E+00	1.09E-15	1.09E-15	1.81E-15	4.64E-15
Bi-210	0.00E+00	7.83E-17	4.59E-16	6.85E-16	4.93E-16
Bi-211	0.00E+00	1.76E-21	1.70E-20	0.00E+00	1.26E-21
Bi-213	0.00E+00	2.75E-22	1.03E-18	4.56E-18	2.52E-18
Bi-214	0.00E+00	1.15E-15	1.21E-15	1.55E-15	4.68E-15
Po-210	0.00E+00	5.96E-16	9.25E-16	1.37E-15	1.04E-15
Po-211	0.00E+00	1.52E-24	2.40E-24	0.00E+00	1.61E-25
Po-213	0.00E+00	2.69E-22	1.01E-18	4.46E-18	2.47E-18
Po-214	0.00E+00	3.85E-16	4.37E-16	7.68E-16	3.90E-15
Po-215	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E-21
Po-216	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.26E-20
Po-218	0.00E+00	6.35E-16	1.00E-15	1.29E-15	4.64E-15
At-217	0.00E+00	2.15E-18	2.87E-18	4.35E-19	1.32E-19
Rn-220	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.27E-20

Table 12.2-26 (6 of 14)

b. Radioactive Concentration of Post-Accident Recirculating Water (Bq/cc) (5 of 5)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Rn-222	0.00E+00	6.01E-16	2.54E-16	4.20E-16	4.01E-15
Fr-221	0.00E+00	0.00E+00	1.23E-17	4.09E-18	7.08E-19
Fr-223	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.54E-22
Ra-224	0.00E+00	4.69E-22	0.00E+00	5.97E-22	1.33E-21
Ra-225	0.00E+00	3.85E-18	2.41E-18	0.00E+00	2. 46E-18
Ra-226	0.00E+00	3.50E-16	7.44E-16	1.36E-15	6.78E-15
Ra-228	0.00E+00	7.14E-22	0.00E+00	0.00E+00	6.90E-20
Ac-225	0.00E+00	1.84E-18	1.09E-20	3.43E-19	1.95E-18
Ac-227	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.79E-20
Ac-228	0.00E+00	3.55E-20	0.00E+00	2.21E-22	4.05E-20
Th-228	0.00E+00	1.03E-20	0.00E+00	4.76E-20	6.32E-20
Th-229	0.00E+00	2.41E-18	1.83E-18	6.54E-18	4.91E-18
Th-230	0.00E+00	9.13E-16	5.72E-13	2.80E-11	5.14E-10
Th-231	0.00E+00	4.34E-13	2.06E-10	4.29E-09	2.24E-08
Th-232	0.00E+00	1.12E-21	0.00E+00	1.41E-19	2.69E-18
Pa-231	0.00E+00	0.00E+00	0.00E+00	7.13E-16	1.85E-14
Pa-233	0.00E+00	1.29E-12	7.41E-10	3.56E-08	6.00E-07
U-233	0.00E+00	0.00E+00	6.16E-18	1.22E-15	7.35E-14
U-234	0.00E+00	1.93E-06	4.63E-05	3.24E-04	1.39E-03
U-235	0.00E+00	3.22E-11	7.76E-10	5.48E-09	2.36E-08
U-236	0.00E+00	1.82E-09	4.37E-08	3.06E-07	1.31E-06
U-237	0.00E+00	1.59E-02	3.63E-01	1.91E+00	3.54E+00
Np-237	0.00E+00	2.41E-09	5.84E-08	4.26E-07	2.07E-06
Np-239	2.11E+07	2.08E+07	1.57E+07	2.69E+06	3.09E+03
Pu-238	5.97E+03	5.97E+03	5.97E+03	5.97E+03	5.99E+03
Pu-239	2.87E+02	2.87E+02	2.88E+02	2.92E+02	2.93E+02
Pu-240	5.39E+02	5.39E+02	5.39E+02	5.39E+02	5.39E+02
Pu-241	1.52E+05	1.52E+05	1.52E+05	1.52E+05	1.51E+05
Am-241	6.53E+01	6.53E+01	6.60E+01	7.00E+01	8.53E+01
Cm-242	3.27E+04	3.27E+04	3.26E+04	3.17E+04	2.88E+04
Cm-244	4.66E+03	4.66E+03	4.66E+03	4.66E+03	4.65E+03

Table 12.2-26 (7 of 14)

c. Radioactive Concentration of Post-Accident Airborne in Containment (Bq/cc) (1 of 6)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Se-84	7.56E+05	2.54E+00	0.00E+00	0.00E+00	0.00E+00
Se-85	3.77E+05	6.15E-23	0.00E+00	0.00E+00	0.00E+00
Se-87	5.60E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	6.19E+06	1.67E+06	1.45E-07	2.35E-89	0.00E+00
Br-85	7.14E+06	3.57E+00	0.00E+00	0.00E+00	0.00E+00
Br-87	1.17E+07	4.43E-13	0.00E+00	0.00E+00	0.00E+00
Br-88	1.18E+07	3.86E-60	0.00E+00	0.00E+00	0.00E+00
Br-89	7.95E+06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	6.62E+05	6.62E+05	6.62E+05	6.61E+05	6.59E+05
Kr-85m	1.78E+07	1.53E+07	4.36E+05	9.20E-05	7.46E-42
Kr-87	3.65E+07	2.12E+07	7.60E+01	6.21E-33	0.00E+00
Kr-88	5.16E+07	4.04E+07	1.47E+05	8.04E-11	2.48E-69
Kr-89	6.63E+07	1.28E+02	0.00E+00	0.00E+00	0.00E+00
Kr-90	7.21E+07	2.13E-26	0.00E+00	0.00E+00	0.00E+00
Kr-91	4.97E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-86	4.37E+04	4.36E+04	4.21E+04	3.37E+04	1.43E+04
Rb-87	0.00E+00	4.70E-08	1.12E-07	1.12E-07	1.12E-07
Rb-88	1.58E+07	4.11E+07	1.65E+05	8.98E-11	2.77E-69
Rb-89	2.10E+07	2.57E+06	3.20E-21	0.00E+00	0.00E+00
Rb-90	1.98E+07	9.08E+00	1.12E-96	0.00E+00	0.00E+00
Rb-90m	0.00E+00	7.75E+01	1.90E-95	0.00E+00	0.00E+00
Rb-91	2.49E+07	5.97E-12	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.29E+06	1.30E+06	1.28E+06	1.18E+06	8.60E+05
Sr-90	1.16E+05	1.16E+05	1.16E+05	1.16E+05	1.16E+05
Sr-91	1.73E+06	1.61E+06	3.00E+05	8.21E+00	2.65E-17
Sr-92	1.75E+06	1.36E+06	3.78E+03	3.81E-13	1.84E-74
Sr-95	1.63E+06	3.40E-36	0.00E+00	0.00E+00	0.00E+00
Y-90	1.22E+03	2.45E+03	2.75E+04	9.73E+04	1.16E+05
Y-91	1.58E+04	1.63E+04	2.51E+04	2.54E+04	1.93E+04
Y-91m	1.00E+04	5.44E+05	1.89E+05	5.17E+00	1.67E-17
Y-92	1.77E+04	2.88E+05	3.98E+04	2.97E-08	3.40E-55

Table 12.2-26 (8 of 14)

c. Radioactive Concentration of Post-Accident Airborne in Containment (Bq/cc) (2 of 6)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Y-93	1.27E+04	1.19E+04	2.45E+03	1.25E-01	4.41E-18
Y-95	1.95E+04	3.71E+02	1.01E-37	0.00E+00	0.00E+00
Zr-93	0.00E+00	6.34E-07	7.72E-06	9.56E-06	9.56E-06
Zr-95	1.85E+04	1.85E+04	1.83E+04	1.71E+04	1.34E+04
Zr-97	1.75E+04	1.68E+04	6.54E+03	1.78E+01	2.62E-09
Nb-93m	0.00E+00	1.74E-12	6.34E-10	7.94E-09	3.65E-08
Nb-95	1.84E+04	1.84E+04	1.84E+04	1.83E+04	1.71E+04
Nb-95m	0.00E+00	1.18E+00	2.57E+01	1.05E+02	1.13E+02
Nb-97	0.00E+00	7.38E+03	7.05E+03	1.92E+01	2.82E-09
Nb-97m	0.00E+00	1.59E+04	6.20E+03	1.69E+01	2.48E-09
Nb-99	1.14E+04	1.88E-72	0.00E+00	0.00E+00	0.00E+00
Mo-99	2.41E+05	2.38E+05	1.87E+05	4.13E+04	1.26E+02
Mo-103	2.15E+05	1.86E-13	0.00E+00	0.00E+00	0.00E+00
Tc-99	0.00E+00	8.89E-05	2.00E-03	7.61E-03	9.20E-03
Tc-99m	2.12E+05	2.12E+05	1.81E+05	4.03E+04	1.22E+02
Tc-103	2.17E+05	4.60E-17	0.00E+00	0.00E+00	0.00E+00
Tc-106	1.24E+05	6.37E-25	0.00E+00	0.00E+00	0.00E+00
Ru-103	2.19E+05	2.19E+05	2.15E+05	1.94E+05	1.29E+05
Ru-105	1.66E+05	1.42E+05	3.92E+03	6.76E-07	2.54E-44
Ru-106	9.82E+04	9.82E+04	9.80E+04	9.69E+04	9.28E+04
Rh-103m	0.00E+00	1.14E+05	2.15E+05	1.93E+05	1.29E+05
Rh-105	1.49E+05	1.49E+05	1.07E+05	6.42E+03	1.28E-01
Rh-105m	0.00E+00	3.49E+04	9.62E+02	1.66E-07	6.23E-45
Rh-106	0.00E+00	9.82E+04	9.80E+04	9.69E+04	9.28E+04
Sb-127	2.44E+05	2.42E+05	2.04E+05	6.92E+04	1.10E+03
Sb-129	8.35E+05	7.13E+05	1.90E+04	2.68E-06	4.59E-44
Sb-131	2.05E+06	3.36E+05	2.92E-13	0.00E+00	0.00E+00
Sb-132	1.19E+06	4.22E-01	0.00E+00	0.00E+00	0.00E+00
Sb-133	1.89E+06	5.63E-02	0.00E+00	0.00E+00	0.00E+00
Sb-134	3.64E+05	1.10E-93	0.00E+00	0.00E+00	0.00E+00
Te-127	2.42E+05	2.42E+05	2.24E+05	1.03E+05	3.54E+04

Table 12.2-26 (9 of 14)

c. Radioactive Concentration of Post-Accident Airborne in Containment (Bq/cc) (3 of 6)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Te-127m	4.07E+04	4.07E+04	4.07E+04	3.99E+04	3.49E+04
Te-129	7.96E+05	7.72E+05	1.23E+05	8.93E+04	5.56E+04
Te-129m	1.63E+05	1.63E+05	1.60E+05	1.42E+05	8.82E+04
Te-131	2.08E+06	4.88E+05	6.88E+04	2.47E+03	7.14E-03
Te-131m	5.32E+05	5.20E+05	3.06E+05	1.10E+04	3.17E-02
Te-132	3.55E+06	3.52E+06	2.87E+06	8.01E+05	6.01E+03
Te-133	3.01E+06	2.90E+05	6.30E-03	7.11E-50	0.00E+00
Te-133m	2.51E+06	1.18E+06	3.76E-02	4.24E-49	0.00E+00
Te-134	5.30E+06	1.96E+06	2.26E-04	1.35E-66	0.00E+00
Te-135	2.61E+06	9.41E-51	0.00E+00	0.00E+00	0.00E+00
I-129	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00
I-131	2.00E+07	1.99E+07	1.84E+07	1.10E+07	1.51E+06
I-132	2.90E+07	2.24E+07	2.97E+06	8.25E+05	6.19E+03
I-133	4.23E+07	4.10E+07	1.91E+07	1.57E+05	1.61E-03
I-134	4.83+07	2.36E+07	3.94E-01	1.41E-50	0.00E+00
I-135	3.99E+07	3.59E+07	3.22E+06	8.91E-01	6.47E-26
I-137	2.06E+07	1.82E-37	0.00E+00	0.00E+00	0.00E+00
I-138	1.03E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	7.36E+05	7.35E+05	7.06E+05	5.42E+05	1.72E+05
Xe-133	1.05E+08	1.05E+08	9.58E+07	4.56E+07	2.20E+06
Xe-133m	3.23E+06	3.20E+06	2.58E+06	4.37E+05	3.03E+02
Xe-135	3.16E+07	3.25E+07	1.37E+07	3.86E+02	2.23E-16
Xe-135m	2.19E+07	7.17E+06	5.53E+05	1.53E-01	1.11E-26
Xe-137	9.66E+07	1.86E+03	0.00E+00	0.00E+00	0.00E+00
Xe-138	9.77E+07	5.15E+06	2.05E-23	0.00E+00	0.00E+00
Xe-140	5.76E+07	1.19E-72	0.00E+00	0.00E+00	0.00E+00
Cs-134	4.67E+06	4.67E+06	4.67E+06	4.64E+06	4.54E+06
Cs-135	1.01E+01	1.01E+01	1.01E+01	1.01E+01	1.01E+01
Cs-136	1.20E+06	1.20E+06	1.14E+06	8.30E+05	2.47E+05
Cs-137	2.59E+06	2.59E+06	2.59E+06	2.59E+06	2.59E+06
Cs-138	3.09E+07	2.55E+07	3.70E-06	6.24E-87	0.00E+00

Table 12.2-26 (10 of 14)

c. Radioactive Concentration of Post-Accident Airborne in Containment (Bq/cc) (4 of 6)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Cs-140	2.67E+07	2.76E-10	0.00E+00	0.00E+00	0.00E+00
Ba-137m	1.65E+05	2.45E+06	2.45E+06	2.45E+06	2.45E+06
Ba-139	1.98E+06	1.20E+06	1.20E+01	6.05E-31	0.00E+00
Ba-140	1.96E+06	1.96E+06	1.86E+06	1.34E+06	3.86E+05
Ba-143	1.60E+06	3.31E-74	0.00E+00	0.00E+00	0.00E+00
Ba-144	1.33E+06	4.03E-93	0.00E+00	0.00E+00	0.00E+00
La-140	1.96E+04	5.27E+04	6.58E+05	1.42E+06	4.44E+05
La-141	1.81E+04	1.52E+04	2.65E+02	2.64E-09	1.77E-51
La-142	1.81E+04	1.17E+04	5.17E-01	2.82E-28	0.00E+00
La-143	1.79E+04	9.18E+02	1.95E-27	0.00E+00	0.00E+00
La-144	1.63E+04	1.32E-23	0.00E+00	0.00E+00	0.00E+00
Ce-141	4.41E+04	4.41E+04	4.33E+04	3.81E+04	2.33E+04
Ce-143	4.50E+04	4.41E+04	2.72E+04	1.32E+03	1.22E-02
Ce-144	3.20E+04	3.20E+04	3.19E+04	3.15E+04	2.97E+04
Pr-143	1.76E+04	1.77E+04	1.85E+04	1.57E+04	4.89E+03
Pr-144	1.28E+04	3.02E+04	3.19E+04	3.15E+04	2.97E+04
Pr-144m	0.00E+00	4.56E+02	4.56E+02	4.50E+02	4.25E+02
Nd-147	7.07E+03	7.05E+03	6.64E+03	4.54E+03	1.06E+03
Pm-147	0.00E+00	2.13E-01	4.95E+00	2.89E+01	6.79E+01
Sm-147	0.00E+00	7.94E-17	4.48E-14	1.94E-12	2.38E-11
Tl-207	0.00E+00	3.42E-23	4.46E-23	1.44E-23	2.44E-23
Tl-208	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.29E-23
Tl-209	0.00E+00	0.00E+00	1.01E-21	8.81E-22	1.85E-22
Pb-209	0.00E+00	2.40E-22	2.13E-22	1.88E-20	1.33E-20
Pb-210	0.00E+00	0.00E+00	1.10E-17	1.27E-17	0.00E+00
Pb-211	0.00E+00	0.00E+00	3.88E-24	0.00E+00	4.44E-24
Pb-212	0.00E+00	8.59E-24	0.00E+00	0.00E+00	2.70E-22
Pb-214	0.00E+00	2.99E-17	3.00E-17	4.98E-17	1.28E-16
Bi-210	0.00E+00	2.15E-18	1.26E-17	1.88E-17	1.35E-17
Bi-211	0.00E+00	4.85E-23	4.67E-22	0.00E+00	3.47E-23
Bi-213	0.00E+00	7.55E-24	2.85E-20	1.26E-19	6.95E-20

Table 12.2-26 (11 of 14)

c. Radioactive Concentration of Post-Accident Airborne in Containment (Bq/cc) (5 of 6)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
Bi-214	0.00E+00	3.15E-17	3.34E-17	4.25E-17	1.29E-16
Po-210	0.00E+00	1.64E-17	2.54E-17	3.76E-17	2.86E-17
Po-211	0.00E+00	4.18E-26	6.62E-26	0.00E+00	4.43E-27
Po-213	0.00E+00	7.39E-24	2.79E-20	1.23E-19	6.80E-20
Po-214	0.00E+00	1.06E-17	1.20E-17	2.11E-17	1.07E-16
Po-215	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.43E-23
Po-216	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.46E-22
Po-218	0.00E+00	1.74E-17	2.75E-17	3.53E-17	1.28E-16
At-217	0.00E+00	5.92E-20	7.92E-20	1.20E-20	3.64E-21
Rn-220	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.48E-22
Rn-222	0.00E+00	1.65E-17	6.97E-18	1.15E-17	1.10E-16
Fr-221	0.00E+00	0.00E+00	3.38E-19	1.13E-19	1.95E-20
Fr-223	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.75E-24
Ra-224	0.00E+00	1.29E-23	0.00E+00	1.64E-23	3.66E-23
Ra-225	0.00E+00	1.06E-19	6.65E-20	0.00E+00	6.79E-20
Ra-226	0.00E+00	9.61E-18	2.04E-17	3.72E-17	1.86E-16
Ra-228	0.00E+00	1.96E-23	0.00E+00	0.00E+00	1.89E-21
Ac-225	0.00E+00	5.06E-20	2.99E-22	9.45E-21	5.38E-20
Ac-227	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.04E-21
Ac-228	0.00E+00	9.75E-22	0.00E+00	6.08E-24	1.11E-21
Th-228	0.00E+00	2.82E-22	0.00E+00	1.31E-21	1.74E-21
Th-229	0.00E+00	6.64E-20	5.03E-20	1.80E-19	1.35E-19
Th-230	0.00E+00	2.51E-17	1.57E-14	7.69E-13	1.41E-11
Th-231	0.00E+00	1.19E-14	5.66E-12	1.18E-10	6.17E-10
Th-232	0.00E+00	3.08E-23	0.00E+00	3.88E-21	7.37E-20
Pa-231	0.00E+00	0.00E+00	0.00E+00	1.96E-17	5.09E-16
Pa-233	0.00E+00	3.56E-14	2.04E-11	9.80E-10	1.65E-08
U-233	0.00E+00	0.00E+00	1.70E-19	3.37E-17	2.03E-15
U-234	0.00E+00	5.30E-08	1.27E-06	8.91E-06	3.82E-05
U-235	0.00E+00	8.88E-13	2.14E-11	1.51E-10	6.50E-10
U-236	0.00E+00	5.00E-11	1.20E-09	8.40E-09	3.60E-08

Table 12.2-26 (12 of 14)

c. Radioactive Concentration of Post-Accident Airborne in Containment (Bq/cc) (6 of 6)

Nuclide	Elasped Time after the onset of the Accident				
	0 hr	1 hr	1 day	1 week	1 month
U-237	0.00E+00	4.37E-04	9.99E-03	5.25E-02	9.74E-02
Np-237	0.00E+00	6.65E-11	1.61E-09	1.18E-08	5.71E-08
Np-239	5.81E+05	5.74E+05	4.33E+05	7.40E+04	8.50E+01
Pu-238	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
Pu-239	7.90E+00	7.90E+00	7.94E+00	8.04E+00	8.06E+00
Pu-240	1.48E+01	1.48E+01	1.48E+01	1.48E+01	1.48E+01
Pu-241	4.18E+03	4.18E+03	4.18E+03	4.18E+03	4.16E+03
Am-241	1.80E+00	1.80E+00	1.82E+00	1.93E+00	2.35E+00
Cm-242	9.01E+02	9.01E+02	8.97E+02	8.75E+02	7.93E+02
Cm-244	1.28E+02	1.28E+02	1.28E+02	1.28E+02	1.28E+02

Table 12.2-26 (13 of 14)

d. MCR Emergency Makeup ACU Filter Inventories (Bq) (1 of 2)

Nuclide	Elasped Time After the Onset of the Accident				
	1 hour	1 day	4 days	1 week	1 month
I-131	2.77E+09	1.75E+10	2.25E+10	2.48E+10	3.12E+10
I-132	3.70E+09	2.33E+10	2.99E+10	3.31E+10	4.16E+10
I-133	5.30E+09	3.34E+10	4.29E+10	4.75E+10	5.96E+10
I-134	5.19E+08	3.28E+09	4.22E+09	4.67E+09	5.87E+09
I-135	3.96E+09	2.49E+10	3.20E+10	3.55E+10	4.45E+10
Co-58	1.67E+05	9.05E+05	9.54E+05	9.72E+05	9.90E+05
Co-60	1.28E+05	6.93E+05	7.30E+05	7.44E+05	7.58E+05
Rb-86	5.68E+06	3.08E+07	3.24E+07	3.31E+07	3.37E+07
Sr-89	1.99E+08	1.08E+09	1.14E+09	1.16E+09	1.18E+09
Sr-90	1.79E+07	9.69E+07	1.02E+08	1.04E+08	1.06E+08
Sr-91	2.11E+08	1.14E+09	1.20E+09	1.23E+09	1.25E+09
Sr-92	1.17E+08	6.32E+08	6.67E+08	6.79E+08	6.92E+08
Y-90	2.46E+05	1.33E+06	1.40E+06	1.43E+06	1.46E+06
Y-91	2.46E+06	1.33E+07	1.40E+07	1.43E+07	1.46E+07
Y-92	9.73E+06	5.27E+07	5.56E+07	5.67E+07	5.77E+07
Y-93	1.56E+06	8.46E+06	8.91E+06	9.09E+06	9.25E+06
Zr-95	2.85E+06	1.54E+07	1.63E+07	1.66E+07	1.69E+07
Zr-97	2.37E+06	1.28E+07	1.35E+07	1.38E+07	1.40E+07
Nb-95	2.84E+06	1.54E+07	1.62E+07	1.66E+07	1.69E+07
Mo-99	3.59E+07	1.95E+08	2.05E+08	2.09E+08	2.13E+08
Tc-99m	3.26E+07	1.77E+08	1.86E+08	1.90E+08	1.93E+08
Ru-103	3.38E+07	1.83E+08	1.93E+08	1.97E+08	2.00E+08
Ru-105	1.54E+07	8.33E+07	8.78E+07	8.95E+07	9.11E+07
Ru-106	1.51E+07	8.21E+07	8.65E+07	8.82E+07	8.97E+07
Rh-105	2.28E+07	1.24E+08	1.30E+08	1.33E+08	1.35E+08
Sb-127	3.67E+07	1.99E+08	2.10E+08	2.14E+08	2.18E+08
Sb-129	7.61E+07	4.12E+08	4.34E+08	4.43E+08	4.51E+08
Te-127	3.72E+07	2.02E+08	2.13E+08	2.17E+08	2.21E+08
Te-127m	6.30E+06	3.41E+07	3.60E+07	3.67E+07	3.73E+07
Te-129	9.43E+07	5.11E+08	5.39E+08	5.49E+08	5.59E+08
Te-129m	2.52E+07	1.36E+08	1.44E+08	1.47E+08	1.49E+08

Table 12.2-26 (14 of 14)

d. MCR Emergency Makeup ACU Filter Inventories (Bq) (2 of 2)

Nuclide	Elasped Time After the Onset of the Accident				
	1 hour	1 day	4 days	1 week	1 month
Te-131m	7.62E+07	4.13E+08	4.35E+08	4.44E+08	4.52E+08
Te-132	5.32E+08	2.88E+09	3.04E+09	3.10E+09	3.16E+09
Cs-134	6.11E+08	3.31E+09	3.49E+09	3.56E+09	3.62E+09
Cs-136	1.56E+08	8.45E+08	8.90E+08	9.07E+08	9.24E+08
Cs-137	3.39E+08	1.83E+09	1.93E+09	1.97E+09	2.01E+09
Ba-139	5.88E+07	3.19E+08	3.36E+08	3.42E+08	3.49E+08
Ba-140	3.00E+08	1.63E+09	1.72E+09	1.75E+09	1.78E+09
La-140	4.57E+06	2.47E+07	2.61E+07	2.66E+07	2.71E+07
La-141	1.56E+06	8.46E+06	8.92E+06	9.09E+06	9.25E+06
La-142	6.40E+05	3.47E+06	3.65E+06	3.72E+06	3.79E+06
Ce-141	6.82E+06	3.69E+07	3.89E+07	3.97E+07	4.04E+07
Ce-143	6.50E+06	3.52E+07	3.71E+07	3.78E+07	3.85E+07
Ce-144	4.94E+06	2.68E+07	2.82E+07	2.88E+07	2.93E+07
Pr-143	2.73E+06	1.48E+07	1.56E+07	1.59E+07	1.61E+07
Nd-147	1.08E+06	5.86E+06	6.18E+06	6.30E+06	6.41E+06
Np-239	8.63E+07	4.67E+08	4.93E+08	5.02E+08	5.11E+08
Pu-238	2.54E+04	1.38E+05	1.45E+05	1.48E+05	1.51E+05
Pu-239	1.22E+03	6.62E+03	6.98E+03	7.11E+03	7.24E+03
Pu-240	2.30E+03	1.24E+04	1.31E+04	1.34E+04	1.36E+04
Pu-241	6.45E+05	3.50E+06	3.69E+06	3.76E+06	3.82E+06
Am-241	2.78E+02	1.50E+03	1.59E+03	1.62E+03	1.65E+03
Cm-242	1.39E+05	7.53E+05	7.94E+05	8.09E+05	8.23E+05
Cm-244	1.98E+04	1.07E+05	1.13E+05	1.15E+05	1.18E+05

Table 12.2-27 (1 of 5)

Radioactive Source Dimensions and Parameters Used in Shielding Analysis

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm^3)	Material	Thickness (cm)
Reactor Containment Building	Pressurizer	Cylinder	244.48	-	700.06 747.71	Water: 48% Vapor: 52%	0.59 0.001293	Steel	12.38
	Reactor coolant pump	Cylinder	185.00	-	126.74	Water: 100 %	0.75	Steel	14.00
	Reactor drain tank	Cylinder	Liquid: 162.90 Vapor: 99.70	-	528.57	Water: 73 % Vapor: 27 %	1.00 0.001293	Not considered	
	Regenerative HX	Cylinder	24.69	-	400.69	Water: 85 % Steel: 15 %	0.85 1.18	Steel	2.22
	Letdown HX	Cylinder	45.72	-	341.36	Water: 88 % Steel: 12 %	0.88 0.95	Steel	2.54
	Steam generator	Annular cylinder	OD: 497.80 ID: 415.80	-	969.57	Water: 100 %	0.70	Steel	12.86
		Semisphere	472.60						
Auxiliary Building	SC HX	Cylinder	137.16	-	803.15	Water: 94 % Steel: 6 %	0.94 0.54	Steel	1.27
	SC miniflow HX(1)	Cylinder	33.66	-	173.43	Water: 93 % Steel: 7 %	0.93 0.59	Steel	0.95
	SC pump	Cylinder	38.1	-	609.6	Water: 100%	1	Steel	1.27
	Charging pump miniflow HX	Cylinder	38.10	-	298.70	Water: 94 % Steel: 6 %	0.94 0.50	Steel	1.27
	CS HX(1)	Cylinder	129.54	-	701.04	Water: 94 % Steel: 6 %	0.94 0.49	Steel	1.59

(1) For post-accident, the volume to estimate source term in the heat exchanger is limited to the volume of tube side. (SC mini flow HX : 4.33E+04 cm^3, CS HX : 3.27E+06 cm^3)

Table 12.2-27 (2 of 5)

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm^3)	Material	Thickness (cm)
Auxiliary Building	CS miniflow HX(1)	Cylinder	31.75	-	186.06	Water: 94 % Steel: 6 %	0.94 0.45	Steel	0.95
	CS pump	Cylinder	38.1	-	609.6	Water: 100 %	1	Steel	1.27
	SI pump	Cylinder	25.5	-	609.6	Water: 100 %	1	Steel	0.93
	Equipment drain tank	Cylinder (Horizontal)	Liquid: 193.59 Vapor: 193.59	-	610.87 (Length of tank)	Water: 50 % Vapor: 50 %	1.00 0.001293	Not considered	
	Boric acid concentrator	Cylinder	Liquid: 193.53 Vapor: 206.58	-	180.52	Water: 47 % Vapor: 53 %	1.00 0.001293	Not considered	
	SFP cleanup demin.	Cylinder	145.70	-	144.17	Water: 100 %	1.00	Not considered	
	Boric acid condensate IX	Cylinder	74.60	-	206.17	Water: 100 %	1.00	Not considered	
	Deborating IX	Cylinder	105.08	-	104.49	Water: 100 %	1.00	Not considered	
	Pre-holdup IX	Cylinder	105.08	-	104. 49	Water: 100 %	1. 00	Not considered	
	Purification IX	Cylinder	105.08	-	104.49	Water: 100 %	1.00	Not considered	
	SFP cooling HX	Rectangular	31.15	134.15	198.28	Water: 67 % Steel: 33 %	0.67 2.63	Not considered	
	Volume control tank	Cylinder	241.44		Liquid: 223.23 Vapor: 330.72	Water: 40 % Vapor: 60 %	1.00 0.001293	Not considered	

(1) For post-accident, the volume to estimate source term in the CS mini flow heat exchanger is limited to the volume of tube side, 3.68E+04 cm^3.

Table 12.2-27 (3 of 5)

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm^3)	Material	Thickness (cm)
Auxiliary Building	SGBD flash tank	Cylinder	304.8	-	455.96	Water: 100 %	1.00	Not considered	
	SGBD HX	Cylinder	84.86	-	487.68	Water: 86 % Steel: 14 %	0.90 1.12	Steel	SGBD HX
	Spent fuel pool	Rectangular	869.00	1,113.50	381.00	Water: 70% UO_2: 22% Zircaloy: 8%	0.70 1.98 0.56	Not considered	
	Cask loading pit	Rectangular parallelepiped	20.23	20.23	381.00	Water: 58% UO_2: 30% Zircaloy: 12%	0.58 2.76 0.79	Not considered	
	GRS header drain tank	Cylinder	45.72	-	172.48	Vapor: 100%	0.001293	Not considered	
	IRWST	Annular Cylinder	O.D: 4378.96 I.D: 3230.88	-	358.04 122.24	Water: 77% Vapor: 23%	1.00 0.001293	Not considered	
	Seal injection filter	Cylinder	10.80	-	54.00	Water: 100%	1.00	Not considered	
	Reactor drain filter	Cylinder	19.35	-	50.50	Water: 100%	1.00	Not considered	
	Boric acid filter	Cylinder	19.35	-	50.50	Water: 100%	1.00	Not considered	
	Purification filter	Cylinder	19.35	-	50.50	Water: 100%	1.00	Not considered	
	Reactor makeup water filter	Cylinder	19.35	-	50.50	Water: 100%	1.00	Not considered	
	Steam generator blowdown mixed-bed	Cylinder	228.60	-	152.40	Water: 100%	1.00	Not considered	

Table 12.2-27 (4 of 5)

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm^3)	Material	Thickness (cm)
Auxiliary Building	Blowdown pre-filter	Cylinder	25.93	-	51.12	Water: 100%	1.00	Not considered	
	Blowdown post-filter	Cylinder	25.93	-	51.12	Water: 100%	1.00	Not considered	
Compound Building	Chemical waste tank	Cylinder	304.80	-	466.91	Water: 100%	1.00	Not considered	
	Floor drain tank	Cylinder	358.14	-	676.38	Water: 100%	1.00	Not considered	
	Low-activity spent resin tank	Cylinder	274.32	-	383.33	Water: 100%	1.00	Not considered	
	Spent resin long-term storage tank	Cylinder	487.68	-	482.92	Water: 100%	1.00	Not considered	
	Waste drum storage	Rectangular parallelepiped	601.98	782.57	262.89	Carbon: 100%	2.62	Not considered	
	Header Drain Tank	Cylinder	76.20	-	121.92	Vapor: 100%	0.001293	steel	0.6
	Waste Gas Dryer	Cylinder	30.318	-	91.72	Vapor: 82.73% Steel: 17.27%	0.001009 1.357	Steel	1.03
	Guard Bed	Cylinder	50.80	-	139.71	Carbon: 100%	0.41	Not considered	
	Delay Bed	Cylinder	180.08	-	456.05	Carbon: 100%	0.41	Not considered	

Table 12.2-27 (5 of 5)

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm^3)	Material	Thickness (cm)
Compound Building	HEPA filter	Cylinder	45.72	-	50.80	Vapor: 100%	0.001293	Not considered	
	Equipment waste tank	Cylinder	358.14	-	676.38	Water: 100%	1.00	Not considered	
	Monitor tank	Cylinder	487.68	-	547.16	Water: 100%	1.00	Not considered	
	Reverse osmosis	Cylinder	66.33	-	101.60	Water: 100%	1.00	Not considered	
	LRS IX (Cation Bed)	Cylinder	120.17	-	124.83	Water: 100%	1.00	Not considered	
	LRS IX (Mixed1 Bed)	Cylinder	120.17	-	124.83	Water: 100%	1.00	Not considered	
	LRS IX (Mixed2 Bed)	Cylinder	120.17	-	124.83	Water: 100%	1.00	Not considered	
Turbine Building	CPS cation bed	Sphere	314.96	-	-	Water: 100%	1.00	Steel	3.40
	CPS mixed bed	Sphere	314.96	-	-	Water: 100%	1.00	Steel	3.40
Yard	Boric Acid Storage Tank	Cylinder	1342.12	-	Liquid : 334.96 Vapor : 334.96	Water: 50% Vapor: 50%	1.00 0.001293	Concrete	40.64
	Holdup Tank	Cylinder	1706.88	-	Liquid : 86.85 Vapor : 607.96	Water: 12.5% Vapor: 87.5%	1.00 0.001293	Steel Concrete	0.635 37.465
	Reactor Makeup Water Tank	Cylinder	1676.40	-	Liquid : 521.62 Vapor : 155.81	Water: 77% Vapor: 23%	1.00 0.001293	Not considered	

Table 12.2-28 (1 of 8)

Assumptions and Parameters Used in Airborne Source Term Calculations

Reactor Containment Building

Assumptions/Parameters	Values
Fuel failure	0.25 % fuel defect
RCS leak rate	1.89 L/min (0.5 gpm) (design basis leak rate)
RCS letdown flow	302.8 L/min (80 gpm)
Iodine spike	Not considered
Low-volume purge flow rate	2.549E + 03 m^3/hr (1,500 cfm) (Assumed that CLVPS operates for 100 hours before shutdown)
High-volume purge flow rate	9.14E + 04 m^3/hr (54,000 cfm) (Assumes that CHVPS operates at cold shutdown operation mode for 29 hours)
Filter efficiency for low-volume purge filters	Particulate: 99 % Halogen: 99 %
Gas stripping	No gas stripping
Containment free volume	8.858E + 04 m^3 (3.128E + 06 ft^3)

Table 12.2-28 (2 of 8)

Fuel Handling Area in Auxiliary Building

Assumptions/Parameters	Values
Surface Area of SFP	1.3852E+02 m^2
SFP Water Volume	1.72E+03 m^3 (=649,000 gal)
SFP Temperature	49 ºC (120 °F) (during power operation) 60 ºC (140 °F) (during refueling and shutdown)
Fuel Handling Area HVAC flow (1)	28,450 cfm
Air Pressure Above SFP	1.033 kg/cm^2
Air Temperature Above SFP	25 ºC (77 °F)
Relative Humidity Above SFP	70 %
Wind Speed on the Water Surface	0.101 m/s (20 ft/min)
RCS Water Mass	3.0E+05 kg
Tritium Production Rate in Reactor Coolant	9.92E+13 Bq/1.5 yr (=6.613E+13 Bq/yr)
RCS Water Density	700 kg/m^3
Evaporation Rate of SFP	1.697E-02 m^3/hr

(1) The SFP area includes the spent fuel pool, cask loading pits, fuel loading & unloading, refueling canal, and new fuel container laydown & inspection area

Table 12.2-28 (3 of 8)

Auxiliary and Compound Building

Cubicle	Volume (m^3)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms (1), (2)	Minimum Required Ventilation Flow (m^3/hr)
CS Pump and Miniflow HX Rm (050-A01C, D)	461	Pump seal (1) Flange (4) Valve 1″ (4) Valve 4″ (2) Valve 14″ (1)	8.33E-04 2.00E-03 6.66E-04 1.33E-03 2.33E-03	IRWST	170
SI Pump Rm (050-A02C, D)	610	Pump seal (1) Flange (3) Valve 4″ (4) Valve 10″ (1)	1.67E-02 1.50E-03 2.66E-03 1.67E-03	IRWST	340
Floor Drain Sump Pump Rm (055-A34A, B, C, D)	94.9	Evaporation from sump	6.74E-02	0.1 PCA	680
Pipe Chase and Valve Rm (055-A14C)	996	Valve 18″ (2) Valve 20″ (1)	6.00E-03 3.33E-03	IRWST	170
Shutdown Cooling HX Rm (055-A30A, B)	925	Valve 1.5″ (1) Valve 10″ (2)	2.50E-04 3.33E-03	IRWST	170

Table 12.2-28 (4 of 8)

Cubicle	Volume (m^3)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms (1), (2)	Minimum Required Ventilation Flow (m^3/hr)
Charging Pump Rm (055-A55B)	833	Pump Seal (1) Flange (6) Valve 1″ (1) Valve 3″ (4) Valve 4″ (2)	8.33E-04 3.00E-03 1.67E-04 2.00E-03 1.33E-03	VCT – Halogens RCS – NG	850
Charging Pump Miniflow HX Rm (055-A43A)	187	Valve 2.5″ (1)	4.16E-04	VCT – Halogens RCS – NG	170
Equipment Drain Tank Rm (055-A51B)	501	Flange (2) Valve 1″ (1) Valve 6″ (1)	9.99E-04 1.67E-04 9.99E-04	EDT	170
Reactor Drain Pump Rm (055-A52B, A53B)	200	Pump Seal (1) Flange (4)	8.33E-04 2.00E-03	RDT	850
Gas Stripper Rm (068-A06A)	697	Flange (1)	5.00E-04	Gas Stripper	1,444
Filter and Demi. Valve Area (068-A10A)	174	Valve 3″ (1) Valve 3″ (27)	5.00E-04 1.35E-02	RDT 1.0 PCA	1,529

Table 12.2-28 (5 of 8)

Cubicle	Volume (m^3)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms (1), (2)	Minimum Required Ventilation Flow (m^3/hr)
SFP Cleanup Pump Rm (078-A38A)	224	Flange (6) Valve 4″ (3) Valve 6″ (4)	5.00E-03 2.00E-03 4.01E-03	SFP	510
Reactor Makeup Water Pump Rm (078-A49B)	168	Flange (8) Pump Seal (2)	4.00E-03 1.67E-03	RMWT	340
Holdup Pump Rm (078-A50B)	311	Flange (8) Pump Seal (2)	4.00E-03 1.67E-03	Holdup Tank	340
Valve Rm (120-A23A)	77	Valve 3″ (1) Valve 3″ (1) Valve 3″ (1)	5.00E-04 5.00E-04 5.00E-04	1.0 PCA RMWT BAST	510
Valve Rm (063-P07)	275	Flange (4) Valve 2″ (3) Valve 2.5″ (2)	2.00E-03 9.99E-04 8.33E-04	1.0 PCA Without Noble Gas	340
		Flange (2) Valve 2″ (6)	9.99E-04 2.00E-03	1.0 PCA	
Equipment Waste Pump Rm (063-P21, P22)	138	Pump Seal (1) Flange (6) Valve 2″ (2) Valve 3″ (6)	8.33E-04 3.00E-03 6.66E-04 3.00E-03	0.32 PCA	340
Equipment Waste Tank Rm (063-R23, P24)	221	Flange (2)	9.99E-04	Equipment Waste Tank	340

Table 12.2-28 (6 of 8)

Cubicle	Volume (m^3)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms (1), (2)	Minimum Required Ventilation Flow (m^3/hr)
Floor Drain Pump Rm (063-P25)	413	Pump seal (1) Flange (2) Valve 2″ (4) Valve 3″ (12)	8.33E-04 9.99E-04 1.33E-03 6.00E-03	0.44 PCA	340
Normal Sump Pump Rm (063-P26)	68.5	Flange (4) Valve 3″ (2) Evaporation from sump	2.00E-03 9.99E-04 4.92E-02	0.01 PCA	510
Chemical Waste Pump Rm (063-P27)	215	Pump seal (2) Flange (10) Valve 3″ (11) Valve 2″ (2)	1.67E-02 5.00E-03 5.49E-03 6.66E-04	0.01 PCA	1,020
Floor Drain Tank Rm (063-P28, P29)	184	Flange (2)	9.99E-04	Floor Drain Tank	170
Chemical Waste Tank Rm (063-P30, P31)	331	Flange (2)	9.99E-04	Chemical Waste Tank	170

Table 12.2-28 (7 of 8)

Cubicle	Volume (m^3)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms (1), (2)	Minimum Required Ventilation Flow (m^3/hr)
Detergent Waste Tank and Pump Rm (063-P32)	657	Pump seal(2) Flange (4) Valve 2″ (2) Valve 3″ (3) Valve 4″ (7) Evaporation from sump	1.67E-02 2.00E-03 6.66E-04 1.50E-03 4.66E-03 3.51E-02	Detergent Waste Tank	170
Chemical Drain Sump Pump Rm (063-P36)	119	Flange (2) Valve 2″ (1) Evaporation from sump	9.99E-04 3.33E-04 4.92E-02	0.01 PCA	934
Monitor Tank Rm (063-P37)	885	Flange (4)	2.00E-03	Monitor Tank	170
Monitor Tank Pump Rm (063-P54)	197	Pump seal (2) Flange (10) Valve 2.5″ (9) Valve 3″ (2) Valve 4″ (14)	1.67E-02 5.00E-03 3.75E-03 9.99E-04 9.33E-03	Monitor Tank	170
Valve Rm (085-P06)	180	Valve 4″ (1)	6.66E-04	1.0 PCA (Except for NG)	170

Table 12.2-28 (8 of 8)

Cubicle	Volume (m^3)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms (1), (2)	Minimum Required Ventilation Flow (m^3/hr)
Valve Rm (085-P15)	263	Valve 3″ (1) Valve 6″ (4)	5.00E-04 5.00E-03	1 PCA (Except for NG)	510
Valve Rm (085-P16)	269	Valve 4″ (1) Valve 6″ (2) Valve 6″ (3) Valve 6″ (1) Valve 4″ (2)	6.66E-04 2.00E-03 3.00E-03 9.99E-04 1.33E-03	0.32 PCA 0.32 PCA 0.1 PCA 0.44 PCA 0.01 PCA	1,530

(1) PCA: Fraction of primary coolant activity concentrations

(2) NG: Noble gases

(3) The HVAC flows listed in this table represent the actual minimum flow rates required for the ventilation for the corresponding rooms

Table 12.2-29

Vapor Activities in Pressurizer

Parameters Used in Pressurizer Vapor Activity Calculation

Parameter	Value
Liquid Volume (cm^3)	3.32E+07
Vapor Volume (cm^3)	3.57E+07
Liquid Mass (g)	1.99E+07
Vapor Mass (g)	3.52E+06
Temperature (°K)	619.44
Henry's Constant (Xe), atm	9.21E+03
Henry's Constant (Kr), atm	1.84E+03
Liquid specific volume, (cm^3/g)(1)	1.67
Vapor specific volume, (cm^3/g)(1)	10.16

Pressurizer Vapor Activities

Nuclide	Activity (Bq/g)
Kr-85m	7.5E+04
Kr-85	3.2E+05
Kr-87	5.9E+04
Kr-88	1.6E+05
Xe-131m	8.0E+06
Xe-133m	4.9E+05
Xe-133	5.2E+08
Xe-135m	1.1E+06
Xe-135	1.1E+07
Xe-137	2.5E+05
Xe-138	9.3E+05

(1) Saturation condition at 154.7 kg/cm^3A (2,200 psia)

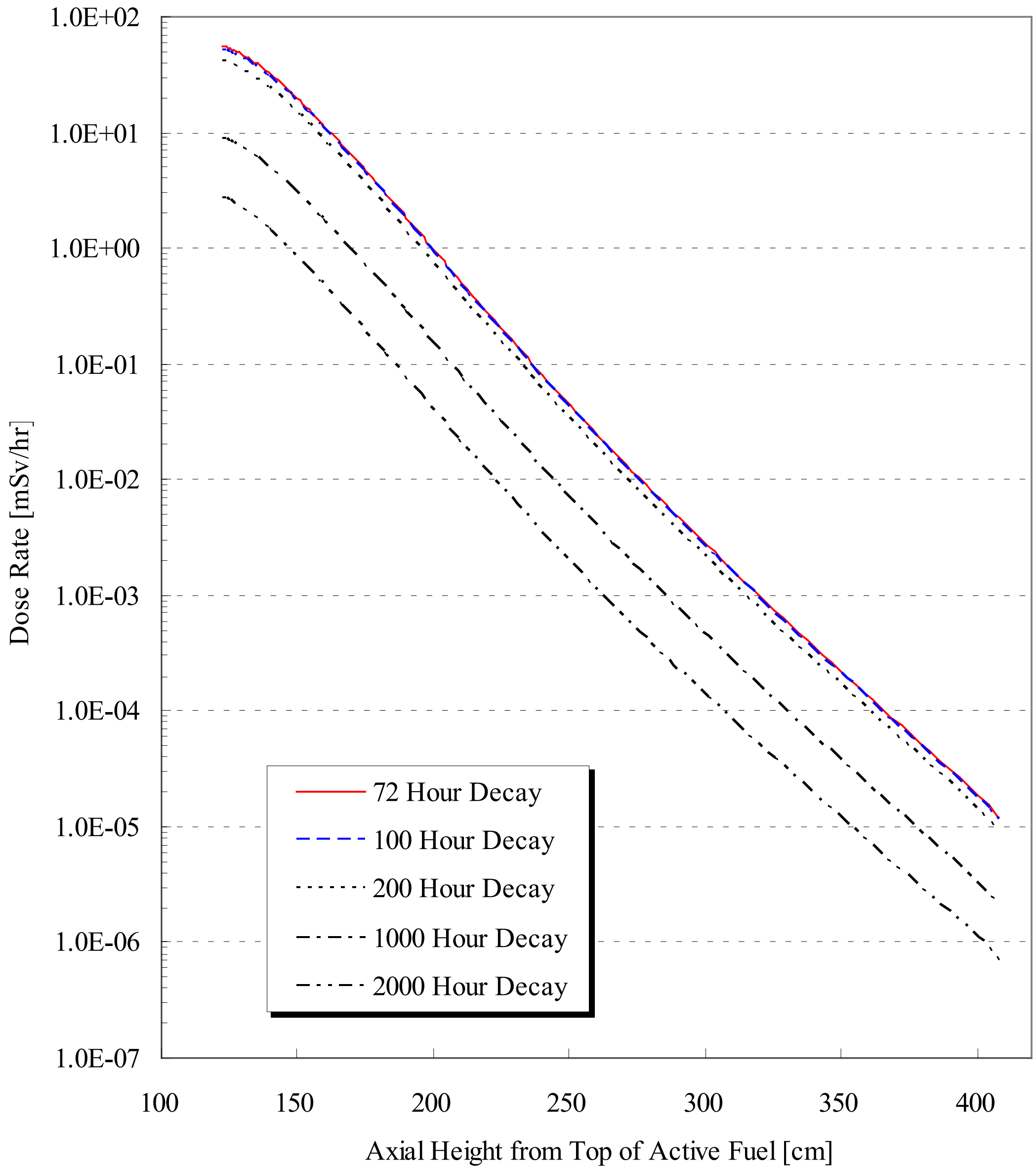
Table 12.2-30 (1 of 2)

Parameters Used in CVCS Inventory Calculations

Components	Parameters	Value	Components	Parameters	Value
Letdown HX	Tube Volume (cm^3)	270658	Equipment Drain Tank	Tank Volume (cm^3)	3.596E+07
Regenerative HX	Tube Volume (cm^3)	27984		Low Level Fraction	0.08
	Shell Volume (cm^3)	174886		High Level Fraction	0.37
CCP Mini Flow HX	Tube Volume	145738		RCS Activity Fraction	0.1
Purification IX	Vessel Volume (cm^3)	1578517		Flow rate (cm^3/sec)	2.19
	Resin Volume (cm^3)	906138	Volume Control Tank	Tank Volume (cm^3)	2.536E+07
Deborating IX	Vessel Volume (cm^3)	1578517		Low Level Fraction	0.31
	Resin Volume (cm^3)	906138		High Level Fraction	0.62
Pre-holdup IX	Vessel Volume (cm^3)	1578517	Holdup Tank	Tank Volume (cm^3)	1.590E+09
	Resin Volume (cm^3)	906138		Low Level Fraction	0.05
Boric Acid Condensate IX	Vessel Volume (cm^3)	1457384		High Level Fraction	0.20
	Resin Volume (cm^3)	906138	Reactor Makeup Water Tank	Tank Volume (cm^3)	1.495E+09
Seal Injection Filter	Vessel Volume (cm^3)	7570		Low Level Fraction	0.58
Reactor Drain Filter	Vessel Volume (cm^3)	15142		High Level Fraction	0.95
Boric Acid Filter	Vessel Volume (cm^3)	15142	Boric Acid Storage Tank	Tank Volume (cm^3)	9.464E+08
Purification Filter	Vessel Volume (cm^3)	15142		Low Level Fraction	0.40
Reactor Makeup Water Filter	Vessel Volume (cm^3)	15142		High Level Fraction	0.95
Reactor Drain Tank	Tank Volume (cm^3)	1.514E+07	In-containment Refueling Water Storage Tank	Tank Volume (cm^3)	3.295E+09
	Low Level Fraction	0.40		Low Level Fraction	0.76
	High Level Fraction	0.79		High Level Fraction	0.78
	Flow rate (cm^3/sec)	11			

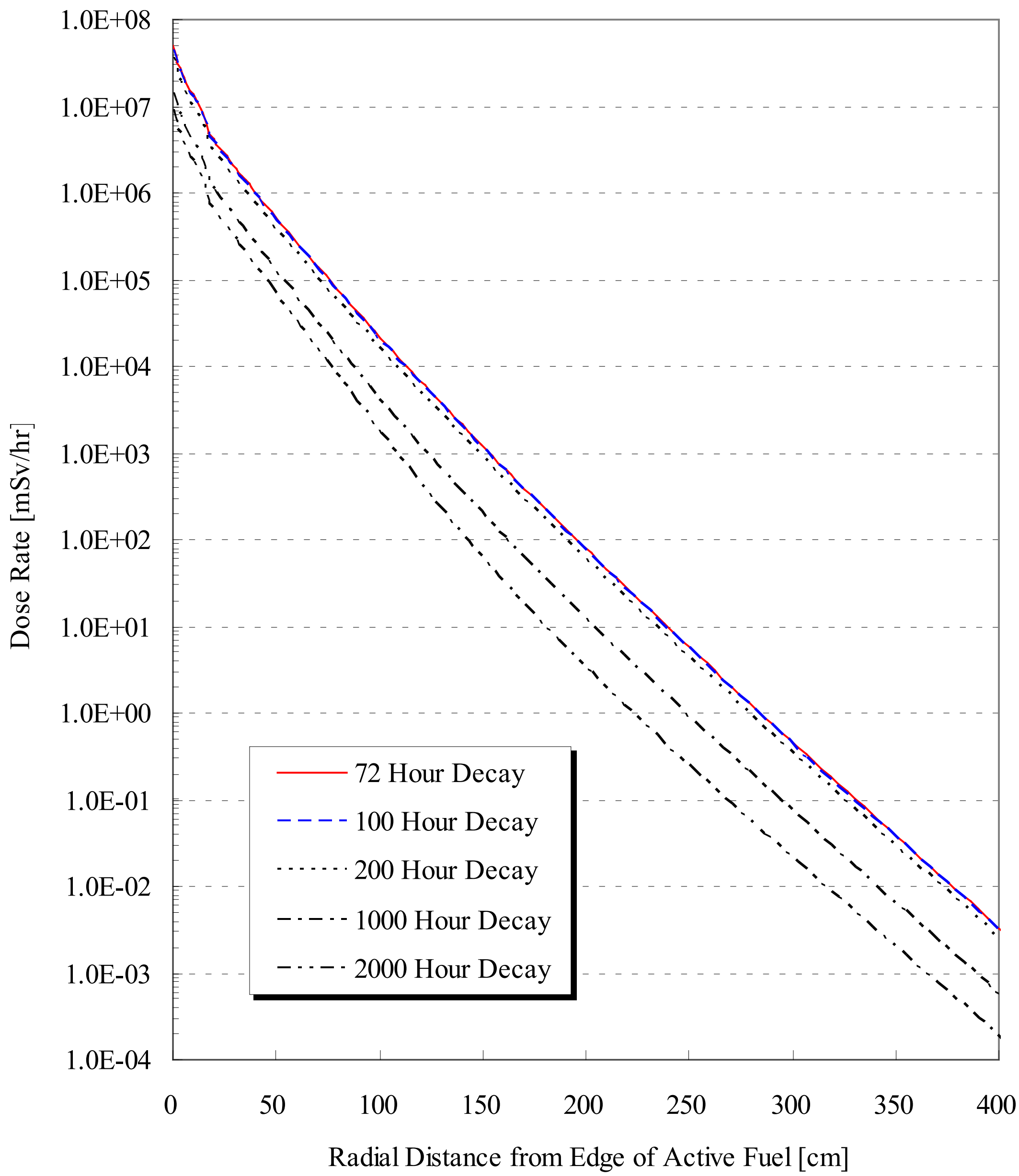
Table 12.2-30 (2 of 2)

Components	Parameters	Value	Remarks
Reactor Drain Tank	Partition Factor	0.001	Particulates
		10	Noble Gas
Equipment Drain Tank	Partition Factor	0.0001	Particulates
		10	Noble Gas
Volume Control Tank	Partition Factor	0.001	Particulates
		10	Noble Gas
Holdup Tank	Partition Factor	0.0001	Particulates
		10	Noble Gas
Reactor Makeup Water Tank	Partition Factor	0.0001	Particulates
		10	Noble Gas
Boric Acid Storage Tank	Partition Factor	0.0001	Particulates
		10	Noble Gas
In-containment Refueling Water Storage Tank	Partition Factor	0.0001	Particulates
		10	Noble Gas
Yearly Generated Waste	Volume (cm^3)	4.164E+09	
Boration Waste	Volume (cm^3)	1.772E+08	
Dilution Waste	Volume (cm^3)	3.358E+08	
Drained Coolant	Volume (cm^3)	1.616E+08	
Contraction Addition	Volume (cm^3)	1.268E+08	
RCS Partial Refill Addition	Volume (cm^3)	4.823E+08	



Note: Shielding effect of the refueling machine is included.

Figure 12.2-1 Spent Fuel Assembly Dose Rate vs. Axial Distance in Refueling Pool



Note: Shielding effect of the refueling machine is included.

Figure 12.2-2 Spent Fuel Assembly Dose Rate vs. Radial Distance in Refueling Pool

12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

The APR1400 design incorporates as low as (is) reasonably achievable (ALARA) principles per U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides (RGs) 8.8 (Reference 1) and 8.10 (Reference 2) to minimize onsite exposures to plant personnel during normal operation and accident conditions. Subsection 12.1.2 describes the approach for incorporating the ALARA principles into the plant layout, equipment selection and arrangement, and material selection. The plant layout is designed to provide adequate shielding between high-radiation areas and the low-radiation areas that are occupied by plant personnel. In addition, cubicles/rooms with similar radiation zone designations are grouped to the extent possible. These design approaches are implemented to provide reasonable assurance that occupational exposures are maintained below the limits established in 10 CFR Part 20 (Reference 3) and are maintained ALARA.

The following subsections detail the design features that provide reasonable assurance that operational and maintenance exposures are ALARA.

12.3.1.1 General Arrangement Design Features

a. Locations of radioactive systems and equipment

   Nonradioactive systems are physically separated from radioactive systems to the extent possible. This approach helps control the spread of contamination and minimizes the necessity of routing piping containing radioactive fluids or slurries through nonradioactive areas such as personnel corridors. The approach also facilitates access control of the radiation control areas and is consistent with NRC RG 8.8 Position 2.a and NRC RG 4.21 (Reference 4).

   Radioactive components are located in separate compartments wherever possible. The compartment design takes into consideration the frequency of access, operational requirements, and radiation level. For example, ion exchangers containing radioactive resins are located in separate compartments, and although the valves are remotely operated, they are located in a separate shielded gallery in the event manual operation or access for maintenance activities is needed. The compartment walls provide shielding, which enables personnel to perform operation and maintenance activities in a lower radiation area with significantly decreased radiation exposure. This approach is consistent with NRC RG 8.8, Position 2.

b. Pipe routing

Components with similar levels of radiological contamination with a direct interface are located close to each other to minimize pipe lengths. Piping carrying radioactive material from these components is routed through shielded pipe chases where practicable. The number of radioactive pipes inside each pipe chases is minimized to reduce the frequency of access into the pipe chase for maintenance activities. This approach is consistent with NRC RG 8.8, Position 2.b(10).

c. Component Access

Components have adequate internal space on all sides for easy access for maintenance and inspection. The spacing includes a laydown area or equipment pull area, as well as transport paths for removal or replacement of equipment. Rigging and lifting equipment is also provided to facilitate the removal, transport, or replacement of equipment or portable shielding during maintenance activities. This design is consistent with NRC RG 8.8, Position 2.a.

d. Hot tool cribs and hot machine shop

Hot tool cribs are located in low-radiation areas adjacent to personnel air locks to minimize the amount of wait time in high-radiation areas, to help prevent the spread of contamination to other plant areas, and to decrease the amount of decontamination work. This design reduces personnel exposure.

The provision of a hot machine shop in the compound building enables personnel to remove equipment and perform maintenance in a lower-radiation area. Access to the hot machine shop is also provided from the truck bay for ease of equipment movement.

e. Staging areas

Large staging areas outside the equipment hatch and personnel airlocks allow for prestaging prior to the start of an outage, and provide space for efficient radiation controls for moving equipment in and out of reactor containment building.

f. Personnel decontamination and change areas

Personnel decontamination and change areas are adjacent to radiation control area access points. Storage space for protective clothing, respirators, shower and toilet facilities, lockers, and containers for contaminated clothing are provided in these areas.

g. Radiation control area

The APR1400 design provides a single point of access into the radiation control areas; however, emergency egress is provided on all elevations. The access area to the radiation control area provides a flexible and adaptable layout to accommodate outage work crews and enhance the availability of immediate interaction with radiation protection personnel stationed at this point.

h. Accessways and entrances to high-radiation areas

Labyrinths are provided at the entrances to high-radiation areas to minimize exposure due to scattering and streaming of radiation through entrances and piping penetrations. The scattered dose rate through the passageway and the transmitted dose rate through the shield wall from all contributing sources shall be below the upper limit of the radiation zone specified for each area. This approach is consistent with NRC RG 8.8, Position 2.b(4).

The plant layout design can accommodate removable shields as necessary to provide shielding during normal operation for adjacent corridors. These shields can provide additional shielding during the removal of radioactive components, such as heat exchangers, for maintenance activities. The removable shield is designed to limit the radiation level so as not to exceed the radiation zone boundary in the surrounding areas. Therefore, its density is required to be greater than the permanent concrete shield wall and the gap with the concrete structure is designed to be less than 1/2” with an offset. This approach is consistent with NRC RG 8.8 Position 2.b (2).

High-radiation areas are provided with locked doors to prevent inadvertent access by plant personnel.

High-radiation areas and very-high-radiation areas are properly marked with signage to alert personnel and to further prevent inadvertent access as part of the radiation protection program (see Section 12.5).

Ingress and egress with offset inner walls with curbs are provided for the pumps and valve galleries for maximum radiation attenuation and shielding.

12.3.1.2 Equipment and System Design Features

The APR1400 design specifies the use of reliable and simplistic equipment with a minimal number of parts and straightforward operation to reduce the frequency of maintenance and radiation exposure to plant personnel. For the purpose of reducing the radiation field and the exposure to personnel during maintenance, surfaces will be polished in those areas of the plant where this treatment will significantly reduce the dose. The COL applicant is to determine the areas that will require either electro or mechanical polishing (COL 12.3(1)). The potential areas that may require polishing are as follows:

a. Steam generator channel head, including divider plate

b. Pressurizer shell (pressure boundary portion) except heads and nozzles

c. Reactor vessel closure head and bottom head excluding reactor vessel shells and flange

d. Reactor permanent pool seal (reactor cavity area)

e. Reactor coolant system main piping (pressure boundary portion) excluding surge line and nozzles attached on the piping

f. J-groove weld surface of the reactor vessel closure head

g. Regenerative heat exchanger

h. Spray valves

i. Refueling pool liner

j. Refueling cavity liner

k. Spent fuel pool liner

l. Cask loading pit liner

The design characteristics of the equipment used in radioactive systems are as follows:

a. Pumps

1) Pumps and associated piping are flanged to facilitate pump removal to a lower-radiation area for maintenance or repair. Pump internals are also removable. This design approach is consistent with NRC RG 8.8, Position 2.b (9).

2) All pump casings are provided with drain connections to facilitate decontamination. The drain connections are free of internal crevices to minimize accumulation of radioactive corrosion products (crud).

3) Pump seals are easily serviceable without the removal of the entire pump or motor. The seals of reactor coolant pump, centrifugal charging pump, shutdown cooling pump, and safety injection pump are of the cartridge type to facilitate their removal for maintenance or repair.

4) The reactor coolant pump casing is fabricated with austenitic stainless steel cladding and that the finished surface of the cladding is machined to have smooth surfaces to limit the buildup of radioactive contamination on its surface.

b. Ion exchangers (demineralizers)

1) Ion exchangers are designed for complete drainage.

2) Spent resin removal is designed to be done remotely by hydraulic flushing from the vessel to the SWMS.

3) Piping, strainers, and resin screens are flushable so that all spent resin is removed from the ion exchange column and connected piping. This minimizes radiation exposure resulting from spent resin transfer operations.

4) Fresh resin is added remotely from a low-radiation area above the shielded compartment housing the ion exchanger.

5) Ion exchangers are designed with minimal internal crevices to prevent an accumulation of radioactive cruds. This design approach is consistent with NRC RG 8.8, Position 2.f (6).

c. Liquid filters

1) Filter housings are provided with vent connections and are designed for complete drainage.

2) Filter housings are designed with minimal internal crevices to prevent an accumulation of radioactive cruds.

3) Adequate space is provided near cartridge filters to allow for cartridge removal and loading for transport to the solid waste storage area.

4) Filter housings and cartridges are designed to permit the remote removal of spent radioactive filter elements. Cartridge filter seals are an integral part of the filter cartridge and can be removed concurrently with the filter elements. This design approach is consistent with NRC RG 8.8, Position 2.i (4).

5) Cartridge filter housing closure heads are designed to swing free for the unobstructed removal of the cartridge.

6) Design features to handle spent filters are described in Subsection 11.4.2.

d. Tanks

1) Tanks are designed for complete drainage and are therefore free of internal crevices and pockets. The drain line is connected to the bottom of the tank.

2) Tanks are provided with at least one of the following means of decontaminating the tank internals based on tank contents and radioactivity levels:

   a) Ample space to permit decontamination of the tank manway

   b) Internal spray nozzles on potentially highly contaminated tanks for internal decontamination

   c) Backflush capability for tank inlet screens

3) Tanks are designed with ellipsoidal or sloped bottoms to facilitate drainage and minimize the accumulation of crud.

4) Tanks are provided with vents to facilitate the removal of potentially radioactive gases. The tanks are vented to the cubicle atmosphere within close proximity to the respective radioactive building HVAC system intake for collection and treatment of radioactive gases. Tanks are provided with internal filters on the vent lines to prevent the release of liquids

5) Non-pressurized tanks are provided with overflow lines routed to a floor drain pump or other suitable collection point to avoid spillage of radioactive fluids onto the floor or ground. The floor drain system (FDS) is connected to the LWMS for additional processing prior to release to the environment.

6) All tanks containing suspended solids are provided with mixing capabilities to prevent settling of radioactive solids.

e. Valves

1) The following descriptions summarize the valve design features that minimize valve leakage and extend valve design life.

a) A packless valve is used as an isolation valve that is non-modulating, is less than 5.08 cm (2 in.) in diameter, and operates on a weekly interval in the radioactive material processing systems to minimize external leakage from the valves.

b) Modulating valves or valves greater than 5.08 cm (2 in.) in diameter use live loading of the packing by conical spring washers or equivalent means to maintain a compressive force on the packing, in addition to double stem packing with a leakoff between the packings and a graphite lantern ring, where possible.

c) For the valves 5.08 cm (2 in) and under, excluding modulating valves, just double stem packing with a leakoff between the packing and a graphite lantern ring is used. Stem leakage is piped to an appropriate drain sump or tank. All lantern rings used are graphite lantern rings.

All valves with stem leakage are shown on Figures 5.1.2-3, 6.3.2-1, and 9.3.4-1.

d) Valves using stem packing are provided with a backseat capability.

e) Radiation-resistant seals, gaskets, and elastomers are used, when practicable, to extend the design life and reduce maintenance frequency.

f) Valves located in high-radiation areas are equipped with reach rods or motor operators to allow operation from lower radiation zones to minimize radiation exposure.

2) Fully ported valves are used so that valves are fully open and radioactive fluid flows freely. This minimizes internal accumulation of cruds.

3) Valves requiring removal during maintenance and inspection activities are flanged.

4) Internal valve surfaces are designed to be as smooth as possible and free of crevices to minimize the accumulation of crud.

5) Valve wetted parts are made of austenitic stainless steel or a similar corrosion-resistant material.

6) Valves are designed so that they may be repacked without removing the yoke or topworks.

7) Valves for highly radioactive components are located in shielded valve galleries to the extent possible to minimize operator exposure.

8) Check valves are used only where necessary. The type and size of valve selected are compatible with system requirements to reduce disc flutter related wear. Check valves are located and oriented properly in the piping system.

9) The application of metallic bellows and diaphragms is limited to the valves with low-stroke length applications, or infrequent movement in the APR1400 design.

f. Piping and penetrations

1) The field run piping is minimized to the extent practicable to reduce the potential for unintended leakage to the environment.

2) Resin and concentrate piping is designed as follows:

   a) The length of pipe runs is minimized by locating the related components as close to each other as possible.

   b) Piping is routed through shielded pipe chases whenever possible to minimize the radiation exposure rate in personnel access corridors.

   c) Large-diameter piping (greater than 12.7 cm [5 in.] in diameter) is used to minimize the potential for clogging during slurry or resin transfer without violating minimum flow requirements.

   d) The number of pipe fittings, such as elbows and tees, is minimized to reduce the potential for radioactive crud accumulation. Where elbows are needed, large-radius bends are used.

   e) Low points, deadlegs, elbows, and vertical pipe runs that may cause accumulation of radioactive material are minimized.

   f) Pipe runs are sloped and gravitational flow is used where practicable.

   g) Crevices on piping internal surfaces are minimized by the use of butt welds instead of socket welds. Socket welds are known to produce crud traps in radioactive systems. The use of butt welds generally results in smoother internal surfaces, reducing crud buildup.

   h) Flushing capability is provided to facilitate the decontamination of piping containing radioactive fluids.

   i) Penetrations are located so that the source and penetration are not in a direct line and are located high above the floor where practicable. This minimizes the potential for personnel exposure due to radiation streaming pathways.

g. Heat exchangers

1) Heat exchangers are designed for complete drainage and with vents.

2) Internal wetted surfaces are designed to be free of crevices to minimize the potential for the accumulation of radioactive crud on internal surfaces.

3) Corrosion-resistant materials, such as stainless steel, are used to minimize the need for part replacement and reduce the frequency of required maintenance.

h. Reactor vessel head vent

A vent nozzle and line are provided on the reactor vessel (RV) head. Use of this design feature allows for a reduction in worker radiation exposure during the head removal process by minimizing the gases discharged directly to the reactor containment building atmosphere while the head is being removed.

i. Reactor coolant system leakage control

Exposure from airborne radionuclides to personnel entering the reactor containment building is minimized by controlling the amount of reactor coolant leakage released to the reactor containment building atmosphere. Examples of such controlled leakage are as follows:

1) Pilot-operated safety relief valve (POSRV) leakage is directed to the in-containment refueling water storage tank (IRWST).

2) Valves larger than 5.08 cm (2 in.) in diameter are provided with a double-packed stem with an intermediate graphite lantern ring and a leakoff connection to the reactor drain tank (RDT) with emphasis on the valve stem finish to reduce wear and leakage.

3) Instrumentation is provided to detect abnormal reactor coolant pump (RCP) seal leakage. The RCPs are equipped with two stages of seals, plus a vapor or backup seal, as described in Section 5.4. The vapor or backup seal prevents leakage to the reactor containment building atmosphere and allows sufficient pressure to be maintained to direct the controlled seal leakage to the volume control tank (VCT) and RDT. The vapor seal is designed to withstand full reactor coolant system (RCS) pressure in the event of failure of any or all of the two primary seals.

j. Refueling equipment

1) All spent fuel transfer and storage operations are designed to be conducted underwater to provide reasonable assurance of adequate shielding and limit the contribution of radioactivity in working areas.

2) Equipment is designed to prevent the fuel from being lifted above the minimum safe water depth, thereby limiting personnel exposure and avoiding fuel damage.

3) The equipment design limits the possibility of inadvertent fuel drops, which could cause fuel damage and personnel exposure.

4) The refueling equipment design facilitates the transfer of new and spent fuel at the same time to reduce overall fuel handling time and personnel exposures.

5) Underwater cameras are used to facilitate safe handling through visual control, thus minimizing errors and potential exposures.

6) Portable hydraulic cutters are provided to cut expended control element assemblies (CEAs) and in-core instrumentation (ICI) leads. The cutters allow underwater handling of these items.

7) Equipment is provided to allow for the underwater inspection of fuel elements.

k. Inservice inspection equipment

Inspection of the reactor coolant pressure boundary can be done with remote equipment to minimize personnel exposures.

l. Remote instrumentation

Systems containing radioactive fluids are designed to be controlled remotely to the extent practicable to minimize personnel exposures.

m. Inservice inspection (ISI) of reactor vessel nozzle welds

The design of welds joining the RV nozzle to the reactor coolant pipe permits ISI to be accomplished from inside the RV. Automated equipment, operated remotely, can be used for reactor vessel pressure boundary inspections in this area.

In the event that an ISI of this area is performed from outside of the RV, removable insulation for the RV and reactor coolant piping is used to permit access. These removable sections are lightweight and held in place mainly by quick-application type buckle fasteners. After the necessary panels are removed, remote equipment can be used to perform the required inspections.

n. Blanket-type thermal insulation

Blanket-type thermal insulation with hook-and-loop fasteners is selected where needed for components and piping containing radioactive fluids. A metal jacket around the insulation is provided. This jacket is held in place by quick-application type buckle fasteners. This insulation is easily removable to facilitate the performance of inservice weld inspections. This minimizes personnel exposures received during ISI.

o. Electrical service and lighting

The APR1400 design provides quality lighting and convenient electrical services to facilitate maintenance and inspection and reduce anticipated personnel exposure. Reliable lamps of extended service life are used in high-radiation areas whenever possible to minimize the frequency of maintenance/replacement. These features are included in the facility layout design in accordance with the guidance of NRC RG 8.8, Position C.2.i.

p. Spent fuel pool (SFP) and refueling pool decontamination

The APR1400 design provides the capability to use high-pressure demineralized water for the decontamination in the SFP and refueling pool. When high pressure demineralized water is used to decontaminate the SFP, and refueling pool, or bulky components, protective covers are required to be used to minimize the spread of contamination. High pressure demineralized water is not to be used in open areas above the pool. Alternative methods of decontamination, such as a strippable coating, may be evaluated by the operator for practicality.

q. Snubbers

Mechanical snubbers, rather than hydraulic snubbers, are used in radiation areas to minimize the frequency of required maintenance and inspections.

12.3.1.3 Source Term Control

Source term control is an important aspect of the APR1400 design. The following design features reduce the doses received by plant personnel from operations, maintenance, and inspection activities:

a. Fuel performance

The APR1400 design features provide reasonable assurance of low primary system source terms, not only because of the extended fuel cycle, but also because of minimized fuel clad leakage based on extremely low fuel clad defects.

b. Corrosion product control

The APR1400 design includes design features that reduce corrosion product production in the primary system.

1) Primary system materials

The APR1400 design specifies that primary system components are to be fabricated from materials with low corrosion rates and low cobalt impurities (target of 0.02 wt% or less) except where no proven alternative exists. This approach is consistent with NRC RG 8.8, Position 2.e.

Cobalt content for components with a large wetted surface area and an operating temperature of greater than 93 °C (200 °F) is restricted to a maximum of 0.05 wt%. For components in or near the core or components that are expected to release significant quantities of corrosion products, cobalt content (mean value) lower than 0.05 wt% is specified.

The cobalt content for primary system materials are presented in Table 12.3-1.

The presence of antimony in RCP bearings has presented a problem with hot particles in the current generation of nuclear plants. In the APR1400 design, RCP bearings are designed to minimize the amount of antimony.

Nickel-based alloy is used for SG tubes based on industry experience for similar applications in Korean domestic plant.

SG tubes are fabricated to relieve stresses to reduce stress corrosion cracking (SCC), thus reducing the probability that the tubes will plug and reducing radiation exposure to maintenance workers.

Control rod drive materials are specified to be fabricated from low-cobalt alloys to the extent possible.

Cobalt-base hard-facing alloys are used to provide wear resistance only when no proven alternative exists.

2) Primary system chemistry

Increasing the primary system water chemistry pH from 6.9 to 7.4 reduces equilibrium corrosion rates and the buildup of activated corrosion products on primary system surfaces. Details are addressed in Subsection 5.2.3.2.1 for the primary system water chemistry pH control. The COL applicant is to establish how the water chemistry pH control reduces radiation fields (COL 12.3(2)).

12.3.1.4 Airborne Contamination Control

In the APR1400 design, plant HVAC systems are designed so that airflow is maintained from low-radiation areas to high-radiation areas. This design minimizes the potential for the spread of contamination to low-contamination areas and is consistent with NRC RG 8.8, Position 2.d(1). The reactor containment building is monitored for particulate, iodine, and gaseous activity. Indication of high containment activity automatically initiates containment purge isolation. This approach meets the intent of NRC RG 8.25, Regulatory Position c.2.1 (Reference 5). In addition, the following confinement devices are used to minimize the spread of contamination in accordance with the ALARA principle and NRC RG 4.21 (Reference 4):

a. Leakage inside reactor containment building

Leakage inside the reactor containment building is collected in local sumps and transferred to floor drain tanks for processing and release. The sumps are designed to be smaller for more frequent transfer to the LWMS to minimize evaporation and suspension of radioactive particulates inside the reactor containment building. Additional design requirements for the reactor containment building sump and the ICI sump are provided in Subsection 9.3.3.2.4.d.

b. Ventilation

HVAC systems are provided for individual buildings as part of the APR1400 design. Each HVAC system is designed to provide the proper ventilation flow to remove contaminated gases and vapor to levels below the limits of 10 CFR Part 20, Appendix B, Table 1, and to keep building areas, cubicles, walkways, and control areas in the proper environmental conditions (temperatures and air quality). For maximum efficiency, airflow generally is directed from clean/low-contamination areas to higher-contamination areas at a velocity suitable for minimizing entrainment of moisture and particulate. There is also system redundancy to minimize interruption of continuous operation. Exhausted airflows are processed through high-efficiency particulate air (HEPA) filters, charcoal adsorbers, and monitors for radiological contamination before being discharged into the atmosphere. This design approach provides proper airborne contamination control and meets the requirements of NRC RG 8.8, Position 2.d. HVAC systems are addressed in Subsection 9.4.8.

c. Hot machine shop

This area provides a dedicated area where maintenance can be performed on radioactive and contaminated equipment. The hot machine shop allows for maintenance and repair activities to be performed in a lower-radiation area with plenty of space for workers to facilitate the efficient completion of the maintenance task.

d. Loop seals

Water-filled loop seals are provided in the floor drain system to preclude the flow of contaminated material from one area or floor to another.

12.3.1.5 Equipment Improvements

a. The APR1400 design RCPs incorporate a cartridge type of RCP seal, which is a proven, reliable, and easily replaceable seal design. The replacement of RCP seals is also facilitated by platforms around the RCPs. This design allows the seal to be removed and repaired outside the crane wall or in another low-radiation area, consistent with NRC RG 8.8, Position 2.b (9). Therefore, the amount of time needed to maintain the RCP seals and the associated occupational exposure is reduced.

b. Steam generator maintenance

The APR1400 design includes several features that enhance accessibility to steam generators during maintenance and inspection. These features, described in Subsection 5.4.2, reduce the overall exposure to personnel during maintenance and inspection activities. The features include:

1) Use of automatic or robotic equipment to perform inspection and maintenance activities

2) Adequate pull and laydown areas

3) Platforms to facilitate access to SG parts for maintenance and inspection

4) Handholes to facilitate worker handling of SG parts

5) Adequately sized manways to facilitate easy access for workers performing maintenance and inspection activities

6) Removable insulation to facilitate weld inspection

7) Alloy 690 for tubes to reduce corrosion product production

c. The APR1400 design includes features that are important to achieving ALARA goals. The features are summarized as follows:

1) Equipment selection that is based on reliability, maintainability, and accessibility

2) Component design such as tank design, piping design, and instrument design to minimize particulate deposition

3) System flushing and decontamination capability for systems and components containing radioactive materials

4) Radwaste handling operations are performed remotely, wherever practicable

5) Separation of radiologically contaminated components and the provision of adequate shielding surrounding the rooms/areas containing radioactive components

6) Controlled access to high-radiation areas via locked doors and administrative controls

7) Piping containing radioactive liquid, resins, or gases routed through shielded pipe chases, wherever possible

12.3.1.6 Radiation Zone Designation

In order to maintain radiation exposures ALARA throughout the layout and shielding design, the plant is divided into several radiation zones. The zones are used in the development of plant procedures to provide reasonable assurance that maintenance, inspection, and testing activities are performed in the lowest radiation zone possible. The zones indicate maximum dose rates based on the shielding design basis source terms presented in Section 12.2. The radiation zone designations during normal operation are summarized in Table 12.3-3. The radiation zone maps for the auxiliary/reactor containment building are shown in Figures 12.3-1 through 12.3-9. The radiation zone maps for the compound building are shown in Figures 12.3-10 through 12.3-16.

The turbine generator building is generally a non-radiation area. The potentially radioactive area is the condensate polisher room during normal operation and AOOs. The potential radioactivity is from the removal of radiological contaminants by the polishing demineralizers.

Because of the potential for primary-to-secondary (PTS) SG leakage or permeation through SG tubes, the secondary-side plant systems may have a low level of radioactive contamination. The SG blowdown liquid is provided with radiation monitors to detect tube leakage; the condenser vacuum vent effluent radiation monitor also provides an indication of an SG tube leak.

The monitoring systems provide continuous monitoring of the SG performance and while the condensate system and blowdown system process secondary water during SG leakage events; the activities of the spent resins and filter cartridges are continuously monitored and are not anticipated to be significantly contaminated by PTS leakage. The condensate polisher room and the condenser areas are also provided with area radiation monitors to provide reasonable assurance that personnel radiation exposure is maintained ALARA. Radiation zone maps for the turbine generator building are presented in Figures 12.3-17 and 12.3-18.

There is a potential for the CCW system to become contaminated through leakage from interfacing radiologically contaminated systems. The CCW system is provided with radiation monitors to ensure that the contamination level is low. As shown in Figures 12.3-19A and 12.3-19B, the CCWS heat exchangers are located in a separate yard building that is designated Radiation Zone 1.

12.3.1.7 General Design Considerations to Keep Post-Accident Exposures ALARA

Direct and airborne sources of radiation exposure are considered when determining access provisions to vital areas necessary for the control of the plant. The plant design provides reasonable assurance that personnel exposures meet General Design Criterion (GDC) 19 of 10 CFR Part 50, Appendix A (Reference 6), and NUREG-0737 (Reference 7) guidelines.

To provide reasonable assurance that personnel exposures are kept ALARA, the APR1400 is provided with a post-accident sampling system (described in Subsection 9.3.2) that meets the requirements of NUREG-0737 and NRC RG 1.97 (Reference 8).

12.3.1.8 Post-Accident Radiation Zones

Post-accident radiation zone maps are developed in accordance with NRC RG 1.183 (Reference 9) to assess the access throughout the plant following a DBA. The layout of the plant facilities is designed with the intent of keeping occupational doses ALARA during a DBA as well as during normal operation. Post-accident source terms are described in Subsection 12.2.3 and Table 12.2-26.

Continuous access is provided during post-accident conditions, with dose rates less than or equal to 0.15 mSv/hr, to the following vital areas:

a. Main control room

b. Technical support center

Required access to the following vital areas and systems does not exceed the occupational dose limit of 50 mSv:

a. Post-accident sampling system

b. Remote shutdown room and remote control console room

c. Class 1E switchgear room

d. I&C equipment room

e. Access areas outside the CS and SC pump rooms

Generic plant emergency procedures are reviewed and the areas listed above are identified as vital areas. The following systems are considered for post-accident access, but do not constitute vital areas:

a. Safety injection system

b. Containment spray system

c. Shutdown cooling system

d. Chemical and volume control system

The safety injection system, containment spray system, and shutdown cooling system are all safety-related systems with ESF functions, which are required during design basis accident conditions in order to achieve safe shutdown and mitigate the accident. However, these functions which are required during accident conditions can be monitored and controlled remotely from the MCR and RSR. Therefore, the plant areas housing these system components (with the exception of the area outside of the CS/SC pumps) do not directly require operator access and are therefore not considered vital areas.

The zone limits are summarized for the design basis loss-of-coolant accident (LOCA) in Table 12.3-4. Post-accident radiation zone maps are shown in Figures 12.3-20 through 12.3-51.

12.3.1.9 Vital Area Access

The following descriptions detail the post-accident access routes to the vital areas listed in Subsection 12.3.1.8. Vital area access routes are illustrated by arrows in Figures 12.3-20 through 12.3-51. During normal operating conditions, the radiation control area has a single point of access in the compound building at elevation 30.48 m (100.0 ft). However, during post-accident conditions, emergency access and egress are possible on all elevations if needed.

a. Main control room (MCR)

   The MCR is occupied continuously during post-accident situations. The design dose for this area does not exceed 0.15 mSv/hr averaged over 30 days for a TEDE of 50 mSv. The MCR is located in the auxiliary building at El. 47.5 m (156.0 ft). Access to the MCR is via the compound building. It is generally accessible on all elevations from stairwells and elevators located at 14-15, AH-AJ, and AB-AD. The locations of the MCR and associated access routes are shown in Figure 12.3-40.

b. Technical support center (TSC)

   The TSC is continuously occupied during post-accident situations. The design dose for this area does not exceed 0.15 mSv/hr averaged over 30 days for a TEDE of 50 mSv. The TSC is located in the auxiliary building at El. 47.5 m (156.0 ft). The TSC is generally accessible on all elevations from stairwells and elevators located at 14-15, AH-AJ, and AB-AD. The locations of the TSC and associated access routes are shown in Figure 12.3-40.

c. Post-accident sampling system (PASS)

   The PASS area is irregularly accessed, not continuously occupied, during post-accident situations. The design dose for these areas may not exceed the TEDE of 50 mSv. The post-accident sampling room is located in the auxiliary building at El. 16.8 m (55.0 ft) and the analysis room is in the compound building at El. 25.9 m (85.0 ft). The locations of the post-accident sampling room, analysis room and associated access routes are shown in Figures 12.3-20 and 12.3-12.

d. Remote shutdown room (RSR) and Remote control console room (RCCR)

The RSR and RCCR are accessed to safely shutdown the reactor in the event that the main control room is inaccessible. The design dose for these areas may not exceed the TEDE of 50 mSv. The RSR and RCCR are located in the auxiliary building at El. 41.9 m (137.5 ft). The locations of RSR and RCCR and associated access routes are shown in Figure 12.3-36.

e. Class 1E switchgear room

The four (4) class 1E switchgear rooms house the safety-related equipment, which require infrequent access to perform the vital function. The design dose for these areas shall not exceed the total effective dose of 50 mSv. The class 1E switchgear rooms are located in the auxiliary building at an elevation of 23.8 m (78.0 ft). The locations of the class 1E switchgear rooms and associated access routes are shown in Figure 12.3-24.

f. I&C equipment room

The four (4) I&C equipment rooms house the safety-related instrument and control systems, which require infrequent access to perform the vital function. The design dose for these areas shall not exceed TEDE of 50 mSv. The I&C equipment rooms are located in the auxiliary building at an elevation of 47.5 m (156.0 ft). The locations of the class 1E switchgear rooms and associated access routes are shown in Figure 12.3-40.

g. Access areas outside the CS and SC pump rooms

The areas outside the CS and SC pump rooms are irregularly accessed, when the manual actuation is required after post-accident situations. The manual actuation involves the opening and closing of valves for the interchanging of the CS and SC pumps and the opening or closing of valves for the connections of these two systems to the IRWST. These configurations may be implemented as back-up mitigation measures depending on the changing post-accident conditions. The design dose for these areas shall not exceed TEDE of 50 mSv. These areas are located in the auxiliary building at an elevation of 16.8 m (55.0 ft). The locations of these areas and associated access routes are shown in Figure 12.3-20.

12.3.2 Shielding

The shielding design is based on the source terms, design dose rates, and established design criteria in Subsection 12.2.1 and is in accordance with the calculation method and guidance in NRC RG 1.69 (Reference 10) and NRC RG 8.8.

12.3.2.1 General Shielding Design Criteria

Shield walls are provided around components that contain and handle radioactive materials for worker safety and to maintain radiation doses ALARA. The wall thicknesses listed in Table 12.3-5 are based on the shielding basis source terms of the component, the design dose rate, and the shielding material. This approach is consistent with NRC RG 8.8.

The shielding design for the MCR and the primary shielding in the reactor containment building is safety-related. The shielding for the MCR meets the requirements of 10 CFR Part 50, Appendix A, GDC 19. Within the reactor containment building, the pressurizer is separated from the operating floor by a shield wall with a minimum thickness of 33 inches. The steam generators, reactor coolant pumps, and associated RCS piping are also separated from the operating floor and annulus area by the secondary shield wall which has a minimum concrete thickness of 48 inches.

Radiation protection of personnel, equipment, and materials is dependent primarily on the adequacy of the design of the plant shielding. Radiation shielding has the passive protection function of radiation attenuation and consists of materials placed between radiation sources and plant personnel as well as the public. The shielding system is designed and constructed to provide reasonable assurance that the station can be operated and maintained so that the resultant radiation levels and doses are below the limits of 10 CFR Part 20 and are ALARA. Design dose rate limits to achieve this objective are addressed in Subsection 12.3.1 and listed in Table 12.3-3.

Shielding is designed for the plant lifetime and under the various radiation source and environmental conditions associated with normal operation, AOOs, and DBA conditions identified in Chapter 15 and summarized as follows:

a. Plant conditions

  1) Normal operating condition

For shielding design, normal station operating conditions are considered to include conditions generally known as AOOs. Two modes of normal station operation are:

a) Normal power operation of the reactor

b) AOOs (startup, shutdown, and refueling of the reactor)

Shielding is designed to provide a protective function under such conditions. Normal operation zone designations are provided in Table 12.3-3.

2) Accident conditions

Shielding provides protection to plant operating personnel and the general public under postulated DBA conditions as defined in Chapter 15.

a) MCR habitability

The MCR and associated areas are designed according to the design requirements in 10 CFR Part 50, Appendix A, GDC 19, and are shielded so that, after a postulated DBA, radiation exposure in the MCR for the duration of the accident does not exceed the TEDE of 50 mSv, including dose contributions from ingress and egress of the MCR.

The radiation shielding protecting the MCR and associated areas is designed based on the anticipated radiation environment resulting from the postulated DBA.

b) Direct offsite doses

Adequate shielding is provided to limit the total effective dose equivalent not to exceed the limits specified in 10 CFR 52.47 (Reference31) and Chapter 15 of the Standard Review Plan (SRP) (Reference 12). The shielding is designed adequately that the radiation dose to an individual located at the exclusion area boundary for a duration of 2 hours during a postulated DBA does not exceed 25 rem TEDE.

b. Seismic and safety classification

Structural walls are designed to meet seismic category requirements. Structural walls are designed as seismic Category I, II, or III depending on the particular design requirements other than the radiation protection requirements such as structural integrity and load-bearing capacity.

The primary shield, the shield walls for the MCR, and the shield walls for the SFP are examples of shield walls that are designed as seismic Category I.

c. Protection of equipment and structures

Adequate shielding is provided for the following purposes:

1) To limit radiation heating of structural concrete

2) To reduce neutron activation of equipment

3) To limit the radiation dose to equipment and materials

d. Maintenance, inspection, and testing considerations

Adequate shielding is provided to provide reasonable assurance of safe personnel access and sufficient stay time near areas containing radioactive equipment for maintenance, inspection, and testing activities.

e. Additional requirements

The other shielding systems functional requirements generally depend on the location of the shield and the access requirements to or from the equipment or areas within the shield walls. Thus, access to an area may be through the shield itself such as through removable shield walls.

12.3.2.2 Shielding Analysis

Calculations to determine the adequacy of the shielding design are based on the source strengths described in Subsection 12.2.1 and the methods described below. Dose points are selected for analysis inside and outside cubicles containing radioactive equipment. Skyshine from the cubicles is negligible because cubicles containing radioactive material are shielded overhead.

The only major source generating radionuclides is the reactor core at full power. The codes ANISN (Reference 13) and MCNP (Reference 14) are used to verify the effectiveness of the primary shield and evaluate the streaming of neutron and gamma radiation from the reactor vessel. Sources of gamma radiation are distributed throughout the reactor containment building and nuclear island. The codes MICROSHIELD (Reference 15) and RUNT-G (Reference 16) are used to verify gamma source shielding. The shielding analyses are based on a concrete density of 2.242 g/cm^3 (140 lbs/ft^3). The following sequence typifies a gamma source shielding analysis:

a. Determine the concentration of each principal nuclide in the source medium.

b. Adjust the concentration to account for issues such as accumulation, dilution, decay, and removal.

c. Convert the resulting concentrations into gamma source strength.

d. Select a model or combination of models to represent the physical geometry of the source container(s) and all shields present.

e. Assemble the necessary data on attenuation properties of the source and the shield materials.

f. Perform the calculation for the desired dose point location and tabulate the results for comparison with design objective dose rates.

Computer codes necessary to perform the above analysis meet the related regulatory requirements and industry standards. Computer codes used in shielding analysis are described below:

a. ANISN

ANISN is a program used to perform the shielding and heat generation rate calculations for the reactor primary shield inside of containment. ANISN is a multigroup, one-dimensional, discrete-ordinates transport program that solves the one-dimensional Boltzmann transport equation for neutrons and gamma rays in slab, sphere, or cylindrical geometry.

b. MCNP

MCNP is a continuous-energy, generalized-geometry, time-dependent, coupled neutron-photon-electron Monte Carlo transport code system. MCNP is widely used for verification of shielding design adequacy in dose rate reduction because it provides more realistic modeling capabilities and more reliable calculation results than other programs. MCNP is used in shielding calculations as described in Section C.1.a of NRC RG 1.69. This code is used to evaluate the dose rate impact on the operating floor in the containment building as a result of the streaming of neutron and gamma radiation from the reactor vessel through the reactor cavity.

c. MICROSHIELD

MICROSHIELD is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used for designing shields, estimating source strength from radiation measurements, and teaching shielding principles. MICROSHIELD has 16 geometries that accommodate offset does points and as many as 10 standard shields, plus source self-shielding and cylinder cladding. This code uses library data for radionuclides, attenuation, and buildup and the dose conversion reflects standard data from International Commission on Radiological Protection (ICRP) Publications 38 and 107, as well as ANSI/ANS standards and Radiation Safety Information Computational Center (RSICC) publications. This code is used to determine the radiation zoning and minimum shield thickness requirements of the cubicles containing radioactive sources.

d. RUNT-G

RUNT-G uses a modified version of the multiple-compartment model from the RACER-II program to calculate time-dependent radionuclide activities, the source geometry models from the Gaussian quadrature version of the ISOSHLD (Reference 17) program to calculate the resulting time-dependent dose rates, and a simple trapezoidal integration scheme to determine cumulative doses. All of the containment model parameters used in RACER-II and shielding geometries available in ISOSHLD are included in RUNT-G. This code is used to determine the post-accident radiation shielding requirements to provide adequate accessible conditions and to calculate the total integrated dose (TID) for equipment qualification.

12.3.2.3 Shielding Design

The plant shielding is designed to achieve the radiation zones designated in Tables 12.3-3 and 12.3-4 for normal operation and post-accident conditions, respectively.

The shielding design thicknesses, as presented in Table 12.3-5, are based on the use of ordinary concrete as the shield material.

In accordance with Standard Review Plan 12.3-4, Sections II.1 and III.3 and Appendix B, and NRC RG 8.38 (Reference 18) C.1.6(3), adequate shielding or operational precautions are provided for all accessible plant areas capable of radiation levels greater than 1 Gy/hr. Areas with the potential for radiation greater than 1 Gy/hr are listed in Table 12.3-6.

Transient sources greater than 1 Gy/hr are considered in the shielding design to provide reasonable assurance that adequate shielding is provided. One such source is a spent fuel assembly. During transfer of a spent fuel assembly through the fuel transfer tube, adjacent areas may have elevated radiation levels. Streaming from this source up through the joint between the reactor containment building and the auxiliary building has been a concern for the current generation of nuclear plants. The APR1400 design uses connected building structures to reduce the potential for streaming. In addition, sufficient concrete shielding is provided to maintain radiation levels in adjacent areas ALARA during spent fuel transfer. This permits personnel to perform maintenance and inspection activities in a lower-radiation area and reduces the potential for high-radiation levels adversely affecting refueling outage schedules. An inspection area is provided for the fuel transfer tube. Access control is provided by the personnel airlock through the reactor containment building. Limited access to the annulus areas within the reactor containment building is allowed during power operation with a tightly controlled occupancy time. Access to these areas is restricted through administrative controls as well as a physical lock on the door.

Sufficient shielding provides reasonable assurance that the areas adjacent to the spent fuel transfer tube are accessible and expected radiation zones are consistent with those in Figure 12.3-52 during transfer of a spent fuel assembly. The shielding design of the fuel transfer tube is based on the 100 hr decayed spent fuel source strengths provided in Table 12.2-9. The gamma source strengths given in units of [MeV/W-sec] are converted to [photons/sec] by multiplying the gamma source strength values by the thermal power per fuel assembly in [W] and dividing by the source energy in [MeV]. Then, the shielding source term is determined by multiplying this calculated value by the radial power peaking factor of 1.55

and by the number of fuel assemblies transferred through the transfer tube, which is two (2). It should be noted that the rod radial power factor of 1.55 used for the shielding analysis is the ratio of the average power per unit length, while the peak assembly power of 1.2353 is the maximum assembly power during the first cycle for steady-state.

The spent fuel pool is designed to comply with the following shielding design criteria: (a) to provide sufficient shielding so as to meet the radiation zoning requirements adjacent to the SFP and (b) to maintain the dose rate to the spent fuel handling operator less than 0.025 mSv/hr as specified in ANSI/ANS 57.1-1992.

In determining the shielding requirements around the SFP, the source terms for the spent fuel assemblies seated in the SFP storage racks and for the SFP water are considered. To add conservatism to the design, the SFP is assumed to be filled with 1,696 spent fuel assemblies decayed for 100 hours, for which the source term is provided in Table 12.2-9. The radial power peaking factor is not considered. The source region is assumed to be homogeneously mixed with UO_2, Zircalory-4 and H_2O in the shielding calculations, while the shielding effects for the SFP storage racks and the SFP liner are not taken into account. For the source terms in the SFP water, the design basis specific activities provided in Table 12.2-17 are used. By using this conservative approach in the shielding calculations, it is ensured that the minimum shielding requirements around the SFP provided in Table 12.3-5 meet the radiation zoning designations presented in Figures 12.3-3 through 12.3-5.

The dose rate to an operator on the refueling platform is determined by the dose rate value corresponding to the axial distance (or water depth) from top of active fuel to the refueling pool water level after 100 hour decay in Figure 12.2-1. Since mechanical stop in both the refueling machine and spent fuel handling machine restrict withdrawal of the spent fuel assemblies above the minimum safe water cover depth of 274.32 cm (9 ft) from top of active fuel, an operator on the refueling platform will not be exposed to the radiation dose limits in the work area when the shielding of the fuel handling equipment is taken into account.

For shielding of the cask loading pit, it is assumed that a single spent fuel assembly is being loaded into the cask within the cask loading pit. The source term of the spent fuel assembly is conservatively assumed to be decayed for 1,000 hours after reactor shutdown. This source term is also provided in Table 12.2-9 and the gamma source strengths used for the cask loading pit shielding are derived using the same methods as those for the fuel transfer tube shielding except that the number of fuel assemblies is one.

Typically, pipe chases do not need to be accessed frequently. The APR1400 design minimizes locating components such as valves in pipe chases to minimize plant personnel access to pipe chases and to reduce the potential for radiation exposure. When access is needed, radiation protection personnel conduct a survey of the area to determine the strength and location of radiation sources within the pipe chase. Temporary shielding is used to minimize personnel exposure. If the primary source of radiation in the pipe chase is spent resin or slurry transfer piping, precautions are taken by operating personnel to provide reasonable assurance that no spent resin is transferred while personnel are in the pipe chase. The resin transfer lines are also provided with a flushing capability to minimize the potential for hot spots in the piping.

Shielding thicknesses and radiation zones in the pipe chases and valve rooms are determined based on the number and characteristics of the pipes in the corresponding areas. The dose rates for different kinds of pipes are determined using the corresponding source terms in each pipe. The length of each pipe is assumed to be 20 feet because the dose rate at 1 foot away from the pipe is maximized when the pipe is 20 feet long. If there are multiple number of pipes in a pipe chase or a valve room, the resultant dose rate in the cubicle and the required minimum shield thicknesses are determined by summing the dose contributions of each pipe.

Two layers of shield plugs are installed around the reactor vessel. The shield plugs are made of ordinary concrete and are designed to reduce the neutron and gamma streaming from the reactor as low as possible such that the dose rate on the operating floor is minimized. Adequate air cooling is provided to the area between the reactor and the shield plugs and the primary biological shield to ensure that the heat generated from the reactor does not cause any functional deterioration of the concrete with respect to the shielding and structural integrity.

The APR1400 ICI design adopts the bottom mounted In-Core Instrument system that has only fixed type detectors for normal operation. During refueling operation, all ICIs are withdrawn from the core up to the full length of 24 feet when the reactor area is flooded with refueling water. The highly irradiated portion of the ICIs still remains inside the ICI guide tubes which are filled with refueling water at the lower head area of the reactor vessel. The withdrawal and disposal operation of ICI is performed remotely and the operators are shielded by the pool of water as well as the distance between the reactor vessel and the CEACP; and administrative procedures are used to monitor radiation level to insure personnel safety and radiation is ALARA.

Components that handle a significant amount of radioactive materials, such as LWMS floor drain tanks and equipment waste tanks, are located in shielded cubicles separated from the pump and valve galleries that are provided with labyrinths for access to the galleries. This design approach minimizes radiation streaming and scattering but permits inspection and maintenance access and removal of smaller items such as pumps, valves, and instruments for repair in lower-radiation areas. This design approach meets the requirements of NRC RG 8.8 2.b(4). The doors or hatches being relied on to maintain doses within the radiation zone designations provided in the Chapter 12 radiation zone figures are as follows:

a. Personnel Air Lock between Containment Annulus Area (100-C01) and Personnel Air Lock Entrance (100-A14A)

b. Personnel Air Lock between Operating Area (156-C01) and Containment Entrance Area (156-A04B)

c. Equipment Hatch between Operating Area (156-C01) and Equipment Hatch Access Room (156-A10A)

d. Door between Equipment Hatch Access Room (156-A10A) and the building exterior

e. Doors between Truck Bay (100-P08) and the building exterior

The COL applicant is to provide the material composition and shielding properties of these doors/hatches, and these thicknesses equivalent to the minimum required concrete shield thicknesses. Also, the COL applicant is to provide the service life of these doors/hatches and perform periodic inservice inspection and maintenance for these doors/hatches to provide reasonable assurance of functionality throughout the life of the plant (COL 12.3(3)). The plant shielding is designed not only to maintain personnel occupational exposure ALARA, but also to maintain exposure to the general public ALARA.

The APR1400 shielding design has target dose rates that are below the limits for radiation zone designations provided in Table 12.3-3 to provide a sufficient margin in maintaining radiation exposure to plant personnel and the public ALARA. COL applicant is to provide information to ensure that radiation levels at the site boundary not exceed the limits of 40 CFR part 190, from all radiation sources, including the outdoor tanks (COL 12.3(4))

12.3.2.4 Access Control to High Radiation and Very High Radiation Areas

The high radiation and very high radiation areas, areas potentially greater than 1 Gy/hr and 5Gy/hr, respectively, as identified in Table 12.3-6, which are located in the containment building have multiple features of access control to prevent inadvertent radiation exposure to plant personnel. These very high radiation areas include the ICI cavity, the hold-up volume tank, the core debris chamber, the reactor cavity, the steam generator cavity, and the reactor drain tank room. Access to the containment building is strictly controlled and built-in design features to prevent inadvertent access include a secure air lock as the only point of entry for personnel, the door to which is locked and equipped with a security alarm. In addition to the access control provided at the point of entry into the containment building, separate barriers with individual locked doors are provided for each of these very high radiation areas in accordance with the guidance of NRC RG 8.38 (Reference 18).

The high radiation areas on Elevations 78′ and 86′ of the auxiliary building are located within a block where thick concrete walls are provided as shielding to the surrounding areas. There are no doors provided to allow access to the high radiation cubicles within this block. These cubicles include the pre-holdup ion exchanger pit, the purification ion exchanger pit, the purification filter pit, and the filter area. This block of filters and ion exchangers can only be accessed from the elevation 100′ level via manway, which are locked at all times and are further under administrative controls to prevent unauthorized access. Also on Elevations 100′ and 120′ of the auxiliary building is the volume control tank cubicle, which is a potentially high radiation area. This cubicle, which is not normally accessed by personnel, is locked and can only be opened by key from the outside.

The areas listed in Table 12.3-6 at the Elevation 120′ level of the auxiliary building, which are high radiation areas during refueling operations, include the transfer tube inspection area, the cask loading pit, the refueling canal, and the spent fuel pool. The cask loading pit and refueling canal, and the spent fuel pool do not allow for inadvertent personnel access as these areas do not have an entrance for personal entry, and since the transfer tube access area is locked normally, the transfer tube inspection area cannot be accessed.

The areas listed in Table 12.3-6 as high and very high radiation areas within the compound building are all provided with access control in the form of locked doors. These rooms are provided with a latch bolt operated by key from the outside or by a rotating inside knob/lever. The two exceptions to this form of access control are the hot pipe way on Elevation 77′ and the charcoal delay bed room. The hot pipe way and the charcoal delay bed room are not provided with a door for personnel access. The only accesses to these areas are via the hatches provided on Elevation 85′ and 120′, respectively. Since these hatches are intended for maintenances or equipment removal, and are equipped with heavy concrete blocks, unauthorized access is not possible.

12.3.3 Ventilation

The APR1400 design provides shielding for ventilation components including cubicle ventilation for cubicles that contain components with radiological contamination, filters in the HVAC air cleaning units (ACUs) that remove radiological contaminants from the air streams before release, and ducts carrying radiologically contaminated vent streams to the maximum extent practicable. The shielding design for the ventilation exhaust treatment components are determined using adjusted design basis airborne radiological source terms presented in Table 11.3-1 by multiplication factors derived from the ratio of the activity increase of each nuclide between design basis and expected source terms. The source for each of the ACU filters is calculated based on the buildup of the removed radionuclides during its period of operation up to one year. The dose rates determined the individual filters are summed to determine the overall dose rate for the ACU(s) for determination of radiation zoning and the minimum shield wall thicknesses required. HVAC duct chases are provided with structural walls of 18-inch concrete walls in the auxiliary building and compound building, for shielding purpose.

The spread of airborne contamination within the plant is minimized by the design of the plant HVAC systems to provide airflow from areas of lower potential for airborne contamination to areas of greater potential for airborne contamination. For building compartments with the potential for contamination, the exhaust from the areas is designed with pressure and flow balances to minimize the amount of uncontrolled exfiltration from these areas. These design features provide reasonable assurance that the average concentration of radioactive material in the air in the areas that are normally occupied is less than the small fraction of DAC prescribed in 10 CFR Part 20 Appendix B. Therefore, personnel exposure due to inhalation of and contact with airborne contamination is maintained ALARA.

Airborne radiation monitoring is provided for areas that are normally occupied and have a significant potential for airborne contamination. The monitors can detect the time-integrated change of the airborne radioactivity within 10 DAC-hours for the most limiting particulate and iodine species in each area.

Airborne radiation monitors are described further in Section 11.5. The locations of the process effluent radiation monitors are shown in Figure 11.5-2. The airborne radiation monitors are located upstream of the filters within the HVAC ventilation systems.

HVAC systems are described in Section 9.4.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation monitoring system (ARMS) supplements the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and provides reasonable assurance of conformance with the personnel radiation protection requirements of 10 CFR Part 20, 10 CFR Part 50, 10 CFR Part 70 (Reference 19); the guidelines of NRC RGs 1.21 (Reference 20), 1.97, 8.2 (Reference 21), 8.25 (Reference 5), and 8.8 (Reference 1); and American National Standards Institute (ANSI) N13.1 (Reference 22) and Institute of Electrical and Electronics Engineers (IEEE) Std. 497 (Reference 23). The ARMS is in conformance with ANSI/ANS HPSSC-6.8.1 (Reference 24).

The process and effluent radiation monitoring system and sampling systems are described in Section 11.5.

Portable instruments are used and the associated training and procedures are provided to accurately determine the airborne iodine concentration in areas within the facility where plant personnel could be present during an accident in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) (Reference 11) and the criteria in Item III.D.3.3 of NUREG-0737. Portable instruments are also used as needed during normal operation in accordance with the guidelines of NRC RG 8.8. The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737 as well as the guidelines of NRC RG 8.8 (COL 12.3(5)).

With regard to the criticality accident monitoring, the requirements in 10 CFR 50.68(b) (Reference 25) are followed to prevent criticality as described in Subsection 9.1.1.

12.3.4.1 Area Radiation Monitoring System

12.3.4.1.1 Design Objective

The ARMS monitors the radiation levels in selected areas throughout the plant. Most area monitors are designed to warn operators and station personnel through visible and audible alarms when unusual radiological events occur. Some area monitors are designed to monitor the post-accident radiation level in areas where access to equipment that is important to safety may be necessary. These post-accident radiation monitors, shown in Section 7.5 and Table 7.5-1, are designed in accordance with NRC RG 1.97, Rev. 4 and NRC RG 1.206 (Reference 26).

Area radiation monitors have local visible and audible alarms. An additional visual indication lamp may be provided in high-noise areas if needed to provide reasonable assurance of prompt recognition by nearby personnel during high-radiation conditions.

12.3.4.1.2 Location of Area Radiation Monitors

Area radiation monitoring equipment is used to alert operators and station personnel of abnormally high-radiation conditions in an area and protect personnel from possible overexposure. The locations of the area monitors are based on the potential for significant radiation levels in an area. Area monitors are also located in areas where accident access to safety-related equipment may be required during post-accident conditions. Area radiation monitors are also used in special process applications. For example, area radiation monitors located next to the main steam lines are used to monitor for an SG tube leak or rupture and high-range area detectors are used to estimate the accident containment airborne activity and primary coolant activity.

12.3.4.1.3 General System Description

Area radiation monitors consist of electronics/displays (RT) and Geiger-Mueller (G-M) tubes or ionization chambers for detecting gamma radiation (RE). Each monitor using RT may be configured with one or more detectors and may cover multiple areas. Selection of detectors is based on the range needed for the particular monitoring application. Some areas may require extended or high-range detector configurations to cover special operational or post-accident monitoring functions.

Radiation level signals and alarms and operation status alarms are generated by each RT for local alarm capability and transmittal to the information processing system (IPS), qualified indication and alarm system (QIAS), and other interfacing systems. The signals and alarms are recorded and can be retrieved by the operators using IPS. Radiation level signals are transmitted from the RT via digital communication ports and analog outputs. Alarm relay contacts are provided to actuate alarms for radiation, high radiation, and operation status.

Table 12.3-7 lists the area radiation monitors and specifies the range, electrical class, and seismic category of each monitor.

Area monitors are located based on the expected frequency of access, occupancy time, and expected and potential radiation levels in plant work areas.

a. Areas that are typically high-radiation areas but require little or no access (e.g., pipe chases) are not provided with an area radiation monitor.

b. Areas that typically have a high frequency of access and are normally low-radiation areas, but are potentially high-radiation areas, are provided with an area radiation monitor. In accident conditions, areas such as corridors outside personnel and equipment hatches may become very-high-radiation areas and are therefore provided with an area radiation monitor to provide reasonable assurance of worker safety.

   During accident conditions, plant personnel are evacuated from the radiation control area. A radiation protection technician then conducts analyses using portable radiation monitors to determine the optimal routes to vital areas through the radiation control area so that personnel exposure is minimized. The radiation protection technician typically escorts maintenance personnel and operators into the radiation control area and continues to monitor radiation levels using portable radiation monitors. The area monitors located in the plant provide an audible and visual alarm if high radiation levels are detected.

12.3.4.1.4 Redundancy, Diversity, and Independence

The ARMS, which performs safety functions, is designed to meet the single-failure criteria, separation criteria, segregation criteria, and environmental and seismic qualification requirements in accordance with the requirements of IEEE Std. 323 (Reference 28), IEEE Std. 344 (Reference 29), and IEEE Std. 603 (Reference 30). The balance-of-plant engineered safety features actuation system (BOP ESFAS) consists of the ARMS and the process and effluent radiation monitoring system described in Section 11.5. The safety evaluation for the ESF system of these monitors is described in Section 7.3.

12.3.4.1.5 ARMS

The ARMS monitors presented in Table 12.3-7 are categorized into subsystems used for process monitoring functions or other monitoring applications. The location and flow diagrams of the monitors are shown in Figure 11.5-1.

The ARMS monitors are as follows:

a. Safety-related area monitors (RE-231A, 232B, 241A, 242B, 233A, and 234B)

   Containment operating area monitors (RE-231A, 232B) and spent fuel pool area monitors (RE-241A, 242B) are installed as safety-related area monitors for actuating engineered safety features. These monitors perform additional safety functions that generate containment purge isolation and fuel area emergency ventilation actuating signals. The fuel area normal ventilation system is isolated, and the emergency ventilation system is initiated by the fuel area emergency ventilation actuating signal. Containment purge isolation actuating signal (CPIAS) isolates the containment purge system. These monitors are accident monitoring instrumentation (AMI), type C, and also listed on Table 7.5-1.

   The containment upper operating area monitors (RE-233A, 234B) consist of physically independent and electrically separated detectors located inside the containment away from the influence of the reactor coolant system to measure high-range gamma radiation. This monitor gives operators a seismically and environmentally qualified indication of containment airborne activity. These monitors conform with the requirements of 10 CFR 50.34(f)(2)(xvii) and the criteria in Attachment 3 to Item II.F.1 of NUREG-0737 and NRC RG 1.97.

   The monitors (RE-233A, RE-234B) transmit the radiation signals to the licensing entity via emergency response data system (ERDS) link.

   One of the spent fuel pool area monitors is located on a wall and the other near the SFP bridge area.

   The containment operating area monitors (RE-231A and 232B) are located near El. 160' directly above the refueling pool. The containment upper operating area monitors (RE-233A and 234B) are azimuthally 180° apart and located just below the containment polar crane rail support girder (near El. 230'). Thus, the two monitors have a wide open, unobstructed view of the entire containment free air volume. The location of RE-231A, RE-232B, RE- 233A, and RE-234B is shown in Figure 11.5-2A.

The location of RT-231A and RT-233A is shown in Figure 11.5-2BB.

The location of RT-232B and RT-234B is shown in Figure 11.5-2M.

The location of RE/RT-241A and RE/RT-242B is shown in Figure 11.5-2O.

b. Non-safety-related area monitors (RE-205, 235, 236, 237, 238, 245, 257, 275, 279, 284, 285, 288, 289, 292, and 293)

A post-accident primary sample room area monitor (RE-205) is provided in the auxiliary building and a normal primary sample room area monitor (RE-285) is provided in the compound building.

Two area monitors (RE-237, 238) are provided in the main steam/feedwater piping penetration area and one area monitor (RE-236) is provided in the containment personnel access hatch area.

An area monitor is provided for the main control room (RE-275) and technical support center (RE-279) to measure gross gamma dose rates.

Other areas where an area monitor is provided are the in-core instrument area (RE-235), hot machine shop area (RE-293), new fuel storage area (RE-245), radiochemistry lab area (RE-257), waste drum storage area (RE-292), compound building truck bay area (RE-288, 289), and dry active waste storage area (RE-284) in compound building.

The location of RE/RT-205 is shown in Figure 11.5-2D.

The in-core instrument (ICI) area monitor, RE-235, is a radiation monitor for the ICI seal table area. The seal table is located at Elevation 130’-0’’ near the integrated head assembly lift rig. The monitor is located above the water level (concrete floor level) of 156’-0’’ and the monitor is at 6’-6” off the 156’ level because the seal table area is flooded and is submerged under water during refueling operations. RE-235 is located directly overlooking the seal table. Containment personnel access hatch area monitor, RE-236 is located on the containment wall near the personnel access hatch at Elevation 160’-6”. The location of RE-235 and RE-236 is shown in Figure 11.5-2A.

The location of RT-235, RT-236, and RE/RT-245 is shown in Figure 11.5-2O.

The location of RE/RT-237 is shown in Figure 11.5-2L.

The location of RE/RT-238 is shown in Figure 11.5-2K.

The location of RE/RT-257 and RE/RT-285 is shown in Figure 11.5-2S.

The location of RE/RT-275 and RE/RT-279 is shown in Figure 11.5-2N.

The location of RE/RT-284, RE-288, RT-288, RE-289, RT-289, RE-292, and RT-292 is shown in Figure 11.5-2T.

The location of RE-293 and RT-293 is shown in Figure 11.5-2U.

c. Alarm location and function of the area monitors

The RT of the area radiation monitor is equipped with alarm components such as beacon light (RL) as a visual alert and a horn alarm (RAH) as an audible alarm.

For the containment operating area and the containment upper operating area monitors (RE-231A, 232B, 233A, and 234B), the local alarms are located at the locations of the RE and at the location of the RT.

Common combined RL/RAHs for RE-231A, 232B, 233A, and 234B are located in the containment operating area (RL/RAH-1) and another one (RL/RAH-2) outside the containment building near the personnel access hatch as shown in Figures 11.5-2A and 11.5-2O. RL/RAH-1 and RL/RAH-2 are in addition to an individual RL/RAH for each RT (RT-231A, 232B, 233A, and 234B) in the Auxiliary Building as shown in Figures 11.5-2M and 11.5-2BB. The common inboard alarm alerts the operation and maintenance crew in the containment operating area to exit the containment and also warns the crew outside to limit the access to the containment building.

For the in-core instrument area monitor (RE-235), two local alarms are provided, one located at the location of the RE and the other one at the location of the RT outside the personnel access hatch as shown in Figures 11.5-2A and 11.5-2O. The local alarm in the in-core instrument seal table area alerts the crew to exit the area and also warns the crew outside to limit access to the area inside containment.

For the containment personnel access hatch area monitor (RE-236), the local alarm is located at the location of the RT outside the personnel access hatch as shown in Figures 11.5-2A and 11.5-2O. The local alarm of the RT alerts the crew outside about high radiation condition inside the personnel access hatch. This alarm at the outboard side of the access hatch alerts the plant crew prior to entering the containment through the access hatch. Since the containment evacuation alarms provided by RL/RAH-1 from RE-231A, 232B, 233A and 234B, a local alarm near RE-236 is not necessary.

For the hot machine shop area monitor (RE-293), the local alarm is located at the location of the RE and at the location of the RT as shown in Figure 11.5-2U.

For the truck bay area monitor (RE-288 and RE-289) and the waste drum storage area monitor (RE-292), the local alarm is located at the location of the RE and at the location of the RT as shown in Figure 11.5-2T.

For the Post-accident primary sample room area monitor (RE-205), the Main steam and FW containment piping penetration area monitors (RE-237 and RE-238), the Spent fuel pool area monitor (RE-241A and RE-242B), the New fuel storage area monitor (RE-245), the Radiochemistry lab area monitor (RE-257), the Main control room area monitor (RE- 275), the TSC area monitor (RE-279), the Compound building dry active waste storage area monitor (RE-284), and the Normal primary sample room area monitor (RE-285), each associated local alarm is located at each location of the RT which is installed adjacent to the RE. The local alarm alerts the crew inside about high radiation condition in the area. Some of these monitors are located in areas where the areas are continuously manned and occupied where radioactive materials are handled, stored or processed. Some are located in an open area where distinguishing the inside or outside of an area is immaterial. All these areas, therefore, would not need to distinguish the outside/inside of an area for the purpose of warning the crews before entry into the area.

12.3.4.1.6 Range and Alarm Setpoints

The ranges of the ARMS are shown on Table 12.3-7. Alarm setpoints for safety-related monitors are determined by plant procedures and the Offsite Dose Calculation Manual (ODCM). The setpoint methodology includes the relationship between the analytical limit, setpoint, and channel uncertainty. The setpoint methodology also provides channel uncertainty calculations associated with the setpoints used for ESF actuation functions. The setpoint methodology follows the methodology in ANSI/ISA-67.04 (Reference 27). The COL applicant is to determine the ARM setpoints for WARN, ALARM, and the containment purge isolation and fuel handling area emergency ventilation actuation signals, based on the site-specific conditions and operational requirements (COL 12.3(6)).

12.3.4.1.7 Calibration Methods and Frequency

The methodology to determine the calibration methods and frequency of the ARMS is provided by the ODCM based on plant procedures.

12.3.4.1.8 Power Supplies

Instrument loops of safety-related monitors are powered from the appropriate train of Class 1E 120 AC distribution panel in the instrument and control power system (IP), which is powered by the onsite Class 1E emergency diesel generator. When the emergency diesel generator restores power to the skid, skid equipment such as sample pumps returns to the original operating status without having to be manually restarted. The TSC area radiation monitor, which is non-safety-related, is powered from permanent non-safety buses that are backed up by an alternate ac generator. Instrumentation and control power are described further in Subsection 8.3.2.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

Airborne radioactivity monitors are installed in selected areas and HVAC systems to provide plant operating personnel with continuous information on the airborne radioactivity levels throughout the plant. These monitors, consisting of gaseous process and effluent radiation monitors (PERMSS), are described in Section 11.5 and listed in Table 11.5-1. The airborne radioactivity monitors are as follows:

a. High-energy line break area exhaust ACU inlet monitor

b. Auxiliary building controlled area (I, II) HVAC normal/emergency exhaust ACU inlet monitor

c. Containment purge effluent monitor

d. Containment air monitor

e. Fuel handling area HVAC effluent monitor

f. Condenser vacuum pump vent effluent

g. Main control room air intake monitor

h. Compound building HVAC effluent monitor

i. Miscellaneous process monitors

12.3.4.2.1 Design Objectives

The objectives of the airborne radioactivity monitors are presented in Subsection 11.5.1.1.

12.3.4.2.2 Location of Airborne Radioactivity Monitors

The criteria for the location of the airborne radioactivity monitors are presented in Subsection 11.5.1.1 and the monitor locations are shown in Figure 11.5-2.

12.3.4.2.3 System Description

Airborne radioactivity monitors and applicable design criteria are described in Subsection 11.5.1.2.

12.3.5 Dose Assessment

The dose assessment is described in Section 12.4.

12.3.6 Combined License Information

COL 12.3(1) The COL applicant is to determine the areas that will require either electro or mechanical polishing.

COL 12.3(2) The COL applicant is to establish how the water chemistry pH control reduces radiation fields.

COL 12.3(3) The COL applicant is to provide the material composition and shielding properties of the following doors/hatches, and these thicknesses equivalent to the minimum required concrete shield thicknesses.

- Personnel Air Lock between Containment Annulus Area (100-C01) and Personnel Air Lock Entrance (100-A14A)
- Personnel Air Lock between Operating Area (156-C01) and Containment Entrance Area (156-A04B)
- Equipment Hatch between Operating Area (156-C01) and Equipment Hatch Access Room (156-A10A)
- Door between Equipment Hatch Access Room (156-A10A) and the building exterior
- Transfer tube access area manway hatch in Room (137-A40B) at elevation 137'-6"
- Doors between Truck Bay (100-P08) and the building exterior

Also, the COL applicant is to provide an ITAAC for the radiation shielding and the service life of these doors/hatches; and perform periodic in-service inspection and maintenance for these doors/hatches to provide reasonable assurance of functionality throughout the life of the plant.

COL 12.3(4) COL applicant is to provide information to ensure that radiation levels at the site boundary not exceed the limits of 40 CFR Part 190, from all radiation sources, including the outdoor tanks.

COL 12.3(5) The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737 as well as the guidelines of NRC RG 8.8.

COL 12.3(6) The COL applicant is to determine the ARM setpoints for WARN, ALARM, and the containment purge isolation and fuel handling area emergency ventilation actuation signals, based on the site-specific conditions and operational requirements.

COL 12.3(7) The COL applicant is to determine the purpose and use of the room (063-P73). The access control for this room shall be changed accordingly in compliance with the guidance in NRC RG 8.38. In addition, COL applicant is to specify any necessary radiation monitoring requirements, and any additional necessary design features and controls to ensure compliance with applicable regulations, including 10 CFR Part 20 and 10 CFR Part 36.

12.3.7 References

1. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be ALARA," Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

2. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," Rev. 1-R, U.S. Nuclear Regulatory Commission, May 1977.

3. 10 CFR Part 20, "Standards for Protection against Radiation," U.S. Nuclear Regulatory Commission.

4. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

5. Regulatory Guide 8.25, "Air Sampling in the Workplace," Rev. 1, U.S. Nuclear Regulatory Commission, June 1992.

6. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission.

7. NUREG-0737, "Clarification of TMI Action Plan Requirements" U.S. Nuclear Regulatory Commission, November 1980.

8. Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Rev. 4, U.S. Nuclear Regulatory Commission, June 2006.

9. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, July 2000.

10. Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, May 2009.

11. 10 CFR 50.34, "Contents of Applications, Technical Information," U.S. Nuclear Regulatory Commission.

12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, various dates and revisions.

13. ANISN, "A One Dimensional Discrete-Ordinates Transport Code with Anisotropic Scattering," Union Carbide Corporation, 1967.

14. RSICC Computer Code Collection CCC-740 MCNPX, "Monte Carlo N-Particle Transport Code System," Los Alamos National Laboratory, 2006.

15. Grove Software, Inc., MICROSHIELD User's Manual Version 9.05, 2011.

16. RUNT-G, "A PC Version of RUNT-G," KEPCO E&C, 1988.

17. ISOSHLD, "A Computer Code for General Purpose Isotope Shielding Analysis," BNWL-236, Pacific Northwest Laboratory, Richmond, Washington, 1966.

18. Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, May 2006.

19. 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," U.S. Nuclear Regulatory Commission.

20. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," Rev. 2, U.S. Nuclear Regulatory Commission, June 2009.

21. Regulatory Guide 8.2, “Guide for Administrative Practices in Radiation Monitoring Measuring,” Rev. 1, U.S. Nuclear Regulatory Commission, May 2011.

22. ANSI/HPS N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substance From the Stack and Ducts of Nuclear Facilities,” Health Physics Society, January 1999.

23. IEEE Std. 497-2002, “Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2002.

24. ANSI/ANS HPSSC 6.8.1, “Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors,” American Nuclear Society, May 1981.

25. 10 CFR 50.68, “Criticality Accident Requirements,” U.S. Nuclear Regulatory Commission.

26. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, June 2007.

27. ANSI/ISA 67.04-1994, “Setpoints for Nuclear Safety-Related Instrumentation,” International Society of Automation, 1994.

28. IEEE Std. 323-2003, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2003.

29. IEEE Std. 344-2004, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2005.

30. IEEE Std. 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 1991.

31. 10 CFR 52.47, “Contents of Applications; Technical Information,” U.S. Nuclear Regulatory Commission.

Table 12.3-1

Cobalt Contents of Finished Surface for Primary System Materials

Component	Cobalt Content
Reactor Vessel Assembly Cladding(1)	0.05 w/o (max)
Reactor Coolant Pipe Cladding(1)	0.05 w/o (max)
Pressurizer Assembly Cladding(1)	0.05 w/o (max)
Steam Generator Cladding(1)	0.05 w/o (max)
Reactor Coolant Pump Cladding(1)	0.05 w/o (max)
Reactor Coolant Pump Impeller(6)	0.05 w/o (max)
Core Support Barrel Assembly(2)	0.05 w/o (max)
Lower Support Structure Assembly(2)	0.05 w/o (max)
Instrument Nozzle Assembly(2)	0.05 w/o (max)
Upper Guide Structure Assembly(2)	0.05 w/o (max)
Inner Barrel Assembly(2)	0.05 w/o (max)
Core Shroud Assembly(2)	0.05 w/o (max)
Surge Line(4)	0.05 w/o (max)
Control Element Driving Mechanisms including RV CEDM nozzles(2),(3)	0.05 w/o (max)
Steam Generator Tube(3)	0.015 w/o (mean value)
The components except where identified in this table	Not Limited

(1) Austenitic Stainless Steel Type 308, or Type 309, or Ni based Alloy (Alloy 690 equivalent weld material)

(2) Austenitic Stainless Steel Type 304

(3) Alloy 690TT

(4) Austenitic Stainless Steel Type 347

(5) Small quantity of cobalt base alloy (Stellite or Haynes alloy) as bar, casting, or hardfacing is used for the CEDMs, RVIs, pumps, or valves.

(6) Martensitic Stainless Steel F6MN or equivalent

Table 12.3-2

Cobalt Contents of Fuel Assembly Components Materials

Components	Cobalt Content
Top Nozzle except Holddown Spring(1)	0.05 w/o (max)
Top Nozzle Holddown Spring(2)	0.10 w/o (max)
Bottom Nozzle(1)	0.05 w/o (max)
Top, Bottom and Protective Grids(2)	0.04 w/o (max)
Top and Bottom Grid Sleeves(1)	0.12 w/o (max)
Guide Thimble Screw(1)	0.12 w/o (max)

(1) Austenitic Stainless Steel Type 304/304L

(2) Nickel Alloy 718

(3) Small surface components compared to Fuel Rod

Table 12.3-3

Normal Operation Radiation Zone Designations

Zone Designation(1)	Design Dose Rate (mSv/hr)(2)	Zone Description
1	DR ≤ 0.0025	Uncontrolled, unlimited access
2	0.0025 < DR ≤ 0.025	Controlled, limited access, 40 hr/wk
3	0.025 < DR ≤ 0.05	Controlled, limited access, 20 ~ 40 hr/wk
4	0.05 < DR ≤ 0.2	Controlled, limited access, 5 ~ 20 hr/wk
5	0.2 < DR ≤ 1	Controlled, limited access, 1 ~ 5 hr/wk
6	1 < DR ≤ 10	Controlled access under supervision of radiation protection personnel
7	10 < DR ≤ 5,000	Controlled access under supervision of radiation protection personnel
8	DR > 5,000	Controlled access under supervision of radiation protection personnel

(1) High-radiation areas include radiation Zones 6, 7, and 8.

(2) DR denotes dose rate at 30 cm (11.8 in.) from the radiation source or from any surface that the radiation penetrates.

Table 12.3-4

Post-Accident Radiation Zone Designations and Design Dose Rates

Zone Designation	Design Dose Rate (mSv/hr)
1	DR ≤ 0.15
2	0.15 < DR ≤ 1
3	1 < DR ≤ 10
4	10 < DR ≤ 100
5	100 < DR ≤ 1,000
6	1,000 < DR ≤ 5,000
7	DR > 5,000

Table 12.3-5 (1 of 16)

Design Basis Radiation Shield Thicknesses

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Containment Building							
069-C01	ICI Cavity	-	60	79	60	Ground	-
078-C01	Reactor Cavity	80	80	79	79	Ground	-
080-C01	Holdup Volume Tank	36	36	-	79	Ground	-
100-C02	Steam Generator Cavity	36	36	30	36	Ground	-
100-C03	Reactor Drain Tank	24	24	48	24	-	24
100-C04	Letdown HX Room	24	24	48	24	-	24
128-C01	Regenerative HX Room	60	60	48	48	24	48
136-C02	Pressurizer Cavity	33	36	30	33	30	-
Auxiliary Building							
050-A01C	CS Pump & Miniflow HX Room	48	48	48	24	Ground	30
050-A01D	CS Pump & Miniflow HX Room	48	48	48	24	Ground	30
050-A02C	SI Pump Room	48	48	48	48	Ground	30
050-A02D	SI Pump Room	48	48	48	48	Ground	30
050-A03A	SI Pump Room	48	48	24	48	Ground	30
050-A03B	SI Pump Room	48	48	24	48	Ground	30
050-A04A	SC Pump and Miniflow HX Room	48	48	27	24	Ground	30
050-A04B	SC Pump and Miniflow HX Room	48	48	27	24	Ground	30
055-A01C	CS HX Room	36	48	48	30	Ground	30
055-A01D	CS HX Room	48	36	48	30	Ground	30
055-A08C	Floor Drain Sump Pump Room	14	10	14	10	Ground	14
055-A08D	Floor Drain Sump Pump Room	10	14	14	10	Ground	14
055-A10C	Tendon Gallery Entrance	24	36	48	48	Ground	24
055-A14C	Pipe Chase and Valve Room	48	48	48	48	Ground	30
055-A14D	Pipe Chase and Valve Room	48	48	48	48	Ground	30
055-A18A	Pipe Chase and Valve Room	42	48	48	20	Ground	30
055-A18B	Pipe chase and Valve Room	48	42	48	10	Ground	30
055-A21A	Pipe Chase and Valve Room	10	48	25	29	Ground	32
055-A21B	Pipe Chase and Valve Room	48	10	20	10	Ground	32

Table 12.3-5 (2 of 16)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)							
055-A22A	Pipe Chase	10	10	33	42	Ground	10
055-A22B	Pipe Chase	10	10	48	42	Ground	10
055-A30A	SC HX Room	36	48	48	48	Ground	30
055-A30B	SC HX Room	48	36	48	48	Ground	30
055-A31B	Chemical Drain Sump Pump Room	27	10	10	10	Ground	20
055-A33A	Equipment Drain Sump Pump Room	10	22	14	10	Ground	14
055-A33B	Equipment Drain Sump Pump Room	19	10	10	10	Ground	17
055-A34A	Floor Drain Sump Pump Room	10	22	10	22	Ground	11
055-A34B	Floor Drain Sump Pump Room	22	10	10	22	Ground	18
055-A38A	Boronometer Room	25	25	34	25	Ground	28
055-A39A	Process Radiation Monitor Room	19	19	11	19	Ground	21
055-A42A	Charging Pump Room	17	24	24	28	Ground	22
055-A43A	Charging Pump Miniflow HX Room	10	18	10	10	Ground	19
055-A45A	Pipe Chase	27	17	25	10	Ground	24
055-A46A	Condensate Return Unit Room	28	37	10	16	Ground	16
055-A47B	Primary Off-Gas Sample Pump Room	72	48	60	24	Ground	24
055-A49B	Post Accident Sample Control Panel RM	24	72	18	60	Ground	24
055-A51B	Equipment Drain Tank Room	18	36	48	36	Ground	24
055-A52B	Reactor Drain Pump Room	18	18	18	36	Ground	24
055-A53B	Reactor Drain Pump Room	13	12	23	10	Ground	17
055-A54B	Aux. Charging Pump Room	66	24	60	24	Ground	24
055-A55B	Charging Pump Room	65	66	60	60	Ground	24
055-A56A	Valve Room	10	18	28	10	Ground	19
055-A56B	Valve Room	14	20	12	10	Ground	10
055-A58A	Pipe Chase	18	10	26	26	Ground	10
055-A59A	Valve Room	19	10	26	34	Ground	24
055-A62A	Chase	18	48	18	27	Ground	30
055-A62B	Chase	48	18	18	27	Ground	30
068-A06A	Gas Stripper Room	37	35	45	35	30	24

Table 12.3-5 (3 of 16)

Room Number	Room Name		Minimum Required Shield Thickness (inches)					
			North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)								
068-A07A	Hot Pipe Way		13	29	31	29(1)/ 24(2)/ 21(3)	24	24
068-A08B	Hot Pipe Way		33	24	24	20	23	30
068-A09B	Valve Room		19	17	23	19	17	10
068-A10A	Filter and Demin. Valve Area		10	10	33	33	30	24
068-A11A	Filter and Demin. Valve Area		10	24	10	22	24	21
068-A12A	Filter and Demin. Valve Area		10	18	10	18	17	29
077-A01A	Reactor Drain Filter Pit		35	10	10	15	10	23
077-A02A	SFP Cleanup Filter Pit		10	10	10	10	23	23
077-A03A	SFP Demin Filter Pit		10	17	10	18	23	23
077-A04A	SFP Cleanup Filter Pit		10	10	10	10	23	23
077-A05A	SFP Demin. Filter Pit		10	10	10	10	23	23
077-A06A	Purification Filter Pit		10	10	10	10	10	39
077-A07A	Reactor Makeup Water Filter Pit		18	32	18	18	10	10
077-A08A	Purification Filter Pit		10	10	10	10	10	39
077-A09A	SGBD Filter Pit		10	10	10	10	22	23
077-A10A	Seal Injection Filter Pit		10	10	10	11	24	24
077-A11A	SGBD Filter Pit		10	10	10	10	22	23
077-A12A	Seal Injection Filter Pit		10	13	10	10	13	24
077-A13A	SGBD Filter Pit		35	10	10	10	22	23
077-A14A	Boric Acid Filter Pit		13	18	10	29	10	22
077-A15A	Filter Cartridge Storage		32	39	39	39	25	39
078-A09C	HAVC Chase	EL. 100'-0”	18	30	48	48	18	18
		EL. 156'-0”	18	24	48	36	18	18
078-A09D	HAVC Chase	EL. 100'-0”	30	18	48	48	18	18
		EL. 137'-6”	18	18	48	48	18	18
		EL. 156'-0”	24	18	48	36	18	18

(1) West wall within the row line from AD to AF along the column line 24

(2) Wall thickness for labyrinth design

(3) West wall within the column line from 25 to 26 and row line from AA to AD

Table 12.3-5 (4 of 16)

Room Number	Room Name		Minimum Required Shield Thickness (inches)					
			North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)								
078-A12C	Essential Water Chiller RM		36	48	48	48	30	24
078-A12D	Essential Water Chiller RM		48	36	48	48	30	24
078-A13D	Duct RM		36	24	48	48	18	24
078-A14C	Buttress Opng	EL. 100'-0”	18	18	48	48	18	-
		EL. 137'-6”	18	18	48	48	18	-
		EL. 156'-0”	18	18	48	36	18	-
078-A15C	Turbine Driven Aux. Feedwater Pump RM		48	48	48	48	30	48
078-A15D	Turbine Driven Aux. Feedwater Pump RM		48	48	48	48	30	48
078-A20A	Motor Driven Aux Feedwater Pump RM		48	48	48	48	30	48
078-A20B	Motor Driven Aux Feedwater Pump RM		48	48	48	48	30	48
078-A21A	Pipe Chase		10	36	48	48	10	24
078-A21B	Pipe Chase		36	48	48	48	24	24
078-A25A	Class 1E Switchgear 01A RM		36	48	48	30	30	30
078-A25B	Class 1E Switchgear 01B RM		48	36	48	30	30	30
078-A32A	SPF Cleanup Demin. Room		10	22	22	21	12	24
078-A33A	SG Blowdown Demin. Room		33	10	23	10	10	23
078-A34A	Pre-Holdup IX Room		33	15	15	33	15	24
078-A35A	Purification IX Room		12	24	29	43	29	42
078-A36A	Boric Acid IX Room		24	18	18	18	18	18
078-A37A	Deborating IX Room		15	12	10	24	24	24
078-A38A	SFP Cleanup Pump Room		23	23	23	10	23	32
078-A39A	Gas Stripper Effluent Radiation Monitor Room		23	23	23	10	16	35
078-A40B	Boric Acid Concentrator Room		16	23	23	14	16	16
086-A01A	Filter Area		-	-	18	21	13	10
100-A32B	SFP Cooling HX Room		10	10	10	10	10	10
100-A29B	Pipe and HVAC Chase		10	12	10	66	10	10
100-A07C	AUX. Feedwater Tank		48	48	48	48	48	48
100-A07D	AUX. Feedwater Tank		48	48	48	48	48	48

Table 12.3-5 (5 of 16)

Room Number	Room Name		Minimum Required Shield Thickness (inches)					
			North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)								
100-A08C	N1E DC & IP Equipment RM		36	48	48	48	48	18
100-A08D	N1E DC & IP Equipment RM		48	30	48	48	24	18
100-A09C	Tendon Access RM		30	36	48	48	24	18
100-A13A	Mechanical Penetration Room		36	48	48	48	30	18
100-A13B	Mechanical Penetration Room		48	36	48	48	30	18
100-A14A	Personnel Air Lock Enterance		48	36	48	48	48	18
100-A16D	Pipe Chase		48	48	48	48	10	23
100-A16C	Pipe Chase		48	48	48	48	13	10
100-A24A	SFP Cooling HX Room		12	10	12	40	24	10
100-A26A	Valve Room		28	41	21	28	32	10
100-A25A	Volume Control Tank Room		42	42	42	47	48	53
111-A01B	Cask Loading Pit		48	14	48	48	42	-
114-A01B	Spent Fuel Pool		62	60	59	68	71	-
113-A01B	Fuel Transfer Tube Inspection Area		44	44	No walls		62	60
119-A01B	Refueling Canal		60/30(1)	59	62	48	62	-
120-A09C	Electrical Penetration RM (C)		18	48	48	48	18	18
120-A16B	Mechanical Penetration Room		29	27	33	48	18	29
120-A16A	Mechanical Penetration Room		20	24	20	48	17	19
120-A23A	Valve Room		18	25	18	18	10	18
120-A14A	SG Blowdown Regen. HX Room		48	48	48	48/ 27 (2)	48	18
120-A21A	Aux. Bldg Controlled Area (I) Emergency Exhaust ACU RM-1		18	48	-	36	18	18
120-A32A	Aux. Bldg Controlled Area (I) Emergency Exhaust ACU RM-2		18	48	48	-	18	18
137-A19A	SG Blowdown Flash Tank Room		42	42	48	48	48	18
137-A11D	Electrical Penetration RM (D)		48	18	48	48	18	18
137-A31C	Main Steam Valve RM	El. 137'-6″	48	60	48	48/ 84 (3)	48	18
		El. 156'-0″	48	60	48	48	48	18

(1) Minimum required shield thicknesses of the portions in contact with spent fuel pool (114-A01B) and pipe & HVAC chase (100-A29B) are 60 inches and 30 inches, respectively.

(2) Shield thickness of the portions in contact with general access area (120-A20A) & Stair (055-A20A) are 48 inches and 27 inches, respectively.

(3) Shield thickness of the portions in contact with 480V Class 1E MCC 03C room (137-A10C) & main steam enclosure (137-A30C) are 48 inches and 84 inches, respectively.

Table 12.3-5 (6 of 16)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)							
137-A31D	Main Steam Valve RM (El. 156'-0″)	60	48	48	48	48	18
156-A14A	Aux. Bldg Controlled Area (I) Normal Exhaust ACU Room	22	22	22	22	22	22
157-A19C	I&C Equip. RM	18	36	48	36	18	18
157-A19D	I&C Equip. RM	36	18	48	36	18	18
157-A20C	I&C Equip. RM	24	18	48	36	18	18
157-A20D	I&C Equip. RM	18	24	48	36	18	18
174-A15B	Containment High- and Low-volume Purge ACU Room	21	21	21	21	15	10
195-A08B	Aux. Bldg. Controlled Area (II) Normal Exhaust ACU Room	22	22	22	22	18	22

Table 12.3-5 (7 of 16)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
	Compound Building						
063-P01	Hot Pipe Chase	10	28	28	10	Ground	17
063-P02	GRS Header Drain Tank Room	35	42	48	41	Ground	18
063-P03	Valve Room	27	30	37	10	Ground	10
063-P04	GRS Inlet Skid Room	22	34	10	22	Ground	13
063-P05	Spent Resin Long-term Storage Tank Room	27	35	48	36	Ground	46
063-P06	Future Use	36	27	48	36	Ground	39
063-P07	Valve Room	16	29	36	30	Ground	14
063-P08	Low-activity Spent Resin Tank Room	27	32	35	10	Ground	10
063-P09	Valve Room	16	36	10	16	Ground	18
063-P13	Hot Pipe Chase	40	33	40	33	Ground	19
063-P14	Hot Tool Room	15	10	10	10	Ground	32
063-P21	Equip. Waste Pump Room	17	19	10	20	Ground	17
063-P22	Equip. Waste Pump Room	10	17	10	21	Ground	17
063-P23	Equip. Waste Tank Room	13	33	20	22	Ground	27
063-P24	Equip. Waste Tank Room	16	13	21	22	Ground	27
063-P25	Floor Drain Pump Room	14	10	11	19	Ground	10
063-P26	Normal Sump Pump Room	14	14	10	19	Ground	16
063-P27	Chemical Waste Pump Room	10	14	10	15	Ground	16
063-P28	Floor Drain Tank Room	16	16	19	20	Ground	29
063-P29	Floor Drain Tank Room	16	16	19	20	Ground	29
063-P30	Chemical Waste Tank Room	10	16	15	10	Ground	15
063-P31	Chemical Waste Tank Room	10	10	15	10	Ground	15
063-P36	DWS Drain Sump Pump Room	10	10	10	10	Ground	10

Table 12.3-5 (8 of 16)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Compound Building (cont.)							
063-P37	Monitor Tank Room	10	18	11	18	Ground	10
063-P38	PSS-Solidification & Drum Conveyer Room	17	24	24	21	Ground	14
063-P39	Spent Resin Long-term Storage Tank Sump Pump Room	18	20	18	21	Ground	18
063-P40	Concentrate Pump Room	27	24	20	16	Ground	19
063-P41	Concentrate Holding Tank Room	21	27	33	28	Ground	10
063-P42	RO Feed Pump Room	10	10	28	16	Ground	24
063-P43	IX Feed Pump Room	16	10	16	10	Ground	24
063-P44	IX Feed Tank Room	14	16	11	10	Ground	23
063-P47	CTS HEPA Vacuum Skid Room	24	10	21	10	Ground	10
063-P48	CTS Dryer Skid Room	31	24	17	21	Ground	15
063-P49	CTS Vacuum Skid Room	10	10	21	10	Ground	18
063-P54	Monitor Tank Pump Room	10	10	10	10	Ground	14
063-P73	Future Use	36	43	18	48	Ground	36
085-P01	Waste Gas Dryer Skid Room	18	31	31	25	17	24
085-P02	Waste Gas Dryer Skid Room	11	18	22	25	17	24
085-P03	Valve Room	48	11	30	26	18	36
085-P04	Charcoal Guard Bed Room	27	27	35	18	10	24
085-P06	Valve Room	19	26	36	30	19	27
085-P07	Valve Room	27	24	30	30	14	24
085-P08	Valve Room	24	19	22	24	19	24
085-P15	Valve Room	10	21	21	18	10	23
085-P16	Valve Room	10	22	18	19	10	22
085-P17	Valve Room	10	10	10	14	10	17
085-P20	Valve Room	16	16	16	16	10	16
085-P21	Charcoal Guard Bed Room	27	27	18	35	10	24
085-P31	Primary Sampling Room	10	10	10	10	10	10

Table 12.3-5 (9 of 16)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Compound Building (cont.)							
085-P32	Primary Sampling Sink Room	10	13	12	12	14	18
085-P42	IX Module Room	10	30	30	27	14	28
085-P43	IX Module Room	30	10	30	30	14	28
085-P44	RO Feed Tank Room	10	27	32	22	19	25
085-P45	Drum Removal Chase	15	15	15	15	-	25
085-P46	MF Membrane Module Room	23	10	20	15	18	16
085-P47	MF Membrane Module Room	23	16	10	12	15	16
085-P48	RO Membrane Module and Valve Skid Room	43	24	43	34	32	36
096-P01	Charcoal Delay Bed Room	49	47	49	15	36	43
096-P02	Charcoal Delay Bed Room	49	47	15	40	36	43
100-P02	GRS Equipment Removal Area	13	11	40	10	23	10
100-P07	Future Extension Area	24	30	36	37	24	31
100-P08	Truck Bay	24	24	36	37	36	31
100-P09	Waste Drum Storage Area	28	24	36	26	34	31
100-P10	Spent Filter Drum Storage Area	36	28	48	37	36	43
120-P01	Gaseous Radwaste Sample Control Panel Room	16	21	21	12	18	26
120-P02	Gaseous Radwaste Sample Valve Rack Room	22	16	26	18	18	26
139-P06	Normal Exhaust ACU Room	20	20	20	20	20	20
Yard							
-	Boric Acid Storage Tank	16	16	16	16	-	-
-	Holdup Tank	15(1)	15(1)	15(1)	15(1)	-	-

(1) Including the Tank wall of 0.25 inches

Table 12.3-5 (10 of 16)

Room Number	Room Name	Face to			Minimum Required Shield Thickness (inches)
		Structure	Room Number	Room Name	
Additional Information on the Rooms which Contain Complex Geometries					
068-A07A	Hot Pipe Way	Slab	055-A45A	Pipe Chase	10
		Slab	055-A42A	Charging Pump Room	22
		Slab	055-A35A	General Access Area	24
		Slab	055-A39A	Process Radiation Monitor Room	21
		Slab	055-A58A	Pipe Chase	17
		Slab	055-A30A	SC Hx Room	19
		Slab	078-A38A	Spent Fuel Pool Clean up Pump Room	23
		Slab	078-A39A	Gas Stripper Effluents Radiation Monitor Room	16
		Slab	078-A31A	General Access Area	24
068-A10A	Filter and Demin Valve Room	Slab	055-A35A	General Access Area	30
		Slab	055-A36A	CVCS Chemical Package Room	30
		Slab	078-A34A	Pre-Holdup Ion Exchange	15
		Slab	078-A35A	Purification Ion Exchange Room	13
		Slab	078-A37A	Deborating Ion Exchange Room	24
		Slab	077-A01A	Reactor Drain Filter Pit	15
		Slab	077-A06A	Purification Filter Pit	10
		Slab	077-A08A	Purification Filter Pit	10
		Slab	077-A10A	Seal Injection Filter Pit	10

Table 12.3-5 (11 of 16)

Room Number	Room Name	Face to (floor)			Minimum Required Shield Thickness (inches)
		Structure	Room Number	Room Name	
Additional Information on the Rooms which Contain Complex Geometries (cont.)					
077-P01	Hot Pipe Way	Slab	063-P27	Chemical Waste Pump Room	14
		Slab	063-P33	Sampling Room	14
		Slab	063-P34	LRS Control Panel Room	14
		Slab	063-P54	Monitor Tank Pump Room	14
		Slab	063-P56	Drop Area & Opening	14
		Slab	063-P57	Sorting Room	14
		Slab	063-P21	Equip. Waste Pump Room	15
		Slab	063-P22	Equip. Waste Pump Room	17
		Slab	063-P25	Floor Drain Pump Room	10
		Slab	063-P26	Normal Sump Pump Room	16
		Slab	063-P38	PSS-solidification & Drum Conveyor Room	16
		Slab	063-P46	Corridor	25
		Slab	063-P11	Corridor	28
		Slab	063-P09	Valve Room	20
		Slab	063-P40	Concentrate pump Room	20
		Slab	063-P41	Concentrate holding tank Room	10
		Slab	063-P42	RO feed pump Room	24
		Slab	063-P43	IX feed pump Room	24
		Slab	063-P44	IX Feed tank Room	23
		Slab	063-P45	Hot tool Room	23
		Slab	063-P04	GRS inlet skid Room	13
		Slab	063-P07	Valve Room	14
		Slab	063-P10	Hot tool Room	24
		Slab	063-P16	Corridor(1)	18
		Slab	063-P16	Corridor(2)	24

(1) Section within the column line from 33 to 36 and from PF to PG
(2) Section within the column line from 36 to 37 and from PB to PG

Table 12.3-5 (12 of 16)

Room Number	Room Name	Openings for			Minimum Required Shield Thickness (inches)			
		Structure	Room Number	Room Name	North	South	East	West
Additional Information on the Rooms which Contain Complex Geometries (cont.)								
077-P01	Hot Pipe Way	Wall	063-P26	Normal Sump Room	16	16	16	19
		Wall	063-P56	Drop Area	14	14	18	14
		Wall	085-P45	Drum Removal Chase	16	16	16	16
		Wall	063-P16	Corridor(1)	24	-	24	20
		Opening within the column line from 36 to 37 and PE to PG						
		Wall	063-P16	Corridor	18	24	-	24
		Wall	057-P01	Elev. Hoist Way	18	-	34	-
		Wall	063-P18	Stair	-	32	32	-
		Opening within the column line from 36 to 37 and PB to PE						
		Wall	063-P16	Corridor	20	20	-	20
		Wall	063-P19	Elect. Riser	29	-	29	-
		Wall	063-P39	Spent Resin Long Term Storage Pump Room	18	21	15	21
		Wall	063-P20	HVAC Chase	-	18	30	-

(1) Opening within the column line from 36 to 37 and from PA to PB

Table 12.3-5 (13 of 16)

Room Number	Room Name	Face to (side wall) Structure	Room Number	Room Name	Minimum Required Shield Thickness (inches)
Additional Information on the Rooms which Contain Complex Geometries (cont.)					
077-P01	Hot Pipe Way	Wall	-	Yard(1)	28
		Wall	063-P23	Equipment Waste Tank Room	15
		Wall	063-P24	Equipment Waste Tank Room	15
		Wall	063-P28	Floor Drain Tank Room	16
		Wall	063-P29	Floor Drain Tank Room	19
		Wall	063-P30	Chemical Waste Tank Room	10
		Wall	063-P31	Chemical Waste Tank Room	10
		Wall	063-P16	Corridor(2)	18
		Wall	063-P52	Chemical Drain Sump Pump Room	14
		Wall	062-P02	Mask Decontamination Room	14
		Wall	063-P64	Corridor	14
		Wall	063-P61	Laundry Storage	14
		Wall	063-P51	Stair	17
		Wall	063-P78	Pipe Chase	14
		Wall	063-P32	Detergent Waste Tank & Pump Room	14
		Wall	063-P37	Monitor Tank Room	18
		Wall	063-P16	Corridor(3)	18
		Wall	063-P46	Corridor	
				- Wall within the column line from 37 to 38 along the row line PG	18
				- Wall within the row line from PG to PH along the column line 38	25
		Wall	063-P38	PSS-Solidification	16
		Wall	085-P45	Opening for Drum Removal Chase (West wall)	16
		Wall	063-P48	CTS-Dryer Skid Room	17
		Wall	063-P13	Hot Pipe Chase	10

(1) Exterior wall within the column line from 35 to 36 along the row line PA
(2) Section within the column line from 33 to 35 and from PF to PG
(3) Side wall within the column line from 36 to 37 along the row line PG

Table 12.3-5 (14 of 16)

Room Number	Room Name	Face to (side wall)			Minimum Required Shield Thickness (inches)
		Structure	Room Number	Room Name	
Additional Information on the Rooms which Contain Complex Geometries (cont.)					
077-P01	Hot Pipe Way	Wall	-	Yard(4)	38
		Wall	063-P11	Corridor(5)	
				- South wall	28
				- East wall	28
		Wall	063-P08	Low Activity Spent Resin Tank Room	10
		Wall	063-P06	Future Use Area	32
		Wall	063-P05	Spent Resin Long term Storage Tank Room	32
		Wall	063-P02	GRS Header Drain Tank	24
		Wall	063-P16	Corridor(6)	24
		Wall	063-P01	Hot Pipe Chase	17
		Wall	-	Yard(7)	32

(4) Exterior wall within the row line from PF to PI along the column line 39
(5) Section within the column line from 38 to 39 and from PE to PF
(6) Section within the column line from 38 to 39 and from PB to PC
(7) Exterior wall within the column line from 37 to 38 along the row line PA

Table 12.3-5 (15 of 16)

Room Number	Room Name	Face to (side wall)			Minimum Required Shield Thickness (inches)
		Structure	Room Number	Room Name	
Additional Information on the Rooms which Contain Complex Geometries (cont.)					
077-P01	Hot Pipe Way	Slab	085-P14	Corridor(1)	18
		Slab	085-P17	Valve Room	10
		Slab	085-P20	Valve Room	10
		Slab	085-P31	Primary Sampling Room	10
		Slab	085-P32	Primary Sampling Sink Room	10
		Slab	085-P33	Hot Tool Room	16
		Slab	085-P35	Storage	14
		Slab	085-P15	Valve Room	10
		Slab	085-P16	Valve Room	10
		Slab	085-P44	RO Feed Tank Room	21
		Slab	085-P45	Drum Removal Chase	16
		Slab	085-P46	MF Membrane Module Room	20
		Slab	085-P07	Valve Room	16
		Slab	085-P08	Valve Room	22
		Slab	085-P14	Corridor(2)	30
		Slab	085-P14	Corridor(3)	24
		Slab	085-P42	IX Module Room	16
		Slab	085-P43	IX Module Room	16
		Slab	085-P01	Waste Gas Dryer Skid Room	15
		Slab	085-P02	Waste Gas Dryer Skid Room	16
		Slab	085-P03	Valve Room	19
		Slab	085-P04	Charcoal Guard Bed Room	10
		Slab	085-P06	Valve Room	22
		Slab	085-P21	Charcoal Guard Bed Room	10

(1) Section within the column line from 35 to 37 and from PF to PG
(2) Section within the column line from 37 to 38 and from PE to PF
(3) Section within the column line from 36 to 37 and from PB to PE

Table 12.3-5 (16 of 16)

Piping Area	Description	Minimum Required Shield Thickness (inches)
Additional Information on the Rooms which Contain Complex Geometries (cont.)		
Piping area in the mezzanine floor adjacent to CSHX RM (055-A01D)	North wall within the column line from 14 to 15 along the row line AJ	36
	North wall within the column line from 15 to 16 along the row line AJ	48
	East wall	36
	South wall within the column line from 15 to 16 along the row line AI	48
	West wall within the row line from AI to AJ along the column line 15	48
	South wall within the column line from 14 to 15	30
	West wall within the row line from AI to AJ along the column line 14	42
	Floor	24
	Ceiling	30
Piping area in the mezzanine floor adjacent to CSHX RM (055-A01C)	South wall within the column line from 14 to 15 along the row line AB	36
	South wall within the column line from 15 to 16 along the row line AB	48
	East wall	36
	North wall within the column line from 15 to 16 along the row line AC	48
	West wall within the row line from AB to AC along the column line 15	48
	North wall within the column line from 14 to 15	30
	West wall within the row line from AB to AC along the column line 14	42
	Floor	24
	Ceiling	30

Table 12.3-6 (1 of 2)

Areas Potentially Greater than 1 Gy/hr[1]

Auxiliary and Containment Building El. 55 ft; See Figure 12.3-1.

Area	Coordinates	HRA / VHRA[3]	Access Control
ICI Cavity	AF-AG, 18-19	VHRA	Locked Door

Auxiliary and Containment Building El. 78 ft; See Figures 12.3-2 and 3.

Area	Coordinates	HRA / VHRA	Access Control
Hold-Up Volume Tank	AE-AG, 20-21	HRA	Locked Door
Core Debris Chamber VH	AE-AG, 17-18	VHRA	Locked Door
Pre-Holdup Ion Exchanger Pit	AC-AD, 23-24	HRA	Hatch
Purification Ion Exchanger Pit VH	AB-AC, 23-24	VHRA	Hatch
Purification Filter Pit VH	AC-AD, 24-25	VHRA	Hatch
Filter Cartridge Storage VH	AA-AB, 24-25	VHRA	Hatch

Auxiliary and Containment Building El. 100 ft; See Figure 12.3-4.

Area	Coordinates	HRA / VHRA	Access Control
Reactor Cavity VH	AE-AG, 18-20	VHRA	Locked Door
Steam Generator Cavity	AD-AE & AG-AH, 19	HRA	Locked Door
Reactor Drain Tank Room	AE-AF, 16-17	HRA	Locked Door
Volume Control Tank Room VH	AD-AE, 24-25	VHRA	Locked Door

Auxiliary and Reactor Containment Building El. 120.0 ft; See Figure 12.3-5.

Area	Coordinates	HRA / VHRA	Access Control
Refueling Pool Area[2]	AE-AG, 17-21	VHRA	No Entrance
Transfer Tube Inspection Area[2]	AF-AG, 21-23	VHRA	Locked Door
Spent Fuel Pool[2]	AG-AH, 23-25	VHRA	No Entrance
Cask Loading Pit[2]	AH-AI, 23-24	VHRA	No Entrance
Refueling Canal[2]	AF-AG, 23-25	VHRA	No Entrance

(1) During normal operating conditions and AOOs

(2) Only when fuel is in the area

(3) HRA : High Radiation Area,
VHRA : Very High Radiation Area (Greater than 5 Gy/hr)

Table 12.3-6 (2 of 2)

Compound Building El. 63 ft; See Figure 12.3-10.

Area	Coordinates	HRA / VHRA	Access Control
GRS Header Drain Tank Room	PB-PC, 38-39	HRA	Locked Door
Spent Resin Long Term Storage Tank Room	PC-PD, 38-39	VHRA	Locked Door
Future Use	PD-PE, 38-39	VHRA	Locked Door
Hot Pipe Chase	PI-PJ, 38-39	HRA	Locked Door
Future Use(4)	PB-PC, 32-33	VHRA(4)	Locked Door(4)

Compound Building El. 77 ft; See Figure 12.3-11.

Area	Coordinates	HRA / VHRA	Access Control
Hot Pipe Way	PA-PI, 33-39	HRA	Hatch

Compound Building El. 85 ft; See Figure 12.3-12.

Area	Coordinates	HRA / VHRA	Access Control
R/O Membrane Module & Valve Skid Room	PI-PJ, 37-39	HRA	Locked Door

Compound Building El. 100 ft; See Figure 12.3-13.

Area	Coordinates	HRA / VHRA	Access Control
Charcoal Delay Bed Rooms (096-P01, P02)	PB-PC, 37-39	VHRA	Hatch
Spent Filter Drum Storage Area	PI-PJ, 38-39	VHRA	Locked Door
Truck Bay(5)	PF-PG, 37-39	HRA	Locked Door
Future Extension Area(5)	PE-PF, 37-39	HRA	Locked Door

(4) COL applicant is to determine the purpose and use of the room (063-P73). The access control for this room shall be changed accordingly in compliance with the guidance in NRC RG 8.38. In addition, COL applicant is to specify any necessary radiation monitoring requirements, and any additional necessary design features and controls to ensure compliance with applicable regulations, including 10 CFR Part 20 and 10 CFR Part 36 (COL 12.3(7)).

(5) Only during transfer and drumming of spent filter and spent resin

Table 12.3-7 (1 of 2)

Area Radiation Monitors

Description	Tag No.	Class(1)			Range					Function and Remarks
		S	SE	E	Airborne Particulate	Iodine	Gas	Liquid	Area(2) (mSv/hr)	Display & Alarm at MCR/RSR/Local
Post-accident primary sample room	RE-205	N	III	N	N/A	N/A	N/A	N/A	10^{-3}~10^{2}	Yes/Yes/Yes
Normal primary sample room	RE-285	N	III	N	N/A	N/A	N/A	N/A	10^{-3}~10^{2}	Yes/Yes/Yes
Main steam and FW containment piping penetration area	RE-237 RE-238	N	II	N	N/A	N/A	N/A	N/A	10^{0}~10^{5}	Yes/Yes/Yes
Containment operating area	RE-231A RE-232B	3	I	A B	N/A	N/A	N/A	N/A	10^{-3} ~ 10^{2}	Yes/Yes/Yes CPIAS
Containment upper operating area	RE-233A RE-234B	3	I	A B	N/A	N/A	N/A	N/A	10^{1} ~ 10^{8}	Yes/Yes/Yes ERDS(3) CPIAS
In-core instrument	RE-235	N	II	N	N/A	N/A	N/A	N/A	10^{-3} ~ 10^{2}	Yes/Yes/Yes
Containment personnel access hatch area	RE-236	N	II	N	N/A	N/A	N/A	N/A	10^{-3} ~ 10^{2}	Yes/Yes/Yes
Spent fuel pool area	RE-241A RE-242B	3	I	A B	N/A	N/A	N/A	N/A	10^{-3} ~ 10^{2}	Yes/Yes/Yes FHEVAS
New fuel storage area	RE-245	N	II	N	N/A	N/A	N/A	N/A	10^{-3} ~ 10^{2}	Yes/Yes/Yes

Table 12.3-7 (2 of 2)

Description	Tag No.	Class(1)			Range					Function and Remarks
		S	SE	E	Airborne Particulate	Iodine	Gas	Liquid	Area(2) (mSv/hr)	Display & Alarm at MCR/RSR/Local
Hot machine shop	RE-293	N	III	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes
Radiochemistry lab	RE-257	N	III	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes
Main control room area	RE-275	N	II	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes
TSC area	RE-279	N	III	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes
Truck bay area	RE-288 RE-289	N	III	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes
Waste drum storage area	RE-292	N	III	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes
Compound building dry active waste storage area	RE-284	N	III	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^{2}$	Yes/Yes/Yes

(1) S: safety Class per ANSI/ANS 51.1; 1=SC-1, 2=SC-2, 3=SC-3, N=NNS
SE: seismic Category I, II, III
E: Electrical Class A, B, C, D=Class 1E Separation Division, N=Non-Class 1E
Refer to Section 3.2 for the definition.

(2) Detector type for area radiation monitor is GM tube or ionization chamber

(3) The monitor transmits the radiation signals to the licensing entity via the emergency response data system (ERDS) link. As described in APR1400 DCD Tier 2 Section 13.3, the ERDS is a real-time electronic data transmission system to the NRC operations center that provides a set of parameters from the onsite computer system in the event of an emergency. The ERDS transmits information to allow the NRC to provide advice and support to the licensee, state, and local authorities, and other federal officials. The ERDS satisfies the requirements in 10 CFR Part 50, Appendix E.

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-1   Radiation Zones(Normal) Auxiliary/Containment Building El. 55'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-2 Radiation Zones (Normal) Auxiliary Building - Partial Plan El. 68'-0", 77'-0", and 86'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-3 Radiation Zones (Normal) Auxiliary/Containment Building El.78'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-4   Radiation Zones(Normal) Auxiliary/Containment Building El.100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-5 Radiation Zones(Normal) Auxiliary/Containment Building El.120'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-6   Radiation Zones(Normal) Auxiliary/Containment Building El.137'-6"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-7   Radiation Zones(Normal) Auxiliary/Containment Building El.156'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-8 Radiation Zones(Normal) Auxiliary/Containment Building El.174'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-9 Radiation Zones (Normal) Auxiliary/Containment Building El. 195'-0" and Roof Plan

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-10   Radiation Zones (Normal) Compound Building El. 63'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-11   Radiation Zones (Normal) Compound Building Partial Plan El. 77′-0″

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-12   Radiation Zones (Normal) Compound Building El. 85'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-13 Radiation Zones (Normal) Compound Building El. 100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-14   Radiation Zones (Normal) Compound Building El. 120'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-15 Radiation Zones (Normal) Compound Building El. 139'-6"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-16   Radiation Zones (Normal) Compound Building El. 157'-9" & Roof Plan

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-17 Radiation Zones (Normal) Turbine Building El. 73'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-18 Radiation Zones (Normal) Turbine Building El. 100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-19A Radiation Zones (Normal) ESW/CCW Hx Building (DIV. I)

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-19B   Radiation Zones (Normal) ESW/CCW Hx Building (DIV. II)

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-20 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 55'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-21   Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 55'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-22 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 55'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-23 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 55'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-24 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 78'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-25 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 78'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-26 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 78'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-27 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 78'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-28 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-29 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-30   Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-31 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 100'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-32 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 120'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-33 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 120'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-34 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 120'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-35 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 120'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-36 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 137'-6"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-37 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 137'-6"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-38 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 137'-6"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-39 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 137'-6"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-40 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 156'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-41 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 156'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-42 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 156'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-43   Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 156'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-44 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 174'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-45 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 174'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-46 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 174'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-47 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 174'-0"

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-48 Radiation Zones (1 Hour after Accident)-Auxiliary/Containment Building El. 195'-0" & Roof Plan

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-49 Radiation Zones (1 Day after Accident)-Auxiliary/Containment Building El. 195'-0" & Roof Plan

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-50 Radiation Zones (1 Week after Accident)-Auxiliary/Containment Building El. 195'-0" & Roof Plan

Security-Related Information - Withhold Under 10 CFR 2.390

Figure 12.3-51 Radiation Zones (1 Month after Accident)-Auxiliary/Containment Building El. 195'-0" & Roof Plan

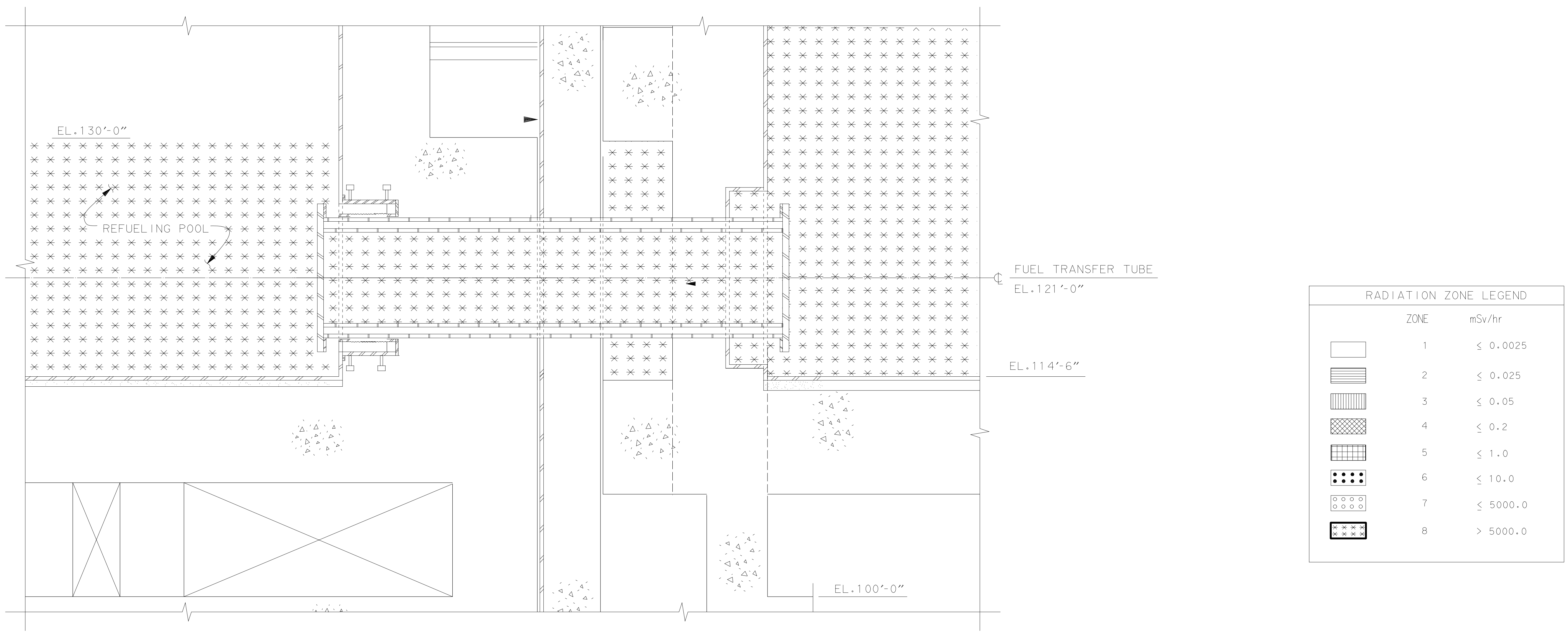


Figure 12.3-52 Sectional View of Fuel Transfer Tube

12.4 Dose Assessment and Minimization of Contamination

12.4.1 Dose Assessment

12.4.1.1 Estimated Annual Occupational Exposures

12.4.1.1.1 ALARA Design Features for Occupational Dose Reduction

A review of the operating data indicates that a significant portion of the occupational exposures (from 50 to 75 percent) are related to the exposure to activated corrosion products. Minimization of primary system corrosion and resultant dose rates is the most effective approach to reduce total personnel occupational exposures.

Leakage of fuel rod cladding accounts for the remaining 25 to 50 percent of pressurized water reactor (PWR) occupational exposure. The fuel rod performance of the APR1400 design is expected to reduce fuel leakage to less than 0.1 percent. This feature also is expected to reduce effluent releases and radwaste activity.

It is expected that proper material selection, chemistry control, and improved fuel cladding leakage reduce the annual occupational exposure.

The design features described in Subsection 12.3.1.5 for SGs greatly reduce the time spent in performing maintenance and inspection. This task represents approximately 25 percent of the total dose for the average PWRs.

The dose received during an outage for SG maintenance is dependent on the number of tubes requiring inspection and repair. The APR1400 SG design provides thermal treatment of the SG tubes and the manufacturing technique found to have avoided the known SG tube cracking. SG tubes are fabricated of Alloy 690 instead of Alloy 600. Alloy 690 has been proven to be less susceptible to SG tube cracking.

The Palo Verde SG tubes are fabricated of Alloy 600 and processed with thermal treatment. The Palo Verde plants are known to have no tube cracking problems. This has resulted in a significant reduction in the annual dose received during SG maintenance. Palo Verde's average occupational exposure due to SG maintenance during a refueling outage has been 0.40 person-Sv, in contrast to the Duke Power Company's average of 0.62 person-Sv. Therefore, a factor of 1.6 reduction in annual occupational exposure is expected for SG maintenance based on improved manufacturing techniques and material selection.

Advanced PWRs like the APR1400 have several additional features that are not provided at Palo Verde or the Duke Power Company's units as described in Subsection 12.3.1.5. These features reduce the time spent in maintaining the SG and are therefore expected to reduce the annual occupational exposure due to steam generator maintenance by a factor of 1.5. In addition, a factor of 2.5 reduction is expected from material selection, chemistry control, and improved fuel performance as described above. Therefore, all of the above advanced PWR design features result in an overall reduction of the annual occupational exposure due to steam generator maintenance by a factor of 6.

The occupational exposure due to RCP maintenance and inspection represents approximately 4 percent of total station exposures. The APR1400 design minimizes the need for seal replacement and reduces the time for performing this task.

Many other features to reduce occupational exposures are applied to the APR1400 design. Some examples are:

a. Extended fuel cycle

b. Identification of RCS leakage

c. Integrated head assembly

d. Single stud tensioner

e. Permanent refueling pool seal

f. RCS equipment and piping materials

   These materials in direct contact with the RCS have a low cobalt content.

g. Steam generator tube material

   This material has an average cobalt content of less than 0.015 wt%.

h. Equipment reliability, maintainability, and accessibility

   The APR1400 is designed so that adequate spacing is provided around components. This enhances accessibility, maintainability, and inspectability. In addition, reliable equipment is used so that the frequency of maintenance is minimized for dose reduction.

i. Components such as tanks, piping, and instruments are designed to minimize particulate deposition in accordance with NRC RG 8.8 (Reference 1) recommendations. For example, tanks with sloped bottoms are used in the APR1400 design.

j. System flushing and decontamination capability

The APR1400 design provides flushing and decontamination capability to lines where particulate deposition is possible, such as resin transfer lines. This enables operators to flush the lines to prevent buildup of radioactive particulates in the lines, which could result in high personnel exposure.

k. Radwaste handling operations

Radwaste handling operations are described in detail in Sections 11.2, 11.3, and 11.4. These systems are designed to operate remotely to the extent practicable, and waste streams are segregated to facilitate processing of waste effectively and efficiently and minimize radiation exposure.

l. Isolation of contaminated components and proper shielding

The APR1400 design segregates the radioactive systems from the nonradioactive systems. In addition, components in the radioactive systems are separately located in cubicles based on the source strength, the nature of operation, and the frequency of access. The design also provides adequate shielding between components, valves, and pump galleries to enable personnel to perform operational activities and maintenance in areas of lower radiation.

m. Design for recovery from abnormal occurrences

1) Divisional separation

The APR1400 design provides separation of SSCs, which meets the single-failure criterion. Fire and flood barriers are provided to prevent the spread of such events and the loss of equipment in adjacent areas. This design approach also facilitates speedier recovery from fire or flood events.

2) RCS depressurization

Use of the IRWST enables the operators to depressurize the RCS by discharging the coolant to the IRWST through the POSRVs. This tank allows the dissipation of a significant portion of energy to the IRWST, thus facilitating the recovery from loss of heat sink events.

3) Safety injection

The provision of a direct header from safety injection, which takes suction from the IRWST to the RV, reduces the likelihood of a failure to provide safety injection water to the reactor.

4) Safety-related equipment power supply

The APR1400 design provides Class 1E and redundant electric power to safety-related equipment, thus reducing the probability of safety-related equipment failure due to a loss of power.

These features are further expected to reduce occupational radiation exposures to ALARA levels.

12.4.1.1.2 Occupational Dose Assessment

The expected annual cumulative occupational radiation exposure (ORE) from the APR1400 operation is estimated according to the guidance in NRC RG 8.19 (Reference 2). The APR1400 ORE estimation is based on operational experience data from the OPR1000 plant, which is the Korean PWR reactor rated at 1,000 MWe. The design features of this Combustion Engineering type OPR1000 are similar to those of the APR1400.

The dose rates for the APR1400 are conservatively assumed to increase in proportion to the thermal power increase from the OPR1000. During a refueling outage, the ORE is strongly dependent on the activated corrosion products. Although the deposition of the activated corrosion products in the APR1400 is expected to be less because of design improvements, the reduction is not taken into account in the ORE estimation.

According to NRC RG 8.19, the ORE assessment should be based on anticipated radiation conditions after at least 5 years of plant operation. The measured ORE data for a rolling 10-year period from 2004 to 2013 for the OPR1000 (Hanul Unit 3), which started commercial operation in August 1998, are used in the ORE assessment. The internal and external doses due to airborne radionuclides are negligible because the measurement data for the OPR1000 and NUREG-0713 (Reference 3) demonstrate that the ORE contribution from airborne radioactivity is minimal.

The estimated annual occupational radiation exposures for the APR1400 are determined within the following work categories:

a. Reactor operations and surveillance

b. Routine maintenance

c. Inservice inspection

d. Special maintenance

e. Waste processing

f. Refueling

The estimated result of occupational exposure for the APR1400 is presented in Table 12.4-7.

12.4.1.1.2.1 Reactor Operations and Surveillance

The performance of various systems and components is monitored by operators. The inspection includes routine inspections and performance tests such as system leakage monitoring, control of instrumentation, electrical and mechanical equipments, and preventive maintenance. Health physics surveys are also included. In some cases, operation of some manual valves requires personnel to enter radiologically controlled areas (RCAs) for short periods.

Examples of routine operation and surveillance activities are as follows:

a. Routine checking and patrol

b. System operation and checking

c. Instrumentation inspections and calibration

d. Unidentified leak checking

e. Communication system checking

These activities may be conducted in the reactor containment, auxiliary building, compound building, and turbine building. The frequency and duration of activities, which are the major factors for ORE dose, are dependent on the operational programs and procedures.

Improvement of the plant layout, access provisions, and operational procedures can reduce exposure time in RCAs for surveillance, inspection, and testing works, thereby minimizing occupational radiation exposure. Details of the design features to minimize ORE are described in Subsection 12.3.1.

Table 12.4-1 provides a breakdown of the collective doses for reactor operations and surveillance activities.

12.4.1.1.2.2 Routine Maintenance

Routine maintenance is required for mechanical and electrical components throughout the plant. Maintenance includes routine scheduled maintenance and inspection activities. Occupational exposure can be reduced by reducing the number of components/equipment that require maintenance and providing ease of maintenance, accessibility of equipment, and ample workspace.

Examples of routine maintenance activities are as follows:

a. Health physics activities

b. Washing and decontamination

c. Leak testing (LLRT, ILRT)

d. Filter & resin replacement

e. Inspection of components

    - RCS pumps

    - Primary circuit

    - Other pumps

    - Other valves

- Other mechanical components
- Electrical components

f. Ventilation and filtration system

g. Insulation/shielding and support

h. Miscellaneous works

Table 12.4-2 provides a breakdown of the collective doses for routine maintenance activities.

12.4.1.1.2.3 Inservice Inspection

ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Reference 4) requires extensive periodic inservice inspection of various safety-related systems/components and associated welds. These inservice inspections are generally performed during refueling outage.

Examples of inservice inspection activities are as follows:

a. Pressure test

b. Visual inspections

c. Nondestructive and ultrasonic examinations

d. Welds examinations

e. Installation and removal of thermal insulation, platform, etc., and cleanup

Table 12.4-3 provides a breakdown of the collective doses for inservice inspection activities.

12.4.1.1.2.4 Special Maintenance

Maintenance beyond routine scheduled maintenance is considered special maintenance and can be performed only during refueling outages. This category includes the modification of equipment to upgrade the plant and repairs to any failed components. For special maintenance activities that lead to significant occupational exposure, ORE reduction can be achieved through improved reliability of components and equipments, ease of maintenance or replacement, use of temporary shielding, improvement of maintenance procedures, and use of remotely operated equipment and robotics.

Examples of special maintenance activities are as follows:

a. Steam generator maintenance

b. Repairs on valves, pumps, heat exchangers, tanks, and other components

c. Pressurizer maintenance

Table 12.4-4 provides a breakdown of the collective doses for special maintenance activities.

12.4.1.1.2.5 Waste Processing

Radioactive wastes, except spent fuels, are processed by the LWMS, SWMS, and GWMS in the compound building. Radioactive waste processing activities include processing, collecting, storage, and handling of radioactive liquids, solid, and gaseous wastes generated during plant operation. The systems are operated remotely to minimize the expected occupational exposure.

Table 12.4-5 provides a breakdown of the collective doses for waste processing activities.

12.4.1.1.2.6 Refueling

Refueling replaces depleted spent fuels with new fuels on an 18-month fuel cycle. This includes receipt and inspection of new fuel assemblies, transfer of fuel assemblies, and replacement of control rods or in-core components. Reactor vessel (RV) opening/closure activities are also included because the RV head must be disassembled and removed before commencing refueling operations. Most of the refueling activities, including cask loading operations, are conducted underwater to provide shielding and maintain occupation dose ALARA. Occupational exposure is also reduced by using automatic cleaning equipment, remote fuel handling, and an integrated head assembly (IHA).

Examples of refueling activities are as follows:

a. RV opening/closure

b. Refueling

c. Stud bolt hole cleaning

Table 12.4-6 provides a breakdown of the collective doses for refueling activities.

12.4.1.2 Post-Accident Actions

Following a DBA, the vital functions can be performed with an operator radiation exposure of no more than 50 mSv total effective dose equivalent (TEDE) in accordance with the alternative source term (AST) approach, 10 CFR 50.34(f)(2)(vii), GDC 19, NUREG-0737 (Reference 5), Item II.B.2, and plant operating and emergency procedures. For the radiation exposure assessment presented in this section and summarized in Table 12.4-8, the radiation exposures resulting from operator actions required during post-LOCA conditions are evaluated as the bounding DBA for post-accident radiation exposures.

The areas requiring continuous occupancy or infrequent access to perform post-accident vital functions are described in Subsection 12.3.1.8.

All vital areas requiring infrequent access are located in various rooms at different floor elevations in the auxiliary building (AB). The only significant submergence radiation dose that an individual would receive in accessing (transit time) and occupying (stay time) areas that are infrequently accessed is that from the airborne source. The plant emergency procedure mandates the use of respiratory protection to limit intake of airborne radioactive material when the potential for a radiation hazard of 1 DAC exists for the airborne activity. The assigned protection factor (APF) of 10,000 for a positive pressure self-contained breathing apparatus (SCBA) is used to calculate the inhaled dose rates for the areas requiring infrequent access. The COL applicant is to provide a respiratory protection program to minimize airborne radiological hazards while performing post-accident vital functions. The respiratory protection program should include the provisions of the positive pressure self-contained breathing apparatus (SCBA) with a minimum rated service life of 1 hour in the control room and air supply systems in areas where post -accident mission times may exceed the 1-hour SCBA supply and where necessary to account for uncertainty in respirator service life in performing post-accident missions. The COL applicant will also assess if replenishing the respirators during vital missions will result in any increase to the vital area mission times and doses (COL 12.4(4)). The respirator protections are not provided for areas requiring continuous occupancy, such as the main control room (MCR) and technical support center (TSC).

The dose contribution from the post-LOCA containment low-volume purge system release is negligible. Therefore, the whole body (WB), which is the deep dose equivalent (DDE), dose rates and the TEDE dose rates due to the post-LOCA containment and ESF leakages are calculated for radiation exposures beginning at 1 hour, 2 hours, 4 hours, 8 hours, and 24 hours (1 day) using AST methodology.

The radiation exposures from the containment shine are negligible because most of the vital areas requiring infrequent access are heavily self-shielded by concrete structures of the auxiliary building and containment building. However, the Class 1E switchgear rooms and I&C equipment rooms which are located adjacent to the containment building have a minimum concrete shielding of just the containment cylindrical wall of 1.2 m (4 ft). Therefore, the post-LOCA containment shine dose rates are only considered in these areas.

The MCR and TSC are identified as the areas requiring continuous occupancies to perform the post-accident vital functions. These areas are within the control room (CR) envelope (CRE) boundary. The post-LOCA doses in the CRE are listed in Table 6.4-2. To calculate the CR average dose rate, the CR operator occupancy time based on a 30-day accident duration and CR occupancy factors contained in NRC RG 1.183 (Reference 6) are considered.

The mission doses and dose rates for areas requiring continuous occupancy are listed in Table 12.4-8. The results show that the doses and dose rates meet the criteria of 50 mSv TEDE for 30 days and 0.15 mSv/hr averaged over 30 days as specified in NUREG-0737, Item II.B.2.

The mission dose and dose rates for areas requiring infrequent access to perform post-accident vital functions are also presented in Table 12.4-8. The estimate of the access times is based on a walk speed of 4 km/h (13000 ft/h) for horizontal pathways and 2 km/h (6500 ft/h) where upstairs/downstairs are included. It is assumed that 20% margin is added to the transit time and the direct dose to the vital area mission pathways is doubled to account for any potential nonconservatism due to simplifications in the calculations. Table 12.4-8 indicates that the doses to the plant personnel calculated based on anticipated stay and access times at various areas at various times are within the criterion of 50 mSv TEDE as specified in NUREG-0737, Item II.B.2.

12.4.1.3 Estimated Annual Dose at the Exclusion Area Boundary

The direct radiation from the reactor containment, auxiliary, compound, and turbine generator buildings is negligible compared with that from outside storage tanks. These tanks are expected to contain radioactivity that yields a dose rate of 0.0025 mSv/hr or less at the tank shielded surface. Therefore, the estimated annual direct dose at the site boundary is expected to be small.

The estimated doses at the site boundary due to released activity are given in Subsections 11.2.3.1 and 11.3.3.1. The resultant annual doses for the general public are within the limits of the applicable regulations.

12.4.1.4 Estimated Dose to Construction Workers

The construction worker doses are the sum of the exposures due to direct radiation and airborne radiation from the adjacent operating unit(s). The dose criterion to the construction workers is 1 mSv/yr. The construction worker doses are site-specific and based on the number of operating units, distances, meteorological conditions, and construction schedule. The COL applicant is to estimate construction worker doses based on site-specific information such as the number of operating units, distances from radiation sources, meteorological conditions, and construction schedule (COL 12.4(1)).

12.4.2 Minimization of Contamination and Radioactive Waste Generation

The APR1400 design includes features and operational programs to conform with 10 CFR 20.1406 and NRC RG 4.21 (Reference 7) to minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste throughout the life cycle of the facility.

The design includes control measures that address the requirements in Regulatory Positions C.1 through C.4 and Appendix A of NRC RG 4. 21. The control measures are designed to work in conjunction with the operational procedures for the collection, processing, sampling, storing, monitoring, and disposal of radioactive waste during normal operation, including AOOs. The COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach in accordance with NRC RG 4.21 and NRC RG 4.22, as applicable, and the documentation required by 10 CFR 20.1501 (COL 12.4 (2)). The design also takes into account the life-cycle planning of the facility and through COL 12.4(2), identifies the operational programs and procedures to address maintenance of structures, systems, and components (SSCs), site radiological environmental monitoring, and documentation of operational incidents for decommissioning planning associated with NRC RG 4.21. The APR1400 systems with the potential to contain radiological contamination are listed in Table 12.4-9 and require full NRC RG 4.21 evaluations.

Because NRC RG 4.21 control measures cover different phases of the plant life cycle, the APR1400 design divides the control measures into the following six design and operational objectives that are used to evaluate the conformance of the measures with 10 CFR 20.1406 and NRC RG 4.21:

a. Objective 1 – Prevention/Minimization of Unintended Contamination

b. Objective 2 – Provision of Adequate and Early Leak Detection Capability

c. Objective 3 – Reduction of Cross-Contamination, Decontamination, and Waste Generation

d. Objective 4 – Decommissioning Planning

e. Objective 5 – Operations and Documentation

f. Objective 6 – Site Radiological Environmental Monitoring

The design features in the systems that are intended to meet the objectives for minimizing contamination and the generation of radioactive waste are described in Table 12.4-10. The table also provides cross-references to the DCD sections that provide further information on how the SSCs meet the requirements of 10 CFR 20.1406 and NRC RG 4.21.

12.4.2.1 Design Considerations

The control measures delineated in Regulatory Positions C.1 through C.4 and Appendix A of NRC RG 4.21 are categorized into four design objectives that address the design features described below.

Objective 1 – Prevention/Minimization of Unintended Contamination

Objective 1 requires control measures to prevent and/or minimize radioactive contamination of the facility and environment. This objective is met by implementing design features such as:

a. Selection of proper materials for SSCs to withstand process conditions without the generation of excess contamination, such as corrosion-resistant piping

b. Use of containment walls or dikes around contaminated or potentially contaminated SSCs and other means of prevention/minimization such as leak detection, drain collection, double-walled piping, and piping sleeves with leak detection for piping between buildings

c. Minimization of buried or embedded piping and drains (alternatively utilizing piping within a concrete structure or double-walled piping, both with leak detection capabilities)

d. Provisions for proper operational interlocks and alarms and administrative controls to avoid inadvertent bypasses and releases

e. Minimization of the potential for component failures through the use of nuclear-industry-proven technologies, reinforced quality control, and strict adherence to the applicable codes and standards

<u>Objective 2 – Provision of Adequate and Early Leak Detection Capability</u>

Objective 2 requires provisions for leak detection capabilities, allowing prompt assessment to support a timely and appropriate response in the event of an unintended release of radioactive contamination. Leak detection systems are included within the facility design to initiate alarm signals in the radwaste control room (where applicable) and MCR for timely operator actions. This objective is met by the implementing the following design features:

a. Adequate leak detection instruments (including individual and separate leak detection instruments for plant SSCs with a potential for leakage to the extent possible)

b. Facility design (e.g., floors that slope toward drainage collection piping, level detection instruments, smooth surfaces for drainage, catch basins)

c. Drainage collection and pathways and the placement of detection instruments as close as practicable to the sources of leaks

d. Sensitive (including smaller liquid volume collection) liquid/moisture detection instruments

e. Facilitation of periodic calibration and replacement of leak detection instruments

f. Adequate access pathways and spaces for prompt assessment of detected leakage and appropriate operator responses, tank overflow management, and adequate sampling points for operational analysis

Objective 3 – Reduction of Cross-Contamination, Decontamination, and Waste Generation

Objective 3 is met by providing capabilities to reduce cross-contamination, the need for decontamination, and the generation of radioactive waste. The control measures that are implemented to meet this objective include:

a. Segregation of components in accordance with contamination types, levels, and characteristics

b. Onsite decontamination facility

c. Flushing capabilities for components and piping that handle or transport radioactive fluid

d. Smooth and cleanable surfaces for component fabrication and facility structures such as epoxy coating for cubicle floors

e. Capability to temporarily hold and isolate contamination and quickly initiate operator action to provide timely and appropriate responses

f. Life-cycle planning of waste processing, packaging, handling, storing, and shipment for disposal

Objective 4 – Decommissioning Planning

Objective 4 is met by providing features that facilitate decommissioning. The design features include:

a. Modular construction

b. Minimization of buried and embedded components and piping

c. Documentation and information in a centralized area for ready recovery

d. Removable block walls to facilitate the removal of SSCs

e. SSCs that minimize the generation of radioactive waste during decommissioning activities

12.4.2.2 Operational/Programmatic Considerations

The operational programs and procedures in 10 CFR 20.1406 and 10 CFR 20.1501 must be met. The COL applicant is to address Objectives 5 and 6, described below, to meet the operational and programmatic requirements of NRC RG 4.21 Positions C.1 through C.4 and NRC RG 4.22 (Reference 8), as applicable (COL 12.4(2)).

The requirements are related to procedures and programs that support the operation and maintenance of the design features described in Objectives 1 through 4 and documentation of any incidents that may occur during the operating life of the facility. The documentation of these incidents, including spillage, leakage, overflows, and associated cleanup requirements, is an integral part of decommissioning planning.

Objective 5 – Operations and Documentation

Objective 5 is met by developing operational procedures for the following:

a. Supplemental control of the SSCs to prevent/minimize the spread of contamination

b. Periodic review of operational procedures to provide reasonable assurance that the procedures reflect lessons learned and up-to-date content, including personnel qualification and training

c. Accurate records and documentation of design, construction, modifications, and operational incidences to facilitate decommissioning

d. Development of radwaste processing and management programs

e. Development of maintenance and surveillance programs in accordance with NRC RG 4.21 and NRC RG 4.22, as applicable, to provide reasonable assurance that SSCs will perform as designed to prevent the spread of contamination

f. Documentation of the results of subsurface radiological surveys in accordance with 10 CFR 20.1501

Objective 6 – Site Radiological Environmental Monitoring

Objective 6 is met by developing a conceptual site model (COL 12.4 (2)) that:

a. Includes site-specific hydrogeology, potential migration, and groundwater transport pathways

b. Assesses the effects of construction on hydrogeological characteristics of the site

c. Establishes a site-specific contamination monitoring program for potential groundwater pathways from release source to receptor point, in accordance with 10 CFR 20.1406 and the guidance of NRC RG 4.21

12.4.2.3 Summary of Design Features

The APR1400 design features that minimize facility and environmental contamination are described in the relevant sections of this DCD. Table 12.4-10 summarizes how these features meet the design objectives in accordance with the requirements of 10 CFR 20.1206 and NRC RG 4.21 and provides cross-references to the relevant DCD subsections.

Many of the features are designed to work together to support the defense-in-depth strategy for the prevention and minimization of the contamination of the facility and the environment.

12.4.2.4 Design Details to Address Minimization of Contamination

12.4.2.4.1 Early Leak Detection Drain Pipe

The early leak detection design for plant areas containing radioactive components consists of a drain pipe with a level switch and a manual valve installed on the exposed pipe portion in the valve pit or corresponding location of the trench. The drain pipe collects drainage from cubicles that may contain radioactive liquid from leakage or overflow. The drain pipe is a 5.08 cm (2 in.) stainless steel pipe embedded in the basemat and is sloped to facilitate drainage. The embedded portion of the drain pipe is encased in an outer pipe. The inner pipe is open to drainage into the trench, which directs the collected liquid to the normal sump for collection. Because the drain pipe is relatively small, a small leak can be detected, thus achieving early detection. The level switch is designed to initiate an alarm in the MCR for operator action. Early leak detection capability is built into each cubicle that may contain radioactive or potentially radioactive fluid for quick assessment. The typical design of the leak detection configuration is presented in Figure 12.4-1. The leak detection instrumentation is shown in Figure 12.4-2.

12.4.2.4.2 Leak Collection Trench

Concrete trenches are provided within the basemat of the auxiliary and compound buildings to facilitate leak drainage to liquid collection sumps. The trench has an epoxy coating and is significantly sloped to facilitate flow and prevent the accumulation of fluid along the flow path. The trench has removable steel grated covers to facilitate access for inspection and calibration of leak detection instruments; trench cleaning; and inspection, repair, and maintenance of the epoxy coating. Where area drainage is expected to contain significant debris, the trench is equipped with screens or a drain pipe to facilitate liquid flow without entraining solids or debris in the sumps. The typical design of the compound building trench design is presented in Figure 12.4-3.

12.4.2.4.3 Penetration Design between Buildings

Piping that extends from one building to another with a separation gap between the buildings is equipped with sleeves or double-walled concentric piping to prevent direct, unintended leakage to the environment. The use of either piping sleeves or double-walled concentric piping depends on the separation distances, with piping sleeves generally used for smaller separation gaps and double-walled piping used for larger separation gaps.

Piping sleeves and the outer wall piping for the double-walled piping are equipped with seals to maintain building pressure differentials and to prevent the infiltration of outside water and air into the buildings. Piping sleeves and outer piping are also equipped with sleeve joints to accommodate some degree of building movement. Piping sleeves and outer wall piping are installed with a mild slope to facilitate drainage of liquid in the event of inner pipe leakage. The leakage is drained directly into a nearby sump. The typical designs for piping penetrations between buildings and building sleeves are presented in Figure 12.4-4.

12.4.2.4.4 Minimization of Embedded and/or Buried Piping

Embedded or buried piping is minimized using the following approach:

a. Piping is routed inside pipe chases to the extent possible to minimize embedded piping. Drain pipes on the floors above the basemat level are allowed to penetrate the floor and are routed inside pipe chases below the floor level as required.

b. To the extent practicable, double-walled piping is used when embedded piping segments cannot be avoided. The outside piping of the double wall pipe is designed to drain to a local sump.

c. For the basemat level, drain pipes are routed in concrete trenches that are sloped toward the local sump. The trenches are coated with epoxy to facilitate drainage and cleaning.

d. For piping that carries contaminated or potentially contaminated fluid and is located outside the plant structures, the piping is routed in underground concrete tunnels to the maximum extent practicable. Systems that contain yard piping routed through underground concrete tunnels may include CCWS, ESWS, CVCS, condensate systems, etc., which are determined based upon site-specific plant layout conditions. The COL applicant is to implement concrete tunnels for those systems that include underground piping carrying contaminated or potentially contaminated fluid to minimize buried piping. The tunnels are coated with epoxy and are equipped with sumps with liquid detection level switches. If liquid is accumulated to the detectable level, an alarm is initiated in the MCR for operator actions (COL 12.4(3)).

12.4.3 Combined License Information

COL 12.4(1) The COL applicant is to estimate construction worker doses based on the site-specific information, such as the number of operating units, distances from radiation sources, meteorological conditions, and construction schedule.

COL 12.4(2) The COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach in accordance with NRC RG 4.21 and NRC RG 4.22, as applicable, and the documentation required by 10 CFR 20.1501.

COL 12.4(3) The COL applicant is to implement concrete tunnels for piping of the systems that may include underground piping carrying contaminated or potentially contaminated fluid to minimize buried piping. The tunnels are coated with epoxy and are equipped with sumps with liquid detection level switches. If liquid is accumulated to the detectable level, an alarm is initiated in the MCR for operator actions.

COL 12.4(4) The COL applicant is to provide a respiratory protection program to minimize airborne radiological hazards while performing post-accident vital functions. The respiratory protection program should include the provisions of the positive pressure self-contained breathing apparatus (SCBA) with a minimum rated service life of 1 hour in the control room and air supply systems in areas where post -accident mission times may exceed the 1-hour SCBA supply and where necessary to account for uncertainty in respirator service life in performing post-accident missions. The COL applicant will also assess if replenishing the respirators during vital missions will result in any increase to the vital area mission times and doses.

12.4.4 References

1. Regulatory Guide 8.8, “Information Relevant to Ensuring the Occupational Radiation Exposures at Nuclear Power Stations will be ALARA,” Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

2. Regulatory Guide 8.19, “Occupational Radiation Dose Assessment in Light Water Reactor Power Plants Design Stage Man-Rem Estimates,” Rev. 1, U.S. Nuclear Regulatory Commission, June 1979.

3. NUREG-0713, “Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities,” Vol. 6 to Vol. 31, U.S. Nuclear Regulatory Commission, 1984 to 2009.

4. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

6. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, July 2000.

7. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

8. Regulatory Guide 4.22, "Decommissioning Planning During Operations," U.S. Nuclear Regulatory Commission, December 2012.

Table 12.4-1

Occupational Dose Estimates During Reactor Operations and Surveillance

Activity	Average Dose Rate [µSv/hr]	Number of Workers (1)	Frequency (1)	Exposure Time (1) [hr]	Annual ORE Dose [mSv/yr]
Routine checking and patrol	1.5	2	Per Shift	1.3	4.3
System operation and checking	1.1	3	Per Shift	3.4	12.3
Instrumentation inspections and calibration	6.4	3	Daily	2.7	18.9
Unidentified leak checking	96.1	1	Monthly	0.8	0.9
Communication system checking	1.0	1	Daily	0.6	0.2
Total					36.6

(1) In order to present statistical information on routine and non-routine activities, values for these parameters (number of workers, frequency and exposure time) are adjusted so that the multiplication of these parameters by the average dose rate equals the annual ORE dose, which is the measured value at Hanul Unit 3.

Table 12.4-2

Occupational Dose Estimates During Routine Maintenance

Activity		Average Dose Rate [μSv/hr]	Number of Workers (1)	Frequency (1), (2)	Exposure Time (1) [hr]	Annual ORE Dose [mSv/yr]
Health physics activities		1.6	8	Daily	6.4	29.9
Washing and decontamination		3.9	4	Daily	1.5	8.5
Leak testing (LLRT, ILRT)		3.6	4	Monthly	11.8	2.0
Filter & resin replacement		46.7	4	Monthly	3.6	8.1
Inspection of components	RCS pumps	12.9	10	Per Cycle	299.0	27.0
	Primary circuit	19.2	10	Per Cycle	220.2	29.6
	Other pumps	2.9	5	Per Cycle	119.8	1.2
	Other valves	14.1	5	Per Cycle	257.6	12.7
	Other mechanical components	1.0	5	Per Cycle	207.3	0.7
	Electrical components	0.6	3	Per Cycle	51.8	0.1
Ventilation and filtration system		3.9	8	Monthly	6.3	2.4
Insulation/shielding and support		8.2	3	Monthly	3.0	0.9
Miscellaneous works		5.5	7	Daily	1.6	22.5
Total						145.6

(1) In order to present statistical information on routine and non-routine activities, values for these parameters (number of workers, frequency and exposure time) are adjusted so that the multiplication of these parameters by the average dose rate equals the annual ORE dose, which is the measured value at Hanul Unit 3.

(2) For the cases of 'Per Cycle', an approximate frequency of 0.7 per year is used to calculate the annual ORE dose, since a fuel cycle of 18 months is applied.

Table 12.4-3

Occupational Dose Estimates During Inservice Inspection

Activity	Average Dose Rate [μSv/hr]	Number of Workers (1)	Frequency (1), (2)	Exposure Time (1) [hr]	Annual ORE Dose [mSv/yr]
Inspections of components and systems	26.1	7	Per Cycle	171.3	21.9
Inspection of welds	39.3	3	Per Cycle	15.4	1.3
ISI support (3)	29.0	5	Per Cycle	112.4	11.4
Total					34.6

(1) In order to present statistical information on routine and non-routine activities, values for these parameters (number of workers, frequency and exposure time) are adjusted so that the multiplication of these parameters by the average dose rate equals the annual ORE dose, which is the measured value at Hanul Unit 3.

(2) For the cases of ‘Per Cycle’, an approximate frequency of 0.7 per year is used to calculate the annual ORE dose, since a fuel cycle of 18 months is applied.

(3) Activities such as installation and removal of platforms, ladders, thermal insulation, etc. and preparation of work, etc.

Table 12.4-4

Occupational Dose Estimates During Special Maintenance

Activity		Average Dose Rate [μSv/hr]	Number of Workers (1)	Frequency (1), (2)	Exposure Time (1) [hr]	Annual ORE Dose [mSv/yr]
Maintenance of components	RCS pump	20.8	5	Per Cycle	161.2	11.7
	Primary circuit	47.8	5	Per Cycle	228.1	38.2
	Other pumps	5.1	5	Per Cycle	4.0	0.1
	Other valves	17.0	5	Per Cycle	42.0	2.5
	Other mechanical components	0.8	5	Per Cycle	52.3	0.1
	Electric components	0.5	4	Per Cycle	5.6	0.0
	Piping	53.1	4	Per Cycle	12.8	1.9
Maintenance of pressurizer		42.1	10	Per Cycle	37.2	11.0
Maintenance of SG primary side	Man-way opening/closure	70.2	5	Per Cycle	18.3	4.5
	Nozzle dam activities	231.5	5	Per Cycle	16.7	13.5
	Tube inspection	51.2	5	Per Cycle	154.2	27.6
	Tube repairs	33.1	5	Per Cycle	243.4	28.2
	Other works	31.3	10	Per Cycle	138.9	30.4
Maintenance of SG secondary side	Man-way opening/closure	17.4	5	Per Cycle	23.3	1.4
	Inspection & cleaning	25.2	8	Per Cycle	68.7	9.7
	Other works	1.5	10	Per Cycle	819.1	8.6
Servicing of control rod drive mechanism		17.1	3	Per Cycle	265.8	9.5
Total						199.0

(1) In order to present statistical information on routine and non-routine activities, values for these parameters (number of workers, frequency and exposure time) are adjusted so that the multiplication of these parameters by the average dose rate equals the annual ORE dose, which is the measured value at Hanul Unit 3.

(2) For the cases of 'Per Cycle', an approximate frequency of 0.7 per year is used to calculate the annual ORE dose, since a fuel cycle of 18 months is applied.

Table 12.4-5

Occupational Dose Estimates During Waste Processing

Activity	Average Dose Rate [μSv/hr]	Number of Workers (1)	Frequency (1)	Exposure Time (1) [hr]	Annual ORE Dose [mSv/yr]
System operation, inspection and testing	1.7	1	Weekly	0.7	0.1
Sampling and analysis	0.5	1	Daily	3.0	0.5
Operation of waste processing and packaging equipment	4.7	3	Weekly	7.6	5.4
Total					6.0

(1) In order to present statistical information on routine and non-routine activities, values for these parameters (number of workers, frequency and exposure time) are adjusted so that the multiplication of these parameters by the average dose rate equals the annual ORE dose, which is the measured value at Hanul Unit 3.

Table 12.4-6

Occupational Dose Estimates During Refueling

Activity		Average Dose Rate [μSv/hr]	Number of Workers (1)	Frequency (1), (2)	Exposure Time (1) [hr]	Annual ORE Dose [mSv/yr]
Works related to RPV	RPV head removal and installation	36.9	10	Per Cycle	170.0	43.9
	Inspection of RPV & reactor internals	59.0	5	Per Cycle	169.3	35.0
	In-core and Ex-core instrumentation	55.6	5	Per Cycle	66.6	13.0
	Inspection and works related threaded holes	70.7	5	Per Cycle	71.0	17.6
	Reactor clean-up	3.3	5	Per Cycle	2.7	< 0.1
	Other works	13.0	15	Per Cycle	209.0	28.5
Works related to fuel handling	Refueling preparation	13.9	3	Per Cycle	77.1	2.3
	Fuel withdrawal	6.0	5	Per Cycle	125.8	2.6
	Fuel inspection	1.8	5	Per Cycle	147.7	0.9
	Fuel loading	6.1	5	Per Cycle	84.6	1.8
	Refueling machine inspection & maintenance	34.2	5	Per Cycle	120.1	14.4
	Other works	4.3	10	Per Cycle	108.1	3.3
Total						163.2

(1) In order to present statistical information on routine and non-routine activities, values for these parameters (number of workers, frequency and exposure time) are adjusted so that the multiplication of these parameters by the average dose rate equals the annual ORE dose, which is the measured value at Hanul Unit 3.

(2) For the cases of 'Per Cycle', an approximate frequency of 0.7 per year is used to calculate the annual ORE dose, since a fuel cycle of 18 months is applied.

Table 12.4-7

Annual Personnel Doses per Activity Categories

Category of Activity	Fraction [%]	Estimated Annual ORE Dose [man·mSv]
Reactor operations and surveillance	6.3%	36.6
Routine maintenance	24.9%	145.6
Inservice inspection	5.9%	34.6
Special maintenance	34.0%	199.0
Waste processing	1.0%	6.0
Refueling	27.9%	163.2
Total		585.0

Table 12.4-8 (1 of 21)

Estimated Accident Mission Dose

a. The vital area mission doses and dose rates for areas requiring continuous occupancy to perform post-accident vital functions

Area Requiring Continuous Occupancy	Dose (mSv)	Dose Rate (mSv/hr)	Regulatory Limit	
			Dose (mSv)	Dose Rate (mSv/hr)
MCR	46.9	0.148	50	0.15
TSC	46.9	0.148	50	0.15

b. The vital area mission doses and dose rates for access to areas requiring continuous occupancy to perform post-accident vital functions

Vital Area	Action	Post-LOCA Time (hr)	Mission Dose			Access Route
			Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
MCR/TSC	Access from Outside (Comp. Bldg)(1)	1	1.06	5.44	0.10	Fig. 12.3-13,28,40
		2	5.23	5.44	0.47	
		4	6.73	5.44	0.61	
		8	4.09	5.44	0.37	
		24	0.92	5.44	0.08	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13

Table 12.4-8 (2 of 21)

c. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (1 hour after LOCA) (1 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Post-accident Sample and Control Panel Room (055-A48B, A49B)	Access from Outside (Comp. Bldg)(1)	37.51	6.07	3.79	Fig. 12.3-13,20,28
	Sampling	1.43	20	0.48	
	Sum			4.27	
Radiochemistry Lab (085-P37)	Access from Post-accident Sample and Control Panel Room	35.24	6.47	3.80	Fig. 12.3-12,13,20,28
	Analysis	0.91	60	0.91	
	Return to MCR/TSC	1.06	6.45	0.11	
	Sum			4.83	
Sample Counting Room (085-P36)	Access from Post-accident Sample and Control Panel Room	33.63	6.79	3.81	Fig. 12.3-12,13,20,28
	Analysis	0.91	60	0.91	
	Return to MCR/TSC	1.06	6.77	0.12	
	Sum			4.84	
Remote Shutdown Room (137-A06D)	Access from Outside (Comp. Bldg)(1)	1.06	5.78	0.10	Fig. 12.3-13,28,36
	Operation	0.91	90	1.37	
	Return to Outside (Aux. Bldg)(2)	1.06	3.35	0.06	
	Sum			1.53	
Remote Control Console Room (137-A41A)	Access from MCR/TSC	1.24	6.12	0.13	Fig. 12.3-32,36,40
	Operation	0.91	90	1.37	
	Return to MCR/TSC	1.24	6.12	0.13	
	Sum			1.62	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28

Table 12.4-8 (3 of 21)

c. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (1 hour after LOCA) (2 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Class 1E Switchgear 01A Room (078-A25A)	Access from MCR/TSC	1.91	6.35	0.20	Fig. 12.3-13,24,28,40
	Inspection	136.51	10	22.75	
	Return to Outside (Comp. Bldg)(1)	10.39	2.78	0.48	
	Sum			23.44	
Class 1E Switchgear 01B Room (078-A25B)	Access from MCR/TSC	1.91	6.31	0.20	Fig. 12.3-24,28,40
	Inspection	139.63	10	23.27	
	Return to Outside (Aux. Bldg)(2)	13.19	2.03	0.45	
	Sum			23.92	
Class 1E Switchgear 01C Room (078-A02C)	Access from MCR/TSC	2.04	5.30	0.18	Fig. 12.3-24,40
	Inspection	0.91	10	0.15	
	Return to MCR/TSC	2.04	5.30	0.18	
	Sum			0.51	
Class 1E Switchgear 01D Room (078-A02D)	Access from MCR/TSC	2.04	5.26	0.18	Fig. 12.3-24,40
	Inspection	0.91	10	0.15	
	Return to MCR/TSC	2.04	5.26	0.18	
	Sum			0.51	
I&C Equip. Room (157-A19C)	Access from Outside (Comp. Bldg)(3)	1.06	5.38	0.10	Fig. 12.3-13,28,40
	Inspection	18.40	30	9.20	
	Return to Outside (Aux. Bldg)(4)	1.06	4.10	0.07	
	Sum			9.37	

(1) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-28
(3) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(4) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28

Table 12.4-8 (4 of 21)

c. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (1 hour after LOCA) (3 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
I&C Equip. Room (157-A25C)	Access from Outside (Comp. Bldg)(1)	1.06	5.31	0.09	Fig. 12.3-13,28,40
	Inspection	0.91	30	0.46	
	Return to Outside (Aux. Bldg)(2)	1.06	4.04	0.07	
	Sum			0.62	
I&C Equip. Room (157-A01D)	Access from Outside (Comp. Bldg)(1)	1.06	6.29	0.11	Fig. 12.3-13,28,40
	Inspection	0.91	30	0.46	
	Return to Outside (Aux. Bldg)(3)	1.06	3.87	0.07	
	Sum			0.64	
I&C Equip. Room (157-A19D)	Access from Outside (Comp. Bldg)(1)	1.06	6.52	0.12	Fig. 12.3-13,28,40
	Inspection	18.40	30	9.20	
	Return to Outside (Aux. Bldg)(3)	1.06	4.09	0.07	
	Sum			9.39	
Access Area outside the CS Pump Room (050-A01C)	Access from MCR/TSC	17.62	9.41	2.76	Fig. 12.3-13,20,28,40
	Operation	1.74	30	0.87	
	Return to Outside (Comp. Bldg)(4)	32.71	4.93	2.69	
	Sum			6.32	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28
(3) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28
(4) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (5 of 21)

c. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (1 hour after LOCA) (4 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Access Area outside the CS Pump Room (050-A01D)	Access from Outside (Comp. Bldg)(1)	35.53	6.47	3.83	Fig. 12.3-13,20,28
	Operation	1.74	30	0.87	
	Return to Outside (Aux. Bldg)(2)	37.09	4.11	2.54	
	Sum			7.24	
Access Area outside the SC Pump Room (050-A04A)	Access from MCR/TSC	18.72	8.77	2.74	Fig. 12.3-13,20,28,40
	Operation	1.74	30	0.87	
	Return to Outside (Comp. Bldg)(3)	37.23	4.28	2.66	
	Sum			6.26	
Access Area outside the SC Pump Room (050-A04B)	Access from Outside (Comp. Bldg)(1)	39.17	5.82	3.80	Fig. 12.3-13,20,28
	Operation	1.74	30	0.87	
	Return to Outside (Aux. Bldg)(2)	43.98	3.43	2.51	
	Sum			7.18	

(1) Section within the row line from PE to PF along the column line 31 in figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-28
(3) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (6 of 21)

d. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (2 hours after LOCA) (1 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Post-accident Sample and Control Panel Room (055-A48B, A49B)	Access from Outside (Comp. Bldg)(1)	41.67	6.07	4.22	Fig. 12.3-13,20,28
	Sampling	5.60	20	1.87	
	Sum			6.08	
Radiochemistry Lab (085-P37)	Access from Post-accident Sample and Control Panel Room	39.40	6.47	4.25	Fig. 12.3-12,13,20,28
	Analysis	5.08	60	5.08	
	Return to MCR/TSC	5.23	6.45	0.56	
	Sum			9.89	
Sample Counting Room (085-P36)	Access from Post-accident Sample and Control Panel Room	37.79	6.79	4.28	Fig. 12.3-12,13,20,28
	Analysis	5.08	60	5.08	
	Return to MCR/TSC	5.23	6.77	0.59	
	Sum			9.94	
Remote Shutdown Room (137-A06D)	Access from Outside (Comp. Bldg)(1)	5.23	5.78	0.50	Fig. 12.3-13,28,36
	Operation	5.08	90	7.61	
	Return to Outside (Aux. Bldg)(2)	5.23	3.35	0.29	
	Sum			8.41	
Remote Control Console Room (137-A41A)	Access from MCR/TSC	6.23	6.12	0.64	Fig. 12.3-32,36,40
	Operation	5.08	90	7.61	
	Return to MCR/TSC	6.23	6.12	0.64	
	Sum			8.89	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28

Table 12.4-8 (7 of 21)

d. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (2 hours after LOCA) (2 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Class 1E Switchgear 01A Room (078-A25A)	Access from MCR/TSC	6.07	6.35	0.64	Fig. 12.3-13,24,28,40
	Inspection	140.68	10	23.45	
	Return to Outside (Comp. Bldg)(1)	14.55	2.78	0.68	
	Sum			24.76	
Class 1E Switchgear 01B Room (078-A25B)	Access from MCR/TSC	6.08	6.31	0.64	Fig. 12.3-24,40
	Inspection	143.80	10	23.97	
	Return to MCR/TSC	6.08	6.31	0.64	
	Sum			25.24	
Class 1E Switchgear 01C Room (078-A02C)	Access from MCR/TSC	6.20	5.30	0.55	Fig. 12.3-24,40
	Inspection	5.08	10	0.85	
	Return to MCR/TSC	6.20	5.30	0.55	
	Sum			1.94	
Class 1E Switchgear 01D Room (078-A02D)	Access from MCR/TSC	6.21	5.26	0.54	Fig. 12.3-24,40
	Inspection	5.08	10	0.85	
	Return to MCR/TSC	6.21	5.26	0.54	
	Sum			1.94	
I&C Equip. Room (157-A19C)	Access from Outside (Comp. Bldg)(2)	5.23	5.38	0.47	Fig. 12.3-13,28,40
	Inspection	22.57	30	11.28	
	Return to Outside (Aux. Bldg)(3)	5.23	4.10	0.36	
	Sum			12.11	

(1) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13
(2) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(3) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28

Table 12.4-8 (8 of 21)

d. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (2 hours after LOCA) (3 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
I&C Equip. Room (157-A25C)	Access from Outside (Comp. Bldg)(1)	5.23	5.31	0.46	Fig. 12.3-13,28,40
	Inspection	5.08	30	2.54	
	Return to Outside (Aux. Bldg)(2)	5.23	4.04	0.35	
	Sum			3.35	
I&C Equip. Room (157-A01D)	Access from Outside (Comp. Bldg)(1)	5.23	6.29	0.55	Fig. 12.3-13,28,40
	Inspection	5.08	30	2.54	
	Return to Outside (Aux. Bldg)(3)	5.23	3.87	0.34	
	Sum			3.42	
I&C Equip. Room (157-A19D)	Access from Outside (Comp. Bldg)(1)	5.23	6.52	0.57	Fig. 12.3-13,28,40
	Inspection	22.57	30	11.28	
	Return to Outside (Aux. Bldg)(3)	5.23	4.09	0.36	
	Sum			12.21	
Access Area outside the CS Pump Room (050-A01C)	Access from MCR/TSC	21.79	9.41	3.42	Fig. 12.3-13,20,28,40
	Operation	5.90	30	2.95	
	Return to Outside (Comp. Bldg)(4)	36.87	4.93	3.03	
	Sum			9.40	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28
(3) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28
(4) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (9 of 21)

d. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (2 hours after LOCA) (4 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Access Area outside the CS Pump Room (050-A01D)	Access from Outside (Comp. Bldg)(1)	39.69	6.47	4.28	Fig. 12.3-13,20,28
	Operation	5.90	30	2.95	
	Return to Outside (Aux. Bldg)(2)	41.25	4.11	2.83	
	Sum			10.06	
Access Area outside the SC Pump Room (050-A04A)	Access from MCR/TSC	22.89	8.77	3.34	Fig. 12.3-13,20,28,40
	Operation	5.90	30	2.95	
	Return to Outside (Comp. Bldg)(3)	41.40	4.28	2.95	
	Sum			9.25	
Access Area outside the SC Pump Room (050-A04B)	Access from Outside (Comp. Bldg)(1)	43.33	5.82	4.21	Fig. 12.3-13,20,28
	Operation	5.90	30	2.95	
	Return to Outside (Aux. Bldg)(2)	48.15	3.43	2.75	
	Sum			9.91	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-28
(3) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (10 of 21)

e. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (4 hours after LOCA) (1 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Post-accident Sample and Control Panel Room (055-A48B, A49B)	Access from Outside (Comp. Bldg)(1)	43.18	6.07	4.37	Fig. 12.3-13,20,28
	Sampling	7.11	20	2.37	
	Sum			6.74	
Radiochemistry Lab (085-P37)	Access from Post-accident Sample and Control Panel Room	40.91	6.47	4.41	Fig. 12.3-12,13,20,28
	Analysis	6.58	60	6.58	
	Return to MCR/TSC	6.73	6.45	0.72	
	Sum			11.72	
Sample Counting Room (085-P36)	Access from Post-accident Sample and Control Panel Room	39.30	6.79	4.45	Fig. 12.3-12,13,20,28
	Analysis	6.58	60	6.58	
	Return to MCR/TSC	6.73	6.77	0.76	
	Sum			11.79	
Remote Shutdown Room (137-A06D)	Access from Outside (Comp. Bldg)(1)	6.73	5.78	0.65	Fig. 12.3-13,28,36
	Operation	6.58	90	9.88	
	Return to Outside (Aux. Bldg)(2)	6.73	3.35	0.38	
	Sum			10.90	
Remote Control Console Room (137-A41A)	Access from MCR/TSC	7.74	6.12	0.79	Fig. 12.3-32,36,40
	Operation	6.58	90	9.88	
	Return to MCR/TSC	7.74	6.12	0.79	
	Sum			11.45	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28

Table 12.4-8 (11 of 21)

e. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (4 hours after LOCA) (2 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Class 1E Switchgear 01A Room (078-A25A)	Access from MCR/TSC	7.58	6.35	0.80	Fig. 12.3-24,40
	Inspection	142.18	10	23.70	
	Return to MCR/TSC	7.58	6.35	0.80	
	Sum			25.30	
Class 1E Switchgear 01B Room (078-A25B)	Access from MCR/TSC	7.58	6.31	0.80	Fig. 12.3-24,40
	Inspection	145.30	10	24.22	
	Return to MCR/TSC	7.58	6.31	0.80	
	Sum			25.81	
Class 1E Switchgear 01C Room (078-A02C)	Access from MCR/TSC	7.71	5.30	0.68	Fig. 12.3-24,40
	Inspection	6.58	10	1.10	
	Return to MCR/TSC	7.71	5.30	0.68	
	Sum			2.46	
Class 1E Switchgear 01D Room (078-A02D)	Access from MCR/TSC	7.71	5.26	0.68	Fig. 12.3-24,40
	Inspection	6.58	10	1.10	
	Return to MCR/TSC	7.71	5.26	0.68	
	Sum			2.45	
I&C Equip. Room (157-A19C)	Access from Outside (Comp. Bldg)(1)	6.73	5.38	0.60	Fig. 12.3-13,28,40
	Inspection	24.07	30	12.04	
	Return to Outside (Aux. Bldg)(2)	6.73	4.10	0.46	
	Sum			13.10	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28

Table 12.4-8 (12 of 21)

e. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (4 hours after LOCA) (3 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
I&C Equip. Room (157-A25C)	Access from Outside (Comp. Bldg)(1)	6.73	5.31	0.60	Fig. 12.3-13,28,40
	Inspection	6.58	30	3.29	
	Return to Outside (Aux. Bldg)(2)	6.73	4.04	0.45	
	Sum			4.34	
I&C Equip. Room (157-A01D)	Access from Outside (Comp. Bldg)(1)	6.73	6.29	0.71	Fig. 12.3-13,28,40
	Inspection	6.58	30	3.29	
	Return to Outside (Aux. Bldg)(3)	6.73	3.87	0.43	
	Sum			4.43	
I&C Equip. Room (157-A19D)	Access from Outside (Comp. Bldg)(1)	6.73	6.52	0.73	Fig. 12.3-13,28,40
	Inspection	24.07	30	12.04	
	Return to Outside (Aux. Bldg)(3)	6.73	4.09	0.46	
	Sum			13.23	
Access Area outside the CS Pump Room (050-A01C)	Access from MCR/TSC	23.30	9.41	3.65	Fig. 12.3-13,20,28,40
	Operation	7.41	30	3.70	
	Return to Outside (Comp. Bldg)(4)	38.38	4.93	3.15	
	Sum			10.51	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28
(3) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28
(4) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (13 of 21)

e. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (4 hours after LOCA) (4 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Access Area outside the CS Pump Room (050-A01D)	Access from Outside (Comp. Bldg)(1)	41.20	6.47	4.44	Fig. 12.3-13,20,28
	Operation	7.41	30	3.70	
	Return to Outside (Aux. Bldg)(2)	42.76	4.11	2.93	
	Sum			11.08	
Access Area outside the SC Pump Room (050-A04A)	Access from MCR/TSC	24.39	8.77	3.56	Fig. 12.3-13,20,28,40
	Operation	7.41	30	3.70	
	Return to Outside (Comp. Bldg)(3)	42.90	4.28	3.06	
	Sum			10.33	
Access Area outside the SC Pump Room (050-A04B)	Access from Outside (Comp. Bldg)(1)	44.84	5.82	4.35	Fig. 12.3-13,20,28
	Operation	7.41	30	3.70	
	Return to Outside (Aux. Bldg)(2)	49.65	3.43	2.84	
	Sum			10.89	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-28
(3) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (14 of 21)

f. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (8 hours after LOCA) (1 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Post-accident Sample and Control Panel Room (055-A48B, A49B)	Access from Outside (Comp. Bldg)(1)	40.54	6.07	4.10	Fig. 12.3-13,20,28
	Sampling	4.47	20	1.49	
	Sum			5.59	
Radiochemistry Lab (085-P37)	Access from Post-accident Sample and Control Panel Room	38.27	6.47	4.13	Fig. 12.3-12,13,20,28
	Analysis	3.94	60	3.94	
	Return to MCR/TSC	4.09	6.45	0.44	
	Sum			8.51	
Sample Counting Room (085-P36)	Access from Post-accident Sample and Control Panel Room	36.66	6.79	4.15	Fig. 12.3-12,13,20,28
	Analysis	3.94	60	3.94	
	Return to MCR/TSC	4.09	6.77	0.46	
	Sum			8.56	
Remote Shutdown Room (137-A06D)	Access from Outside (Comp. Bldg)(1)	4.09	5.78	0.39	Fig. 12.3-13,28,36
	Operation	3.94	90	5.92	
	Return to Outside (Aux. Bldg)(2)	4.09	3.35	0.23	
	Sum			6.54	
Remote Control Console Room (137-A41A)	Access from MCR/TSC	5.09	6.12	0.52	Fig. 12.3-32,36,40
	Operation	3.94	90	5.92	
	Return to MCR/TSC	5.09	6.12	0.52	
	Sum			6.96	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28

Table 12.4-8 (15 of 21)

f. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (8 hours after LOCA) (2 of 4)

<table>
<tr><th rowspan="2">Vital Area
(Room No.)</th><th rowspan="2">Action</th><th colspan="3">Mission Dose</th><th rowspan="2">Access Route</th></tr>
<tr><th>Dose Rate (mSv/hr)</th><th>Occupancy Time (min)</th><th>Dose (mSv)</th></tr>
<tr><td rowspan="4">Class 1E Switchgear 01A Room (078-A25A)</td><td>Access from MCR/TSC</td><td>4.94</td><td>6.35</td><td>0.52</td><td rowspan="4">Fig. 12.3-13,24,28,40</td></tr>
<tr><td>Inspection</td><td>139.54</td><td>10</td><td>23.26</td></tr>
<tr><td>Return to Outside (Comp. Bldg)(1)</td><td>13.42</td><td>2.78</td><td>0.62</td></tr>
<tr><td colspan="3">Sum</td><td>24.40</td></tr>
<tr><td rowspan="4">Class 1E Switchgear 01B Room (078-A25B)</td><td>Access from MCR/TSC</td><td>4.94</td><td>6.31</td><td>0.52</td><td rowspan="4">Fig. 12.3-24,28,40</td></tr>
<tr><td>Inspection</td><td>142.66</td><td>10</td><td>23.78</td></tr>
<tr><td>Return to Outside (Aux. Bldg)(2)</td><td>16.22</td><td>2.03</td><td>0.55</td></tr>
<tr><td colspan="3">Sum</td><td>24.85</td></tr>
<tr><td rowspan="4">Class 1E Switchgear 01C Room (078-A02C)</td><td>Access from MCR/TSC</td><td>5.07</td><td>5.30</td><td>0.45</td><td rowspan="4">Fig. 12.3-24,40</td></tr>
<tr><td>Inspection</td><td>3.94</td><td>10</td><td>0.66</td></tr>
<tr><td>Return to MCR/TSC</td><td>5.07</td><td>5.30</td><td>0.45</td></tr>
<tr><td colspan="3">Sum</td><td>1.55</td></tr>
<tr><td rowspan="4">Class 1E Switchgear 01D Room (078-A02D)</td><td>Access from MCR/TSC</td><td>5.07</td><td>5.26</td><td>0.45</td><td rowspan="4">Fig. 12.3-24,40</td></tr>
<tr><td>Inspection</td><td>3.94</td><td>10</td><td>0.66</td></tr>
<tr><td>Return to MCR/TSC</td><td>5.07</td><td>5.26</td><td>0.45</td></tr>
<tr><td colspan="3">Sum</td><td>1.55</td></tr>
<tr><td rowspan="4">I&C Equip. Room (157-A19C)</td><td>Access from Outside (Comp. Bldg)(3)</td><td>4.09</td><td>5.38</td><td>0.37</td><td rowspan="4">Fig. 12.3-13,28,40</td></tr>
<tr><td>Inspection</td><td>21.43</td><td>30</td><td>10.72</td></tr>
<tr><td>Return to Outside (Aux. Bldg)(4)</td><td>4.09</td><td>4.10</td><td>0.28</td></tr>
<tr><td colspan="3">Sum</td><td>11.36</td></tr>
</table>

(1) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-28
(3) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(4) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28

Table 12.4-8 (16 of 21)

f. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (8 hours after LOCA) (3 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
I&C Equip. Room (157-A25C)	Access from Outside (Comp. Bldg)(1)	4.09	5.31	0.36	Fig. 12.3-13,28,40
	Inspection	3.94	30	1.97	
	Return to Outside (Aux. Bldg)(2)	4.09	4.04	0.28	
	Sum			2.61	
I&C Equip. Room (157-A01D)	Access from Outside (Comp. Bldg)(1)	4.09	6.29	0.43	Fig. 12.3-13,28,40
	Inspection	3.94	30	1.97	
	Return to Outside (Aux. Bldg)(3)	4.09	3.87	0.26	
	Sum			2.67	
I&C Equip. Room (157-A19D)	Access from Outside (Comp. Bldg)(1)	4.09	6.52	0.44	Fig. 12.3-13,28,40
	Inspection	21.43	30	10.72	
	Return to Outside (Aux. Bldg)(3)	4.09	4.09	0.28	
	Sum			11.44	
Access Area outside the CS Pump Room (050-A01C)	Access from MCR/TSC	20.65	9.41	3.24	Fig. 12.3-13,20,28,40
	Operation	4.77	30	2.38	
	Return to Outside (Comp. Bldg)(4)	35.74	4.93	2.93	
	Sum			8.56	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-28
(3) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-28
(4) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (17 of 21)

f. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (8 hours after LOCA) (4 of 4)

<table>
<tr><th rowspan="2">Vital Area
(Room No.)</th><th rowspan="2">Action</th><th colspan="3">Mission Dose</th><th rowspan="2">Access Route</th></tr>
<tr><th>Dose Rate (mSv/hr)</th><th>Occupancy Time (min)</th><th>Dose (mSv)</th></tr>
<tr><td rowspan="4">Access Area outside the CS Pump Room (050-A01D)</td><td>Access from Outside (Comp. Bldg)(1)</td><td>38.56</td><td>6.47</td><td>4.16</td><td rowspan="4">Fig. 12.3-13,20,28</td></tr>
<tr><td>Operation</td><td>4.77</td><td>30</td><td>2.38</td></tr>
<tr><td>Return to Outside (Aux. Bldg)(2)</td><td>40.12</td><td>4.11</td><td>2.75</td></tr>
<tr><td colspan="3">Sum</td><td>9.29</td></tr>
<tr><td rowspan="4">Access Area outside the SC Pump Room (050-A04A)</td><td>Access from MCR/TSC</td><td>21.75</td><td>8.77</td><td>3.18</td><td rowspan="4">Fig. 12.3-13,20,28,40</td></tr>
<tr><td>Operation</td><td>4.77</td><td>30</td><td>2.38</td></tr>
<tr><td>Return to Outside (Comp. Bldg)(3)</td><td>40.26</td><td>4.28</td><td>2.87</td></tr>
<tr><td colspan="3">Sum</td><td>8.44</td></tr>
<tr><td rowspan="4">Access Area outside the SC Pump Room (050-A04B)</td><td>Access from Outside (Comp. Bldg)(1)</td><td>42.20</td><td>5.82</td><td>4.10</td><td rowspan="4">Fig. 12.3-13,20,28</td></tr>
<tr><td>Operation</td><td>4.77</td><td>30</td><td>2.38</td></tr>
<tr><td>Return to Outside (Aux. Bldg)(2)</td><td>47.01</td><td>3.43</td><td>2.68</td></tr>
<tr><td colspan="3">Sum</td><td>9.17</td></tr>
</table>

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-28
(3) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (18 of 21)

g. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (24 hours after LOCA) (1 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Post-accident Sample and Control Panel Room (055-A48B, A49B)	Access from Outside (Comp. Bldg)(1)	4.21	6.07	0.43	Fig. 12.3-13,21,29
	Sampling	0.77	20	0.26	
	Sum			0.68	
Radiochemistry Lab (085-P37)	Access from Post-accident Sample and Control Panel Room	4.01	6.47	0.43	Fig. 12.3-12,13,21,29
	Analysis	0.77	60	0.77	
	Return to MCR/TSC	0.92	6.45	0.10	
	Sum			1.30	
Sample Counting Room (085-P36)	Access from Post-accident Sample and Control Panel Room	3.86	6.79	0.44	Fig. 12.3-12,13,21,29
	Analysis	0.77	60	0.77	
	Return to MCR/TSC	0.92	6.77	0.10	
	Sum			1.31	
Remote Shutdown Room (137-A06D)	Access from Outside (Comp. Bldg)(1)	0.92	5.78	0.09	Fig. 12.3-13,29,37
	Operation	0.77	90	1.15	
	Return to Outside (Aux. Bldg)(2)	0.92	3.35	0.05	
	Sum			1.29	
Remote Control Console Room (137-A41A)	Access from MCR/TSC	1.92	6.12	0.20	Fig. 12.3-33,37,41
	Operation	0.77	90	1.15	
	Return to MCR/TSC	1.92	6.12	0.20	
	Sum			1.54	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-29

Table 12.4-8 (19 of 21)

g. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (24 hours after LOCA) (2 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Class 1E Switchgear 01A Room (078-A25A)	Access from MCR/TSC	0.99	6.35	0.10	Fig. 12.3-25,41
	Inspection	10.99	10	1.83	
	Return to MCR/TSC	0.99	6.35	0.10	
	Sum			2.04	
Class 1E Switchgear 01B Room (078-A25B)	Access from MCR/TSC	0.99	6.31	0.10	Fig. 12.3-25,41
	Inspection	11.22	10	1.87	
	Return to MCR/TSC	0.99	6.31	0.10	
	Sum			2.08	
Class 1E Switchgear 01C Room (078-A02C)	Access from MCR/TSC	1.00	5.30	0.09	Fig. 12.3-25,41
	Inspection	0.77	10	0.13	
	Return to MCR/TSC	1.00	5.30	0.09	
	Sum			0.31	
Class 1E Switchgear 01D Room (078-A02D)	Access from MCR/TSC	1.00	5.26	0.09	Fig. 12.3-25,41
	Inspection	0.77	10	0.13	
	Return to MCR/TSC	1.00	5.26	0.09	
	Sum			0.30	
I&C Equip. Room (157-A19C)	Access from Outside (Comp. Bldg)(1)	0.92	5.38	0.08	Fig. 12.3-13,29,41
	Inspection	0.87	30	0.44	
	Return to Outside (Aux. Bldg)(2)	0.92	4.10	0.06	
	Sum			0.58	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-29

Table 12.4-8 (20 of 21)

g. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (24 hours after LOCA) (3 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
I&C Equip. Room (157-A25C)	Access from Outside (Comp. Bldg)(1)	0.92	5.31	0.08	Fig. 12.3-13,29,41
	Inspection	0.77	30	0.38	
	Return to Outside (Aux. Bldg)(2)	0.92	4.04	0.06	
	Sum			0.53	
I&C Equip. Room (157-A01D)	Access from Outside (Comp. Bldg)(1)	0.92	6.29	0.10	Fig. 12.3-13,29,41
	Inspection	0.77	30	0.38	
	Return to Outside (Aux. Bldg)(3)	0.92	3.87	0.06	
	Sum			0.54	
I&C Equip. Room (157-A19D)	Access from Outside (Comp. Bldg)(1)	0.92	6.52	0.10	Fig. 12.3-13,29,41
	Inspection	0.87	30	0.44	
	Return to Outside (Aux. Bldg)(3)	0.92	4.09	0.06	
	Sum			0.60	
Access Area outside the CS Pump Room (050-A01C)	Access from MCR/TSC	2.33	9.41	0.36	Fig. 12.3-13,21,29,41
	Operation	0.77	30	0.38	
	Return to Outside (Comp. Bldg)(4)	3.61	4.93	0.30	
	Sum			1.04	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 14 to 15 along the row line AA in Figure 12.3-29
(3) Section within the column line from 14 to 15 along the row line AK in Figure 12.3-29
(4) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-8 (21 of 21)

g. Vital area mission dose for areas requiring infrequent access to perform post-accident vital functions (24 hours after LOCA) (4 of 4)

Vital Area (Room No.)	Action	Mission Dose			Access Route
		Dose Rate (mSv/hr)	Occupancy Time (min)	Dose (mSv)	
Access Area outside the CS Pump Room (050-A01D)	Access from Outside (Comp. Bldg)(1)	4.01	6.47	0.43	Fig. 12.3-13,21,29
	Operation	0.77	30	0.38	
	Return to Outside (Aux. Bldg)(2)	3.98	4.11	0.27	
	Sum			1.09	
Access Area outside the SC Pump Room (050-A04A)	Access from MCR/TSC	2.43	8.77	0.35	Fig. 12.3-13,21,29,41
	Operation	0.77	30	0.38	
	Return to Outside (Comp. Bldg)(3)	4.01	4.28	0.29	
	Sum			1.02	
Access Area outside the SC Pump Room (050-A04B)	Access from Outside (Comp. Bldg)(1)	4.35	5.82	0.42	Fig. 12.3-13,21,29
	Operation	0.77	30	0.38	
	Return to Outside (Aux. Bldg)(2)	4.59	3.43	0.26	
	Sum			1.07	

(1) Section within the row line from PE to PF along the column line 31 in Figure 12.3-13
(2) Section within the column line from 23 to 24 along the row line AK in Figure 12.3-29
(3) Section within the row line from PI to PJ along the column line 31 in Figure 12.3-13

Table 12.4-9 (1 of 2)

APR1400 Plant Systems Requiring Full NRC RG 4.21 Evaluations

System Code	System Title	Sheet No. In Table 12.4-10
AF	Auxiliary Feedwater	50
AS	Auxiliary Steam	63
AT	Auxiliary Feedwater Pump Turbine	50
CA	Condenser Vacuum	55
CC	Component Cooling Water	24
CD	Condensate	47
CM	Containment Monitoring	73
CP	Condensate Polishing	52
CS	Containment Spray	12
CV	Chemical and Volume Control	18
CW	Circulating Water	57
DE	Radioactive Drain	39
DM	Miscellaneous Building Drain	69
DT	Turbine Generator Building Drain	69
ES	Extraction Steam	47
FC	Spent Fuel Pool Cooling and Cleanup	30
FT	Feedwater Pump Turbine	47
FW	Feedwater	47
GW	Gaseous Waste Management	32
HD	Heater Drain	47
HG	Containment Hydrogen Control	14
IW	In-Containment Water Storage	15

Table 12.4-9 (2 of 2)

System Code	System Title	Sheet No. In Table 12.4-10
MS	Main Steam	45
PR	Radiation Monitoring	71
PS	Process Sampling	42
PX	Primary Sampling	42
RC	Reactor Coolant	1
RG	Reactor Coolant Gas Vent	7
SC	Shutdown Cooling	10
SD	Steam Generator Blowdown	22
SI	Safety Injection	8
SX	Essential Service Water	28
TA	Main Turbine and Auxiliaries	45
VB	Compound Building HVAC	66
VF	Fuel Handling Area HVAC	66
VK	Auxiliary Building Controlled Area HVAC	66
VP	Reactor Containment Building HVAC	66
VQ	Reactor Containment Building Purge	66
WH	Turbine Generator Building Open Cooling Water	59
WT	Turbine Generator Building Closed Cooling Water	61
WV	Liquid Waste Management	34
WX	Solid Waste Management	36

Table 12.4-10 (1 of 75)

NRC RG 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste

System: Reactor Coolant System

Objective		SSC Control Measures[1] / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The RCS is designed in accordance with ASME Section III and proven nuclear power industry experience. The RCS is designed with an over-pressure protection system and other engineered safety features (ESF) to provide reasonable assurance of safe operation and mitigation of accident conditions, thus preventing unintended contamination. • The RCS components are designed to provide the first barrier to contain the radioactivity resulting from power operation and damaged fuel. These components are designed with proven materials and fabrication techniques, and are housed inside the reactor containment structure, which provides the second barrier. The basemat of the reactor containment structure is equipped with steel liners to prevent and minimize the spread of contamination from leaked reactor coolant. The in-core instrumentation (ICI) sump is provided at the basemat level to collect leaks, and the reactor containment sump is provided at the reactor vessel operating floor to collect leaks from the other components. The reactor floor is also lined with a steel layer to prevent contamination of the concrete structure. • The system design, including wetted parts of the reactor vessel, the steam generators, the pressurizer, the reactor coolant pumps, and the associated piping, is designed with stainless steel cladding or base material, and has welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The components are designed with material resistant to primary water stress corrosion cracking and reduced SG hot leg temperature to minimize the potential for stress corrosion and cracking. • The containment also includes the IRWST and a holdup volume tank with sufficient capacity to provide temporary storage of reactor coolant from anticipated operational occurrences and design basis accidents.	5.4.3

Table 12.4-10 (2 of 75)

Objective		SSC Control Measures[1] / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	• The RCS components are designed with multiple level, pressure, and temperature, and radiation detection instruments to provide reasonable assurance of safe operation of the SSCs, including the associated piping, and provide alarms to the operating personnel in the event of off-normal conditions. Releases from over-pressurization and leaks are collected in the reactor drain tank (RDT), the ICI sump, the containment sump, and the IRWST. A detailed description of the early leak detection instruments, locations, leak collections and pathways, and operation notifications for response actions is presented in Subsection 5.2.5. • Leak detection methods are segregated into two classifications: “unidentified leakages” and “Identified leakages”. Unidentified leakages are monitored by containment sump level, area particulate radiation detection, and containment atmosphere humidity instruments. Leakages are collected in sumps and a computer program is developed to estimate leakage rates based on flow measurements. This approach can detect 0.5 gpm leakage within 1 hour. Additionally, 19 locations within the containments are equipped with acoustic leak detection instruments where leakage is considered critical. See Subsection 5.2.5.1.1 for details. • Containment area radiation monitors, temperature and pressure monitors, sump levels and sump pump flow rates are programmed to provide leakage information and notification in the MCR for operator actions. • Acoustic leak monitoring instruments are installed in other critical areas within the reactor coolant pressure boundary, including in-core instrumentation nozzles at reactor lower head, SG man-way seals, reactor hot and cold leg welding areas, reactor coolant pumps shaft seal housings, pressurizer lower head area, and CEA nozzle areas near the reactor head. These leak monitoring instruments provide early detection of leak and alarm signals for operator actions when required. • The “identified leakages” for the primary RCS leak detection methods and locations, and other indications of reactor coolant leakage are summarized in Subsections 5.2.5.1.2. The leakage sources and instruments for early detection of reactor coolant leakage instruments are as follows: include SG tube and tubesheet leaks, pressurizer pilot operated safety relief valve, reactor coolant pump seals, valves, reactor vessel head flange seal, and SG tube and tubesheet leakages.	5.4.3

Table 12.4-10 (3 of 75)

Objective		SSC Control Measures[1] / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	(cont.)	• Steam generator leaks are monitored by N-16 monitors on each of the main steam lines. An increase in radioactivity, as indicated by the condenser vacuum vent effluent monitor, and SG blowdown monitor will reveal reactor coolant leakage through SG tubes to the secondary side. Routine analysis of the SG secondary side will also indicate leakage of reactor coolant into the secondary system. • The RCPs are designed with two mechanical seals and one mechanical vapor seal made of renewable silicon carbide rings. Each of these three seals can withstand full operating pressure. Any leakage from these seals is collected and routed to the RDT. An acoustic sensor is provided with each pump to detect any leak through each of the seal. • Temperatures downstream of the pilot-operated safety relief valves (POSRVs) are continuously monitored for early detection of leaks. • Leakage between the two metal O-rings that seal the reactor vessel head flange is routed through a leakoff line to the RDT. A normally closed, remotely activated isolation valve and a pressure indicator are installed in the leakoff line. Leakage from the reactor vessel head flange causes pressure in the leakoff line to rise. The pressure in the leakoff line is continuously monitored to detect the presence of a leak. Any leakage is bled off to the RDT by opening the isolation valve. • Other primary coolant leak detection methods and locations are discussed in Subsection 5.2.5.2.2.	
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• Each RCP is equipped with the oil collection system in accordance with NRC RG 1.189 that collect oil from non-welded parts. Leaked oil is collected and drained to an oil collection tank that can store the entire inventory. This design approach prevents the potential generation of mixed waste. • Shop welding of major piping to larger pieces is maximized to reduce the field welding. Piping safe-ends are provided for field welding connections to minimize the potential for leaking. • The SSCs are designed with life-cycle planning through the use of nuclear-industry proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation.	5.4.3

Table 12.4-10 (4 of 75)

Objective		SSC Control Measures[1] / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	(cont.)	• The RCS is designed with continuous letdown of reactor coolant for purification in the CVCS system and provisions of sampling to maintain reactor coolant chemistry and quality. Similarly, the steam generators are designed with continuous blowdown of dissolved and suspended solids on the secondary side for treatment and with provisions for sampling and analysis. This design approach minimizes the buildup of radioactivity and impurities, enhance safe operation, and reduces the decontamination and waste generation during decommissioning. • The RCS components are made of steel with low cobalt content as much as it is practicable. The approach minimizes waste generation. • Manual valves 2 inches and larger are designed with a double-packed stem and a lantern ring with an intermediate leak drain connection. The drains are routed to the drain system for collection and treatment. This design approach minimizes the spread of contamination and the contamination of the facility. • Sampling lines from the RCS are designed to be as short and straight as possible to minimize traps and pockets. Flow velocity is maintained in the turbulent region to prevent settling of suspended solids. Gaseous sampling lines are sloped upward to facilitate draining of condensate. This design approach minimizes accumulation of condensate and provides more accurate samples, and thus minimizing waste generation. Sampling lines are purged with adequate demineralized water or nitrogen gas in order to maintain quality of sampling. • The process piping containing primary coolant is properly designed to be the shortest distance between components, sized for turbulent flow, and laid out to minimize traps and erosion. This approach facilitates easier flow and with sufficient velocities to prevent settling of solids, thus reducing the decontamination needs and waste generation. • Utility connections are designed with a minimum of two barriers to prevent cross-contamination.	

Table 12.4-10 (5 of 75)

	Objective	SSC Control Measures[1] / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	• The top surface of the containment basemat is sloped and lined with a layer of steel to prevent contamination of the basemat by facilitating draining of leakage to the ICI sump and the containment sump. This layered approach facilitates easier decommissioning and minimize waste generation. • The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. • The RCS is designed with no embedded or buried piping, thus facilitating decommissioning planning and preventing unintended contamination.	5.4.3
5	Operations and Documentation	• The RCS is designed for remote and automatic operation with operator supervision. Adequate instrumentation, including level, flow, temperature, and pressure elements, radiation monitors, and acoustic leak detection instruments, is provided to monitor and control the operations to prevent undue interruption, and minimize the spread of contamination and waste generation. • The RCS is designed for a minimal amount of maintenance. The layout takes into account the necessary access, laydown areas, handling and lifting cranes, and the auxiliary equipment in the event that maintenance is required. • The RCS components are equipped with loose parts monitoring instruments and vibration monitoring instruments at specific locations. This design approach enhances equipment life and minimizes waste generation. • Areas adjacent to the RCS pressure boundary are designed to have adequate clearance to access for inservice inspections. Personnel access is provided for component maintenance and inspections. • The COL applicant is to prepare operational procedures and maintenance programs for leak detection and contamination control of the RCS (COL 5.4(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item (COL 5.4(2)).	5.4.3

Table 12.4-10 (6 of 75)

Objective		SSC Control Measures(1) / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The RCS is located inside the reactor containment building. Because of its location and associated safety design features, the potential for environmental contamination of soil and groundwater from reactor coolant is minimal. Therefore, RCS inclusion in the site radiological environmental monitoring program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection radiological of contamination.	5.4.3

(1) The “SSC Control Measures” consist of excerpts/summary from the referenced DCD sections.

Table 12.4-10 (7 of 75)

System: Reactor Coolant Gas Vent System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The RCGVS is designed to vent non-condensable gases from the pressurizer and the reactor vessel upper head and depressurizes the reactor coolant system in the event that the pressurizer main spray or auxiliary spray systems are unavailable during plant cooldown. The piping directs the vented gases to the IRWST or the reactor drain tank (RDT) and is sloped to facilitate the drainage of condensation, thus minimizing leakage and unintended contamination of the facility and the environment.	5.4.12
2	Adequate and Early Leak Detection	• The RCGVS is designed not to be used during normal operation, and the piping is designed to slope downward to drain to the RDT and the IRWST. The potential for leakage is very low and a leak detection system is not required.	5.4.12
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The RCGVS piping is made of stainless steel material for life-cycle planning. The material is compatible with the chemical, physical, and radiological environment, thus minimizing waste generation.	5.4.12
4	Decommissioning Planning	• The RCGVS piping is designed for the full service life of the plant and is accessible for easy removal during decommissioning.	5.4.12
5	Operations and Documentation	• The RCGVS piping is used during refueling or post-accident conditions. Adequate instrumentation is provided to indicate the venting operations to prevent undue interruption.	5.4.12
6	Site Radiological Environmental Monitoring	• The RCGVS piping is part of the overall plant and is designed to vent non-condensable gases from the reactor vessel and the pressurizer. The RCGVS piping does not generate any radioactive materials and thus, site radiological environmental monitoring is not required.	5.4.12

Table 12.4-10 (8 of 75)

System: Safety Injection System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The SIS components are located inside the reactor containment building and the auxiliary building. The cubicle floors, where the components are located, are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs and sumps. This design approach prevents unintended contamination of the facility and the environment. • The system tanks and pumps are fabricated from stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment.	6.3.6
2	Adequate and Early Leak Detection	• The SI tanks are equipped with level instruments to monitor liquid levels. The level instruments provide alarm signals in the event that leakage occurs. • The SIS is not used during normal power operation. Any leakage is drained to the floor and is collected in the local sump which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator actions to investigate the source of leakage.	6.3.6
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning using nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The pump shaft seals are of the mechanical type and constructed of materials compatible with the SI fluid. This design approach minimizes contamination of the facility and the environment. The SIS system is designed to be isolated from the RCS during normal power operation to minimize cross-contamination of systems.	6.3.6
4	Decommissioning Planning	• Design features such as welding techniques used and surface finishes are intended to minimize the need for decontamination and the resultant waste generation. The SIS is designed with a minimal amount of embedded piping for contaminated or potentially contaminated fluid, to facilitate decommissioning.	6.3.6
5	Operations and Documentation	• The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control (COL 6.3(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations (COL 6.3(2)).	6.3.6

Table 12.4-10 (9 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The SIS is on standby mode during normal power operation and is designed to have low levels of contamination. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the SIS is well maintained, resulting in a very low potential for contamination of the facility. Because of its location, the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Hence, the SIS is not required to be part of the site radiological environmental monitoring program.	6.3.6

Table 12.4-10 (10 of 75)

System: Shutdown Cooling System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The SCS components are located in individual cubicles inside the auxiliary building. The floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs and sumps. This design approach prevents unintended contamination of the facility and the environment. • The SCS components (pumps, heat exchangers, piping) are fabricated from stainless steel material and are of welded construction, thus minimizing leakage and unintended contamination of the facility and the environment.	5.4.7
2	Adequate and Early Leak Detection	• The SCS is used only in post-shutdown periods. Any leakage is drained to the floor and is collected in the local sump, which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator actions to investigate the source of leakage.	5.4.7
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The pumps are equipped with drains directly routed to radioactive drain system. This design approach minimizes contamination to the facility and the environment. • The SCS is provided with isolation from the RCS when the RCS is at high pressure (during normal power operation). The interlocks associated with six valves on the two SCS suction lines are provided to prevent the valves from opening in the event RCS pressure exceeds SCS operating pressure, thus minimizing cross-contamination between the systems.	5.4.7
4	Decommissioning Planning	• The SSCs are designed for extended service life and are fabricated as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities. Design features, such as the welding techniques used, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation. • The SCS is designed with minimal embedded piping for contaminated or potentially contaminated fluid, which minimizes the potential for unintended contamination of the environment.	5.4.7

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item.	5.4.7
6	Site Radiological Environmental Monitoring	• The SCS is on standby mode during normal power operation and is designed to have low levels of contamination. Through monitoring, inservice inspection, and lessons learned from industry experiences, the integrity of the SCS is well maintained, resulting in a very low level of contamination of the facility. Hence, the SCS is not required to be part of the site radiological environmental monitoring program.	5.4.7

Table 12.4-10 (12 of 75)

System: Containment Spray System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The CSS components are located in individual cubicles inside the auxiliary building. The cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents unintended contamination of the facility and the environment. • The system heat exchangers and pumps are fabricated with stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The pump shaft seal is the mechanical type and is constructed of materials compatible with the spray fluids. A flow restrictor is provided to minimize the loss of fluid in the event of a seal failure. This design approach minimizes contamination to the facility and the environment.	6.5.2
2	Adequate and Early Leak Detection	• The CSS is not used during normal power operation. Any leakage is drained to the floor and is collected in the local sump, which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator actions to investigate the source of leakage.	6.5.2
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • Shell and tube type heat exchangers are used to transfer heat from the CSS side to the component cooling water system side. The heat exchangers are designed with stainless steel tubes to minimize the potential for cross-contamination between containment spray and component cooling water. The leakage from the heat exchangers is collected in the local floor drain and sumps. • Process sampling connections are provided to determine levels of contamination and determine treatment requirements.	6.5.2

Table 12.4-10 (13 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	• The SSCs are designed for the full service life and fabricated as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities. Design features (e.g., utilized welding technique, surface finishes) are included to minimize the need for decontamination and the resultant waste generation. • The CSS is designed without any embedded or buried piping for contaminated or potentially contaminated fluid, which minimizes the potential for unintended contamination of the environment.	6.5.2
5	Operations and Documentation	• The COL applicant is to provide the operational procedures and maintenance program as related to leak detection and contamination control (COL 6.5(1)). • The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations (COL 6.5(2)).	6.5.2
6	Site Radiological Environmental Monitoring	• The CSS is on standby mode during normal power operation and designed to have only low levels of contamination. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the CSS is well maintained, resulting in a very low level of contamination of the facility. Because of its location, the potential for environmental contamination of the soil and the groundwater from liquid leakage is minimal. Hence, the CSS is not required to be part of the Site Radiological Environmental Monitoring Program.	6.5.2

Table 12.4-10 (14 of 75)

System: Containment Hydrogen Control System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The passive autocatalytic recombiners (PARs) and hydrogen igniters (HIs) are designed to control the hydrogen by hydrogen recombination and hydrogen burning, respectively. There is no piping associated with the PARs and the HIs, and the system has no interfaces with non-contaminated systems, except electrical power supply, thus minimizing leakage and unintended contamination of the facility and the environment.	6.2.5
2	Adequate and Early Leak Detection	• All components of the containment hydrogen control system are inside the containment or the IRWST and the components operate in atmospheric conditions. There is no leakage from the components and a leak detection system is not required.	6.2.5
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The PARs and the HIs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The PARs and the HIs are located inside containment and the IRWST and are not in contact with contaminated liquid. Hence decontamination by water is not required.	6.2.5
4	Decommissioning Planning	• The PARs and the HIs are modular units and are designed for the full service life of the plant. The PARs and the HIs are fabricated as individual assemblies for easy removal to the maximum extent practicable.	6.2.5
5	Operations and Documentation	• The PARs are self-actuated and the HIs are actuated manually from the MCR or RSR with adequate instrumentation to the control the operation. This controls the hydrogen concentration inside the containment and minimizes the potential for fire and explosion hazards that could result in the spread of contamination and waste generation.	6.2.5
6	Site Radiological Environmental Monitoring	• The containment hydrogen control system is part of the overall plant and is designed to prevent the buildup of hydrogen concentrations. The containment hydrogen control system does not generate any radioactive materials and is thus not required to be included in site radiological environmental monitoring.	6.2.5

Table 12.4-10 (15 of 75)

System: In-Containment Water Storage System

	Objective	SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The IRWST components, primarily the IRWST and the HVT located inside containment, are constructed of reinforced concrete lined with austenitic stainless steel. The IRWST and the HVT are interconnected by two spillways. The bottom of the spillway is submerged below the surface of the water in the IRWST and the top of the spillway is below the IRWST ceiling during normal operation. This limits the amount of water vapor and possible gaseous radioactivity that can enter the containment atmosphere. This design approach prevents unintended contamination of the facility and the environment. • The top of the basemat inside the containment is provided with a steel liner. This design approach prevents unintended contamination of the basemat and the environment. • The IRWST uses the SFP cooling and cleanup system (FC) components to maintain and purify the water contents, thus minimizing the spread of contamination and unintended contamination of the facility and the environment. • The system piping is fabricated from stainless steel to minimize corrosion and erosion. Granulated tri-sodium phosphate (TSP) is provided in a stainless steel basket to minimize the corrosion of the stainless steel piping during LOCA conditions. • The IRWST sump pits are designed with steel liner plates to provide impermeable barriers to prevent contamination of the basemat.	6.8.2
2	Adequate and Early Leak Detection	• The IRWST is designed to prevent the spread of contamination due to overflow. Four wide range (0 to 100 percent) level indicating channels are provided locally and in the MCR. Adequate instrumentation, including temperature and pressure elements, is provided to monitor and control the operations of preventing overpressurization or vacuum, thus minimizing the spread of contamination and waste generation • The HVT has one narrow range and four wide range level indicating channels. The narrow range level channel is used to detect the presence of fluid in the HVT. Upon detection of water, the narrow range channel actuates an alarm in the MCR to alert for operator actions. • The reactor cavity has four wide range level indicating channels. The four wide range channels provide level indication in the MCR to facilitate monitoring fluid level after an accident.	6.8.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • During normal operation, the IRWST water can be cooled by the SC heat exchangers and purified by the FC system. The CVCS provides a means to add makeup and adjust the boron concentration in the IRWST. This design approach minimizes the spread of contamination to the facility and the environment, and waste generation.	6.8.2
4	Decommissioning Planning	• The SSCs are designed for the full service life. All structures inside the IRWST are simple and smooth in contour to facilitate its lining with stainless steel. The bottom of the tank is flat to avoid having sharp corners. This design approach minimizes potential for damages to the liner, which can result in unintended leakage and contamination. • The SSCs are designed with smooth surface finishes to minimize the need for decontamination and hence waste generation during decommissioning.	6.8.2
5	Operations and Documentation	• The IRWST is designed with adequate instrumentation for remote operation and monitoring. • A maintenance access hatch is provided to facilitate inspections and the maintenance of spargers and other hardware inside the tank. • The COL applicant is to provide the operational procedures and maintenance program related to leak detection and contamination control (COL 6.8(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to provide the preparation of cleanliness, housekeeping, and foreign materials exclusion program (COL 6.8(2)). This program addresses other debris sources, such as latent debris, inside containment. This program minimizes foreign materials in the containment. • The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations (COL 6.8(3)). Documentation requirements are included as a COL Information item.	6.8.2

Table 12.4-10 (17 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The IRWST is designed to provide safety features for uninterrupted operation of the plant. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the IRWST is well maintained, and contributes a very low level of contamination to the facility. Because of its location inside containment, the potential for environmental contamination of soil and groundwater from liquid leakages is minimized. Hence, this system is not required to be included in the site radiological environmental monitoring program.	6.8.2

Table 12.4-10 (18 of 75)

System: Chemical and Volume Control System

	Objective	SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system design, including the VCT, the EDT, and the letdown purification equipment inside the auxiliary building (AB); the RDT located inside the reactor containment building; and the holdup tank, BAST, and RMWT located outside the auxiliary building are designed with stainless steel and carbon steel materials that are compatible with the chemical and radiological environment. All components are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The CVCS is designed with provisions to collect drainage from system components, including pump seal leaks, valve leaks, and relief valve leakages, for purification and recycle. This design approach minimizes unintended contamination of the facility and the environment. • The heat exchangers inside the containment are designed to minimize the potential for leakage by using shell-and-tube type designs that comply with ASME Section III. • The piping tunnel connection from the yard tanks to the AB is designed to seismic Category I, constructed with low-porosity concrete, and to has above-ground building entrance and penetrations, thereby minimizing underground piping penetrations. This design approach minimizes the potential for the infiltration of water and the spread of contamination to the environment. Additionally, yard tank overflows are routed back to the AB sump for collection and processing to minimize unintended contamination.	9.3.4

Table 12.4-10 (19 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	(cont.)	• Drains from cleaning and maintenance activities are collected and routed for treatment in the DE system.	
2	Adequate and Early Leak Detection	• The CVCS is equipped with instrumentation to assist the detection of RCS leakage. Variations of VCT level, RDT level, EDT level, letdown flow rate, and charging flow rate indicate the potential for RCS leakage. • The VCT, RDT, EDT, and CVCS processing equipment is designed with sufficient capacity to provide temporary holding of primary coolant for normal operation, including anticipated operational occurrences. The tanks are equipped with level indicating and control instruments to prevent overflow and provide reasonable assurance of a timely processing, thus minimizing interruption of normal processing operation, the spread of contamination, and waste generation. • The sumps in the cubicles in which the CVCS tanks are located are designed with leak detection instruments to initiate alarms for operator actions in the event of leakage or overflow. The leak detection design has the capability to detect a small quantity of leakage to provide early warning to operators. • The piping connecting the outside tanks to the SSCs that are inside the AB is routed through a pipe tunnel. The floor of the piping tunnel is sloped and epoxy coated, and has a sump with a level switch to detect piping leakage and infiltrated water. In the event that liquid is detected, the level switch sends a signal to the MCR for operator action. This design approach minimizes contamination of the environment.	9.3.4

Table 12.4-10 (20 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The primary coolant purity is maintained with chemical injection for optimal reactor operation. Boric acid is added to compensate for reactivity changes, fuel burnup, and xenon variations, and to provide shutdown margin. Lithium hydroxide is added for pH control, and hydrogen is added in order to minimize the occurrence of radiolysis. The addition of chemicals helps to minimize the generation of contaminated waste. • Boric acid in the letdown flow is recovered for reuse to the maximum extent possible. In the event that the boric acid concentrate contains an abnormal quantity of radioactivity, the concentrate is sent to the liquid waste management system (LWMS) for neutralization, treatment, and release. The boric acid concentrator operates automatically to the desired boron concentration and sends the concentrate to the BAST for reuse. This design approach minimizes waste generation. • The holdup tank, BAST, and the RMWT are located outside in a tank house designed to prevent the infiltration of rainwater and the spread of contamination. The tank house is designed with a sloped floor that is coated with epoxy to facilitate draining and cleaning, and is equipped with a sump that has level switch instrumentation to detect leakage and overflows. In the event that leakage is detected, the level switch sends a signal to the MCR for operator actions. This design approach minimizes the spread of contamination and waste generation. • The process piping containing contaminated fluids is properly sized to facilitate flow with sufficient velocities to prevent the settling of solids. The piping is designed to reduce fluid traps, thus reducing the decontamination needs and waste generation. Decontamination fluid is collected and routed to the LWMS for processing and release. • Utility connections are designed with a minimum of two barriers to prevent the spread of contamination to clean systems.	9.3.4

Table 12.4-10 (21 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	• The SSCs are designed with decontamination capabilities. Design features, such as integrated component packages, utilized welding techniques used, and surface finishes are intended to minimize the need for decontamination and the resulting waste generation. • The SSCs are designed for the full service life of the plant and are fabricated as individual assemblies for easy removal to the maximum extent practicable. • The CVCS is designed with minimal embedded and buried piping. Piping between buildings is equipped with piping sleeves so that leakage is directed back to the building for collection, thus preventing unintended contamination.	9.3.4
5	Operations and Documentation	• The CVCS is designed for automated operation with manual initiation. Boron injection is controlled by the makeup subsystem to maintain the desired boron concentration. Adequate instrumentation, including level, flow, and pressure elements, as well as process sampling, is provided to monitor and control the CVCS operations to prevent undue interruption, thus minimizing the spread of contamination and waste generation. • Leak detection instruments are provided to detect individual tank leakage. Adequate space is provided around the equipment to enable prompt assessment and responses when required. • Operational procedures and maintenance programs as related to leak detection and contamination control are to be prepared by the COL applicant (COL 9.3(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant (COL 9.3(2)). Documentation requirements are included as a COL information item.	9.3.4
6	Site Radiological Environmental Monitoring	• The CVCS is part of the overall plant and is included in the site radiological environmental monitoring program for monitoring of facility and environmental contamination. The program includes sampling and analysis of boric acid and contaminated fluid leakage from the BAST, RMWT, holdup tank, and piping tunnel, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. The program is included as a COL information item (COL 9.3(3)).	9.3.4

Table 12.4-10 (22 of 75)

System: Steam Generator Blowdown System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The SGBS components are located in elevated cubicles inside the auxiliary building. The cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination through the facility and to the environment. • The SGBS is designed with sufficient capacity for different modes of operation, including CBD, ABD, HCBD, and emergency BD. The system piping is adequately sized to prevent blockage and is sloped to facilitate drainage and prevent crud buildup. • The flash tank, heat exchangers, filters, demineralizers, and the wetted parts of the WLS recirculation pumps are fabricated from stainless steel material and use welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The facility areas that house the system components, including the equipment cubicles that contain radioactively contaminated or potentially contaminated fluid, are designed with sloped floors with epoxy coating to facilitate the draining of fluid into drain pipes that direct liquid into a local sump. The SGBS component cubicles have epoxy-coated walls to facilitate cleaning. The facility layout facilitates the operators' prompt assessment and fast response when needed.	10.4.8
2	Adequate and Early Leak Detection	• The SGBS is designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level, flow rate, temperature, and pressure elements, and a process radiation monitor, is provided to monitor the system operation to prevent undue interruption.	10.4.8
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The SGBS components are provided with demineralized water for decontamination. Nitrogen and other utilities are provided to facilitate operations. The utility connections are designed with a minimum of two barriers to prevent contamination of clean systems.	10.4.8

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	(cont.)	• Process sampling connections are provided to determine the levels of contamination, treatment requirements, and confirmation of the continual radiation monitoring output. Continuous process radiation monitoring is provided on the outlet line of the treated blowdown water. The detection of high radiation levels initiates automatic valve closure for isolation and operator actions, minimizing cross-contamination.	
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal to the maximum extent possible. • The SSCs are designed to facilitate decontamination. Design features, such as the welding techniques used and surface finishes, minimize the need for decontamination and the resultant waste generation. • The SGBS is designed without any embedded or buried piping. Piping between buildings is equipped with piping sleeves or tunnel, as applicable, with leak detection features, thus preventing contamination of the environment.	10.4.8
5	Operations and Documentation	• The removal and packaging of spent filter elements and spent resin are designed for remote manual operation. Adequate space is provided around the equipment to enable prompt assessment and responses. • The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control (COL 10.4(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of the system design, construction, design modifications, field changes, and operations (COL 10.4(2)). Documentation requirements are included as a COL information item.	10.4.8
6	Site Radiological Environmental Monitoring	• The SGBS is part of the overall plant and is included in the site radiological environmental monitoring program for monitoring the potential for environmental contamination. The program includes sampling and analysis of waste samples, meteorological conditions, hydrogeological parameters, and potential migration pathways of the radioactive contaminants. The COL applicant is to prepare the Site Radiological Environmental Monitoring Program (COL 10.4(8)).	10.4.8

Table 12.4-10 (24 of 75)

System: Component Cooling Water System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system components, including the CCWS surge tanks, pumps, and associated piping, are fabricated from carbon steel material and are of welded construction for life-cycle planning. The CCWS is injected with a corrosion inhibitor, and the surge tanks are covered with nitrogen gas to minimize the oxidation of steel and formation of corrosion. The CCWS heat exchangers are made of titanium, and pump impellers are made of stainless steel. The system design includes periodic sampling and analysis to maintain water quality and provisions for in-service testing and inspection to maintain system integrity. This design approach minimizes leakage and unintended contamination of the facility and the environment. • The CCWS surge tanks, heat exchangers, and pumps are designed with multiple divisions and sufficient capacity to accommodate different modes of operation, including normal and anticipated operational occurrences. The tanks are designed to meet ASME Section III, Division I, and are equipped with dual-level instruments to facilitate control of the content liquid level, thus minimizing the spread of contamination and waste generation. • The facility area that houses the system components, including the equipment cubicles that contain radioactively contaminated or potentially contaminated fluid, are designed with sloped floors with epoxy coating to facilitate the draining of fluid into drain pipes that direct liquid into a local sump. The facility layout facilitates the operators' prompt assessment and fast response when needed. • Sumps that contain contaminated or potentially contaminated fluid are equipped with stainless steel liners and level switches to initiate pumping when sump levels reach a predetermined setpoint. The sumps are designed to facilitate periodic maintenance and inspection of the epoxy coating.	9.2.2

Table 12.4-10 (25 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	• CCWS surge tanks are designed with dual-level instruments to provide reasonable assurance of safe operation and to provide alarms to the operating personnel in the event of overflow or leakage. • Two radiation monitors, one for each division, are provided to continuously monitor contamination levels at the discharge of the CCWS pumps and indicate radiation activity in the MCR. In the event that radiation is detected above the pre-determined limit, an alarm is initiated by one of the monitors for operator action. Because the CCWS is segregated into two independent and parallel divisions, each division can be isolated for inspection, mitigation, and maintenance. This design approach provides early leak detection and minimizes the spread of contamination to other components, the facility, and the environment.	9.2.2
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• SSCs are designed with life-cycle planning through the use of nuclear industry- proven materials that are compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • CCWS surge tanks are located at a high elevation to maintain liquid pressure in order to prevent infiltration and cross-contamination of the CCWS. This location also helps to prevent unintended contamination of the environment because any leaks from the tanks are drained to the local sumps for collection and forwarded to the LWMS for treatment and disposal. • The cubicles where the CCWS SSCs are housed are designed with sloped floors, epoxy coating to provide drainage and cleanable surfaces, and local sumps to collect leakages and overflows. Cubicle curbs are also provided to reduce cross- contamination and the spread of contamination to other areas. CCWS heat exchangers are located in a separate structure that is close to the essential service water building in order to minimize radiation exposure to the essential service water piping. • Plate-type heat exchangers are used to transfer heat from the component cooling water side to the essential service water side. The heat exchangers are designed with titanium plates to minimize the potential for pinhole leaks between the component cooling water and essential service water. The gaskets between the plates are also designed to direct leakages to the outside of the heat exchangers. Leakages are then collected in the floor drain sumps and routed for treatment and release. This design approach minimizes contamination of clean systems as well as to the facility and the environment.	9.2.2

Table 12.4-10 (26 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	(cont.)	• Sampling points are provided to facilitate sampling and analyses to provide reasonable assurance that the water quality is maintained and also to determine the need for the addition of corrosion inhibitor. This design approach minimizes waste generation. • Utility connections (e.g., demineralized water, nitrogen) are designed with a minimum of two barriers to prevent the contamination of clean systems.	
4	Decommissioning Planning	• SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal to the maximum extent possible. • The CCWS is designed with minimum embedded or buried piping. Piping between buildings is equipped with piping sleeves or tunnels, as applicable, with leak detection features. Piping to the component cooling water heat exchanger structures is routed through a seismic Category I reinforced concrete pipe tunnels (one per division) under the yard. The tunnel is coated with epoxy and is designed with a collection sump with a level switch to initiate an alarm signal in the MCR for operator actions in the event liquid is detected. This design approach thus minimizes unintended contamination of the facility and the environment.	9.2.2
5	Operations and Documentation	• The CCWS is designed for automated operation with manual initiation for the different modes of operation. CCW surge tanks are designed with dual-level instruments to provide reasonable assurance of safe operation. A high-level signal, indicating in-leakage, or a low-level signal, indicating out-leakage, is transmitted to the MCR for operator action. A low-low-level signal isolates the nonessential headers and the RCP headers from the remaining portion of the system, thus minimizing the spread of contamination and waste generation. • Adequate space is provided around the equipment to enable prompt assessment and responses as needed. • The COL applicant is to provide operational procedures and maintenance programs as related to leak detection and contamination control in the CCWS (COL 9.2(10)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations in the CCWS (COL 9.2(11)). Documentation requirements are included as a COL information item.	9.2.2

Table 12.4-10 (27 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The CCWS is considered to have a potential low level of contamination through leakage from contaminated systems through heat exchangers. Through monitoring, inservice inspection, and lessons learned from industry experience, the integrity of the CCWS is maintained, resulting in a minimal level of contamination. Leakage from the system to the facility and the environment is captured by the drainage collection provisions. Any residual contamination of the hydrogeology is not likely to be distinguishable from other contamination sources. Hence, contamination characteristics of the CCWS are not monitored in the site-wide program. However, the COL applicant is to include a site-wide radiological environmental monitoring program to monitor environmental contamination in the CCWS (COL 9.2(12)).	9.2.2

Table 12.4-10 (28 of 75)

System: Essential Service Water System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The ESW facility is designed not to be in close contact with radioactively contaminated components and is located away from contamination areas to prevent unintended contamination. • All openings at the pump house floor are sealed to prevent water entry and preclude the flooding of the ESW pumps and other safety-related equipment within the structure.	9.2.1
2	Adequate and Early Leak Detection	• Radiation monitors are provided to detect contamination after the exchanging heat with CCW heat exchangers. This design approach provides early detection of contamination. • The ESWS is designed to be readily accessible for inspection and maintenance.	9.2.1
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The heat exchangers are constructed with titanium material to minimize pinhole leaks. • The heat exchanger seals are designed to leak toward the outside of the heat exchangers where leakage is collected in the building sump. • A sump is provided for collection of any leakage. The sump is designed with steel liners and is equipped with level instruments to initiate an alarm signal for operator actions.	9.2.1
4	Decommissioning Planning	• The ESWS is designed for the full service life and is fabricated as individual assemblies for easy removal. • The ESWS is designed with minimal embedded or buried piping. Piping between buildings is designed to be routed in seismic Category I, reinforced concrete pipe tunnels (one per division) under the yard.	9.2.1
5	Operations and Documentation	• The ESWS is designed for automated operations with manual initiation for the different modes of operation in conjunction with the CCWS. • Adequate ingress and egress spaces are provided for prompt assessments and appropriate responses when and where they are needed.	9.2.1

Table 12.4-10 (29 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The ESWS is designed to minimize the potential for contamination through leakage in the heat exchangers. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the ESW/CCW heat exchangers is expected to be well maintained, resulting in no contamination or to a very low level of contamination of the system. Leakage from the system to the facility and the environment is captured by the design. Any residual contamination of the hydrogeology is not likely to be distinguishable from other contamination sources. Hence, ESWS has low risk and low radiological consequence, and this design is in compliance with NRC RG 4.21.	9.2.1

Table 12.4-10 (30 of 75)

System: Spent Fuel Pool Cooling and Cleanup System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The SFPCCS components are located in elevated cubicles inside the auxiliary building. The cubicle floors are sloped and coated with epoxy and are provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination within the facility and to the environment. • The spent fuel pool, refueling pool, refueling canal, and cask loading pit are installed with stainless steel liner plates and welded seam drain channels. Other components, including heat exchangers, filters, demineralizers, and pumps, are fabricated from stainless steel material and utilize welded construction for life- cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment.	9.1.3
2	Adequate and Early Leak Detection	• The SFPCCS is designed to include sight glasses for visual inspections. Early leak detection can be achieved through the use of sight glasses and operating procedures. • Adequate instrumentation, including level, flow rate, temperature, and pressure elements, is provided to monitor the system operation to prevent undue interruption.	9.1.3
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials that are compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • Plate-type heat exchangers are used for thermal transfer from the spent fuel pool cooling side to the component cooling water side. The heat exchangers are designed with stainless steel plates to minimize the potential for pinhole leaks between the potentially contaminated system and component cooling water. The heat exchangers are designed for the component cooling water to operate at a higher pressure than the process fluid side, which prevents the spread of contamination to the clean system side through leakage. The gaskets between the plates are also designed to direct leakage to the outside of the heat exchangers. The leakage is then collected in the local floor drain sump. This design approach minimizes the spread of contamination to the facility and the environment. • The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems.	9.1.3

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal, with the exception of the liner plates. • The SSCs are designed to facilitate decontamination. Design features, such as the welding techniques that are used and surface finishes, are included to minimize the need for decontamination and the resultant waste generation. • The SFPCCS is designed with minimum embedded or buried piping. Piping between buildings is equipped with piping sleeves or tunnel, as applicable, with leak detection features, thus preventing unintended contamination to the environment.	9.1.3
5	Operations and Documentation	• The removal and packaging of spent filter elements and spent resin is designed for remote manual operation. Adequate space is provided around the equipment to enable prompt assessment and responses when required. • The COL applicant is to provide operational procedures and maintenance programs for the inspection, calibration, testing, and maintenance of the SFP leak detection provision (sight glasses), pool water temperature, pool water level, SFP area radiation monitor, and the SFP steel liner. The COL applicant is also to provide the inspection interval for the maintenance program. In addition, the COL applicant is to provide operational procedures and maintenance programs for the design features implemented for SFP contamination control, including heat exchanger seals, epoxy coating, and SFP filters and demineralizers. The contamination control procedures and programs can be integrated into an overall plant-wide NRC RG 4.21 program following the guidance from NEI 08-08A (COL 9.1(1)). Procedures and maintenance programs are to be completed before fuel is loaded. • The COL applicant is to maintain complete documentation of system design and system design modifications (COL 9.1(7)). Documentation requirements are included as a COL information item.	9.1.3

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The SFPCCS is designed to manage radioactive contamination through the storage of spent fuel. The integrity of the SFPCCS is maintained through monitoring, in- service inspection, and the implementation of lessons learned from industry experience. Maintaining the SFPCCS results in a low level of contamination in the facility. Because the SFPCCS is located at higher plant elevations, the potential for environmental contamination of soil and groundwater from pool liquid leakages is minimized. However, because the pool is open, contamination from the evaporation of water from the SFPCCS and other systems is included in the site radiological environmental monitoring program. The program is included as a COL information item.	9.1.3

Table 12.4-10 (33 of 75)

System: Gaseous Waste Management System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system contains sufficient charcoal material to hold the noble gas nuclides for a period of decay to reduce the release of radioactivity, thus minimizing contamination of the facility and the environment. • The system design, including the waste gas dryers and guard beds, is configured in two parallel trains, one operating and one standing by, each with sufficient capacity to remove the moisture to protect the charcoal beds. • A HEPA filter is provided downstream of the charcoal beds to prevent the spread of contaminated charcoal fines. • Cubicles in which contaminated materials are stored and processed are epoxy-coated to facilitate cleaning. The GRS header drain tank is equipped with level instrumentation to detect fluid accumulation and drain the fluid to the radioactive drain system. • The system is designed with above-ground piping to the extent practicable. Buried and embedded piping is minimized. Piping is sloped to facilitate drainage of condensate to the header drain tank. • The system uses valves with leak-tight characteristics, such as the bellows or metal diaphragm types, to minimize leakage. • The system uses welded construction to the maximum practicable extent to minimize leakage.	11.3.2
2	Adequate and Early Leak Detection	• The system is designed with gas analyzers and a radiation monitor to provide reasonable assurance of the integrity of the SSCs, including piping, and to provide alarms to warn operators of the potential for explosive gas concentrations. • The system is designed with adequate space around all components to enable prompt evaluation and response to leakage detection.	11.3.2
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear–industry-proven equipment and materials that are compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation. • The process piping containing contaminated fluid is sloped to facilitate flow and reduce fluid traps, thus reducing decontamination and waste generation. Decontamination fluid is collected and routed to the LWMS for processing and release.	11.3.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	(cont.)	• Utility connections are designed with a minimum of two barriers to prevent contamination of nonradioactive systems from radioactive systems.	
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities using low-pressure nitrogen. Design features such as welding techniques and surface finishes are included to minimize the need for decontamination and minimize waste generation. • The GRS is designed with minimal embedded or buried piping. The drain gas header between buildings is equipped with piping sleeves with leakage directed back to the compound building for collection, thus preventing the spread of contamination.	11.3.2
5	Operations and Documentation	• The GRS is designed for remote and automated operations. The system is equipped with instruments to actuate the drain valve from the header drain tank. • The COL applicant is to prepare the operational procedures and maintenance programs related to leak detection and contamination control (COL 11.3(4)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations (COL 11.3(5)). Documentation requirements are included as a COL information item.	11.3.2
6	Site Radiological Environmental Monitoring	• The GRS is part of the plant and is included in the site process control program and the site radiological environmental monitoring program for monitoring of facility and environmental contamination. The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program (COL 11.3(6)).	11.3.2

Table 12.4-10 (35 of 75)

System: Liquid Waste Management System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system components, including the collection tanks and the monitor tanks, are fabricated of stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The LWMS tanks are designed with sufficient capacity to provide temporary storage of the liquid waste generated from normal operation including anticipated operational occurrences. The tanks are equipped with cross-connected inlet headers for the control of overflow and to provide reasonable assurance of timely processing. The design minimizes the interruption of normal processing operation, the spread of contamination, and waste generation. • The LWMS tanks are designed with mixing eductors to minimize settling of suspended solids. The tanks have polished internal surfaces to minimize crud traps. • The LWMS is designed with minimum embedded or buried piping. Piping between buildings is designed to be equipped with piping sleeves with leakage directed back to the building for collection, thus preventing the spread of contamination.	11.2.2
2	Adequate and Early Leak Detection	• All LWMS tanks are designed with level instruments to provide reasonable assurance of safe operation of the SSCs. The instruments provide alarms to the main control room and the radwaste control room for operator action in the event of high liquid levels. • The cubicles in which the LWMS tanks are located are designed to include leak detection instrumentation to initiate alarms for operator actions in the event of leakage. The leak detection design has the capability to detect a small quantity of leakage for early detection.	11.2.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The floor drains, equipment drains, chemical drains, and the detergent wastes are collected in segregated tanks in separate cubicles. Because the LWMS is designed to operate in batches, treatment for these collected wastes is determined through sampling and analyses. This design approach minimizes cross-contamination and waste generation. • The process piping containing contaminated solids is sized to facilitate flow and to provide for velocities that are sufficient to prevent the settling of solids. The piping is designed to reduce crud traps, thus reducing decontamination and waste generation. Decontamination fluid is collected and processed. • Utility connections are designed with a minimum of two barriers to prevent contamination of nonradioactive systems from potentially radioactive systems.	11.2.2
4	Decommissioning Planning	• The SSCs are designed with decontamination capabilities. Design features such as spargers, welding techniques, and surface finishes are included to minimize the need for decontamination and minimize waste generation. • The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal.	11.2.2
5	Operations and Documentation	• The COL applicant is to maintain complete documentation of the system design, construction, design modification, field changes, and operations (COL 11.2(10)).	11.2.2
6	Site Radiological Environmental Monitoring	• The LWMS is included in the site process control program and the site radiological environmental monitoring program for monitoring facility and environmental contamination. The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program (COL 11.2(11)). The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants.	11.2.2

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System: Solid Waste Management System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system components, including the low-activity spent resin tank and the spent resin long-term storage tank, are designed with stainless steel material and welded construction for life-cycle planning. The tanks are designed to have sufficient capacity to contain the spent resins for a decay period to reduce radioactivity. • The concentrate treatment subsystem and the spent resin dewatering subsystem are designed as skid packages with self-containing drip pans to contain leakage. The drains connected to the drip pan are routed to a floor drain sump for collection and then pumped to the LWMS for treatment and release. • The temporary waste storage area is designed with epoxy-coated floors and a drainage system to direct drainage to a floor drain sump for collection and subsequent pumping to the LWMS for treatment and release. • Cubicles in which contaminated materials are handled and stored have floors that are sloped and epoxy-coated to facilitate cleaning and facilitate drainage to early leak detection piping. Sump tanks are equipped with level switches to detect liquid accumulation, and pumps are provided to transfer the fluid for proper treatment. • The SWMS is designed with above-ground piping to the extent practicable. Buried and embedded piping is minimized. In lieu of embedded piping, a drainage pipe is provided in a covered concrete trench to collect any component overflow or leakage and route it to the building sump. In the event that buried or embedded piping cannot be avoided, double-walled piping and leak detection instruments are considered and evaluated based on the risks and the radiological consequences associated with the contamination of the facility and the environment.	11.4.2
2	Adequate and Early Leak Detection	• The low-activity spent resin tank and the spent resin long-term storage tank are designed with level instrumentation to provide reasonable assurance of the integrity of the SSCs including the associated piping and to provide alarms to warn the operators of leakages. • The temporary waste storage area is designed with remote cameras for waste handling operation as well as for periodic surveillance. • Other components are designed for batch operation and are provided with adequate space to enable prompt assessment and response when required.	11.4.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation. • The process piping containing contaminated slurry is sized to facilitate flow and provide velocities that are sufficient to prevent solids settling in accordance with ANSI/ANS 55.6. The piping is designed to reduce fluid traps, thus reducing decontamination and waste generation per 10 CFR 20.1406. Pipe flushing fluid is collected and routed to the LWMS for processing and release, per ANSI/ANS 55.1. • Utility connections are designed with a minimum of two barriers to prevent contamination of nonradioactive systems from potentially radioactive systems.	11.4.2
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities. Design features such as welding techniques and surface finishes are included to minimize the need for decontamination and minimize waste generation.	11.4.2
5	Operations and Documentation	• The spent resin removal and packaging are designed for remote and automated operations. The system is equipped with instruments to provide alarms for operator actions in the event leakage is detected. • The operational procedures and maintenance programs related to leak detection and contamination control are to be prepared by the COL applicant (COL 11.4(10)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • A NRC RG 4.21 program, Life Cycle Planning of Minimization of Contamination, is to be developed by the COL applicant to identify plant-wide components that contain radioactive materials, buried piping, embedded piping, leak detection methods and capabilities, and the methods utilized for the prevention of unnecessary contamination of clean components, facility areas, and the environment following the guidance of NEI 08-08A (COL 11.4(11)). The leak identification program for the SWMS is to be integrated into this plant-wide program to facilitate the timely identification of leaks, prompt assessment, and appropriate responses to isolate and mitigate leakage.	11.4.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	(cont.)	• Complete documentation of the system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item (COL 11.4(12)).	
6	Site Radiological Environmental Monitoring	• The SWMS is part of the plant and is included in the site process control program and the site radiological environmental monitoring program for monitoring of facility and environmental contamination. The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. Both programs are included as COL information items (COL 11.4(5)).	11.4.2

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System: Radioactive Drain System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system components, including all of the EFDS sumps that handle radioactive or potentially radioactive liquid, are designed with stainless steel liners encased in concrete cavities. The concrete cavities are epoxy-coated to provide smooth surfaces for cleaning and decontamination for life-cycle planning. The EFDS sump liners are designed to be sealed at the rim to prevent infiltration of contaminated drainage. The sump liners are also designed to be removable to facilitate cleaning, inspection, and maintenance activities on a predetermined frequency based on the service life of the epoxy coating. This design approach minimizes leakage and unintended contamination of the facility and the environment. • The EFDS sumps are equipped with duplex pumps and level instruments to facilitate automated pump starting and stopping operation. Normally one pump in the sump is used. In the event of excessive collected drainage, the second pump in the sump is automatically started on the high-high-level signal. This design approach minimizes sump overflow and thus minimizes the spread of contamination and waste generation.	9.3.3
2	Adequate and Early Leak Detection	• In areas that are not normally accessible during power operation, such as the ICI cavity sump, the RCB drain sump, and the AB floor drain sumps, continual level indication is provided in the MCR and the RSR. In the event the sump liquid level reaches a preset limit, the level instrument initiates a signal to alarm in the MCR and the RSR for operator actions. This design approach, supplemented with operational procedures, provides adequate leak detection capability and minimizes the potential for the spread of contamination.	9.3.3
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The sumps that handle contaminated or potentially contaminated fluids are designed to minimize the spread of contamination through the use of liners, epoxy coating, and seals. This design approach is nuclear-industry proven and is compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The EFDS sumps are strategically located to collect floor and equipment drainage. Drains are segregated for different handling and processing requirements to minimize cross-contamination. The pumps are adequately sized for maximum input to prevent overflow; one pump is normally used, but when the sump liquid content level continues to rise, the second pump is initiated at the high-high-level. This design approach minimizes the spread of contamination.	9.3.3

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	(cont.)	• The sumps are housed in individual cubicles. Cubicle curbs and/or doors are provided to reduce the spread of contamination to other areas and additional needs for decontamination. • Sampling points are provided to facilitate sampling and analyses to properly route drainage for treatment and disposal. This design approach minimizes waste generation. • Utility connections such as demineralized water and instrument air are designed with a minimum of two barriers to prevent cross-contamination.	
4	Decommissioning Planning	• The sump liners are designed for the full service life and are fabricated as individual assemblies for easy removal. The drain pipe flange connections to the sumps are also designed to permit disconnect to facilitate removal of the liners for inspection, cleaning, and maintenance of the epoxy coating of the concrete cavity. This design approach minimizes contamination and facilitates decommissioning. • The EFDS is designed with minimum embedded or buried piping through the use of pump transfer. Drain pipes are routed through pipe chases as much as practicable. Piping between buildings is equipped with piping sleeves with leakage directed back to the originating facility for collection. This design approach minimizes the spread of contamination and facilitates decommissioning.	9.3.3
5	Operations and Documentation	• The EFDS is designed for automated operation. The sumps are designed with dual-level instruments to provide reasonable assurance of a safe operation. A high level signal initiates liquid transfer with one pump. If the liquid level continues to rise, a high-high signal initiates additional liquid transfer with the secondary pump. Level signals are transmitted to the MCR for monitoring and operator response. • Adequate space is provided in the vicinity of the sumps to enable prompt assessment and responses when required. • The COL applicant is to provide operational procedures and maintenance programs as related to leak detection and contamination control (COL 9.3 (1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of the system design, construction, design modifications, field changes, and operations (COL 9.3 (2)). Documentation requirements are included as a COL information item.	9.3.3

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The EFDS is designed to handle radioactive fluids, and the sumps are located at lower elevations to facilitate drainage collection. The EFDS is part of the overall plant and is included in the site radiological environmental monitoring program for monitoring of facility and environmental contamination. The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. The COL applicant is to prepare the Site Radiological Environmental Monitoring Program (COL 9.3 (3)).	9.3.3

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System: Primary Sampling System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The NPSS and SSS components for normal operation, including anticipated operational occurrences, are located in elevated cubicles inside the compound building. The normal primary and secondary sampling cooler racks and the associated sinks are equipped with drain lines that are connected to the local drain piping. A hood is provided to remove radioactive gases from the collected samples during analysis to minimize the spread of contamination. In addition, the cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents unintended contamination of the facility and the environment. • The post-accident sampling cooler rack and sink are located at the basemat level in the auxiliary building. The post-accident sampling rack and the sink are equipped with drain lines routed to the local drain piping. A hood is provided to remove radioactive gases from the collected samples during analysis to minimize the spread of contamination. In addition, the cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents unintended contamination of the facility and the environment. • The NPSS, PASS and SSS are designed to be segregated with individual coolers and sampling points. Sampling piping sizes and components are designed to provide desired flow and sampling conditions in order to obtain representative samples. • The normal primary sample sink, normal primary sample cooler rack, secondary sample sink, secondary sample cooler rack, post- accident primary sample sink, and post-accident primary sample cooler rack are constructed of stainless steel material. The primary off-gas sample pump is fabricated from stainless steel material and is of welded construction for life-cycle planning, thus facilitating decontamination and minimizing the spread of contamination through leakage. • The facility area that houses the system components, including the equipment cubicles that contain radioactively contaminated or potentially contaminated fluid, are designed with sloped floors with epoxy coating to facilitate the draining of fluid into drain pipes that direct liquid into a local sump. The equipment cubicles for the NPSS, PASS and SSS have epoxy coated wall to facilitate cleaning. The facility layout facilitates the operators' prompt assessment and fast response when needed.	9.3.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	• The NPSS, PASS and SSS are designed with automated operation with manual initiation for the two modes of operation. Adequate instrumentation is provided to control the sampling operations, thus minimizing waste generation.	9.3.2
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation. • The NPSS, PASS and SSS components are provided with demineralized water and nitrogen gas for decontamination and purging. The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems.	9.3.2
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The components are located in accessible areas for operation, maintenance, and decommissioning purposes. • The SSCs are designed with decontamination capabilities. Other design features, such as the welding techniques that are used and surface finishes, are implemented in order to minimize the need for decontamination and the resultant waste generation. • The NPSS, PASS and SSS are designed without any embedded or buried piping. Piping between buildings is equipped with piping sleeves or tunnel, as applicable, with leak detection features, thus preventing unintended contamination of the environment.	9.3.2
5	Operations and Documentation	• The NPSS, PASS and SSS are designed for automated and remote operations with manual initiation for the collection of samples. Adequate space is provided around the components to enable prompt assessment and response when required. • The NPSS, PASS and SSS are a packaged vendor design. The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control (COL 9.3(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations (COL 9.3(2)). Documentation requirements are included as a COL information item.	9.3.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• The NPSS, PASS and SSS are designed to be used only for the purpose of collecting and analyzing samples. The procedures for proper handling and treatment of sampled materials include small volumes in the sampling operation. The NPSS, PASS and SSS are not included in the site radiological environmental monitoring program.	9.3.2

Table 12.4-10 (46 of 75)

System: Main Steam System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The main steam lines are equipped with radiation monitors to detect radiological contamination and to provide alarms for operator notification. Following a steam generator tube rupture, the MSIVs for the affected steam generator are closed in order to isolate the affected steam generator. This design minimizes radiological cross-contamination and unintended contamination of the facility. • The main steam piping is sloped in the direction of steam flow to avoid water entrenchment and the collection of condensate drainage. • To minimize the possibility of water impingement into the main turbine, drain traps are provided at low points in the main steam piping where water may collect. Condensate from these drain traps is continuously removed with direct piping to the main condenser during normal plant power operation. This design approach prevents the spread of contamination within the facility.	10.3.2
2	Adequate and Early Leak Detection	• Radiological monitoring of the MSS is provided continuously through N-16 gamma detection with a scintillation detector and microprocessor on each main steam line from the steam generator. • Radiation monitors are provided on the condenser vacuum pump exhaust line and the steam generator blowdown line to monitor contamination levels associated with the condensate and steam generator blowdown systems. • The radiation monitors are designed to provide indication of steam generator leakage and alert the operator of a steam generator tube leak condition during power operation, including the identification of the affected steam generator.	10.3.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The safety-related portion of the main steam piping is designed to comply with the ASME Section III. The non-safety-related portion of the main steam piping is designed to ASME B31.1 for safe operation and the minimization of waste generation. • Main steam piping is designed to minimize the effects of erosion/corrosion, is adequately sized to limit velocities, and is routed with long-radius elbows to minimize potential erosion and waste generation. • The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, minimizing waste generation. • Process sampling connections are provided downstream of the MSIVs for monitoring of steam chemistry.	10.3.2
4	Decommissioning Planning	• The main steam piping is designed for the full service life and is fabricated as individual segments for easy assembly and removal. • The main steam piping is designed with cleanout capabilities. Design features such as welding techniques used and surface finishes are intended to minimize the need for decontamination, and hence reduce waste generation. • The MSS is designed without any embedded or buried piping, preventing contamination to the environment.	10.3.2
5	Operations and Documentation	• The MSS piping and components are located in the reactor containment building, auxiliary building, and turbine generator building. Adequate space is provided around the equipment to enable prompt assessment and responses. • The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control (COL 10.3(2)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.	10.3.2
6	Site Radiological Environmental Monitoring	• The MSS is part of the overall plant but does not have direct release points or contamination migration pathways to the environment during normal operation. Therefore, the MSS is not specifically required to be included in the radiological environmental monitoring program.	10.3.2

Table 12.4-10 (48 of 75)

System: Condensate System and Feedwater System (includes Feedwater Pump Turbine System, Extraction Steam System and Heater Drain System)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The components for the condensate and feedwater system and the associated subsystems are located inside the turbine generator building (TGB). The floors are sloped, coated, and provided with drains that are routed to the TGB sumps. The drains from the TGB sumps are monitored for radiological contamination. Drainage that is detected to be contaminated is routed to the LWMS for treatment and release. The balance is routed to the wastewater treatment (WWT) facility for processing and release with other non-radiological wastewater. This design approach prevents the spread of contamination through the facility and to the environment. • The condensate and feedwater systems and the associated subsystems are designed with sufficient capacities to accommodate different modes of operation. The piping is adequately sized to prevent blockage; primary process piping is sloped to facilitate drainage and prevent fluid accumulation and crud buildup. • Process sampling connections are provided under each tube bundle in condenser hotwell to detect condenser tube leaks and determine location of condenser tube leaks. • The feedwater heaters are designed to Heat Exchange Institute (HEI) and Tubular Exchanger Manufacturers Association (TEMA) Standards and heater tubes are manufactured using high-grade austenitic stainless steel for minimization of unintended leakage. The feedwater heaters are constructed of carbon steel and welded for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The heat exchangers are designed so that tube-side pressure is higher than shell-side pressure to protect against leakage from the potentially contaminated steam to the clean circulating water system. • The feedwater pump turbine system is designed to exhaust the steam into the condenser, and to direct the casing leakage from the turbine to the condenser pit sump. • To minimize the possibility of water impingement into the main turbine, drain traps are provided at low points in the extraction steam piping where water may collect. Condensate from these drain traps is continuously removed with direct piping to the main condenser during normal plant power operation. Drain pots are provided with level instrumentation that opens the low-point drain valves upon a high signal. This design approach prevents the spread of contamination within the facility.	10.4.7

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	(cont.)	• Extraction steam piping material is ferritic alloy steel (Cr-Mo) and is designed to minimize the effects of erosion / corrosion.	
2	Adequate and Early Leak Detection	• The condensate and feedwater systems and the associated subsystems are designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level, flow rate, temperature, and pressure elements, is provided to monitor and control the operations to prevent undue interruption. This design approach minimizes the spread of contamination and waste generation. • Radiation monitors are provided on the condenser vacuum pump exhaust line and the steam generator blowdown line to monitor contamination levels associated with the condensate and steam generator blowdown systems. • Leak detection trays are included at all tube-to-tubesheet interfaces. Provisions for early leak detection are provided for the tubesheet trays and in each hotwell section.	10.4.7
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, minimizing the spread of contamination and waste generation. • Normal and emergency drains are routed to the condenser and are forwarded to the condensate polishers for treatment, minimizing the spread of contamination. • If leakage occurs from the condensate and feedwater system equipment, the water is drained to local drain hubs by gravity and transferred to the TGB building drain system sump for collection. • Extraction steam piping is designed to minimize the effects of erosion/corrosion, is adequately sized to limit velocities, and is routed with long-radius elbows to minimize potential erosion and waste generation.	10.4.7

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities. Design features such as welding techniques used and surface finishes are intended to minimize the need for decontamination, and hence reduce waste generation. • The systems are designed without any embedded or buried piping. The feedwater piping between the TGB and the AB is routed at a high elevation and is provided with a piping sleeve. Any leakage is drained back to the building and is collected by the floor drain system, thus minimizing unintended contamination of the environment.	10.4.7
5	Operations and Documentation	• The components of these systems are located in open areas inside the AB and TGB. Adequate spaces are provided around the equipment to enable prompt assessment and responses. • The COL applicant is to provide operational procedures and maintenance programs as related to leak detection and contamination control (COL 10.4(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of the system design, construction, design modifications, field changes, and operations (COL 10.4(2)). Documentation requirements are included as a COL information item.	10.4.7
6	Site Radiological Environmental Monitoring	• These systems have the potential to contain low levels of contamination. However, these systems are located inside the buildings and drainage is collected and treated before release. Hence, these systems are not required to be directly and individually monitored for environmental contamination.	10.4.7

Table 12.4-10 (51 of 75)

System: Auxiliary Feedwater System and Auxiliary Feedwater Pump Turbine System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The auxiliary feedwater pump turbine system consists of two non-condensing steam turbines for the auxiliary feedwater pumps and their associated piping and valves. The auxiliary feedwater pump turbine system is not used during normal power operation, startup, or shutdown. The turbines are either manually or automatically actuated by an auxiliary feedwater actuation signal from the ESFAS or the diverse protection system, which is initiated when the SG water level is low. • The auxiliary feedwater pump turbine system piping uses stainless steel material and is fabricated to ASME Section III, Class 3 requirements.	10.4.9
2	Adequate and Early Leak Detection	• The valve stem leak-offs and drains are collected and directed to the liquid radwaste system for treatment and release.	10.4.9
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• Normally, the AFWS is not in use and is not radioactively contaminated. There is a segment of pipe, connected to the high-pressure feedwater lines after the containment penetration to the SGs, that may become contaminated by the feedwater (recycle condensate) up to the containment isolation check valves. This segment of auxiliary feedwater piping is fabricated of stainless steel material, is of welded construction, and is designed to safety Class 3 and seismic Category I requirements. In addition, any leakage developed on this segment of piping is expected to be collected in the sump. Hence, the risk for leakage and the consequence of contamination is low. This design approach minimizes leakage and unintended contamination of the facility and the environment.	10.4.9
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal.	10.4.9
5	Operations and Documentation	• Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item.	10.4.9

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	• Because of its normal status in standby mode and low possibility for contamination, the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Therefore, inclusion of the auxiliary feedwater system inclusion in the site radiological environmental monitoring program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection of radiological contamination.	10.4.9

Table 12.4-10 (53 of 75)

System: Condensate Polishing System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system design, including all the polishing cation and mixed-bed demineralizer columns, is designed as vendor-packaged modular units with carbon steel material with rubber lining and welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. • The CPS cation- and mixed-bed demineralizer columns are designed with sufficient capacity to facilitate continuous partial to full flow during normal operation, including anticipated operational occurrences. The columns are designed with periodic resin replacement to prevent buildup of contaminants, thus minimizing the potential of accumulation of significant amount of contamination and waste generation. • The facility area that houses the system components, including the equipment cubicles that contain radioactively contaminated or potentially contaminated fluid, shall have sloped floors with coating to facilitate the draining of fluid into drain pipes and/or trenches that direct liquid into a local sump. In addition, the coated walls provide smooth surfaces for cleaning and to facilitate liquid draining. To the extent practicable, the cubicles shall also have early leak detection capabilities to detect component leakage, and shall have provisions to initiate alarm signals for operator actions. The facility layout shall facilitate the operators' prompt assessment and fast response when needed. • Sumps that contain contaminated or potentially contaminated fluid shall be equipped with sump tanks and level switches to initiate pumping when sump tank levels reach a predetermined setpoint. The concrete sumps shall be coated and the sump tanks shall have seals to prevent unintended infiltration of liquid into the sumps. The sumps and sump tanks shall be designed to facilitate periodic maintenance and inspection of the coating. • Piping embedment shall be minimized to the extent practicable. Where embedment cannot be avoided, consideration shall be given to minimizing embedded piping lengths and using double-walled piping with leak detection capabilities on the outer piping.	10.4.6

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	• The CPS vessels are designed with pressure instruments to provide reasonable assurance of safe operation of the SSCs, including the associated piping, and provide alarms to the operating personnel in the event of high pressure due to accumulation of impurities and low pressure from channeling and/or leakages. • The facility in which the CPS vessels are located is designed with a trench to collect leakage to the sump for processing and disposal. This design minimizes contamination of the facility and the environment from the accumulation of contaminated water.	10.4.6
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation. • The condensate is located separately in an area that is shielded and is equipped with a leak collection trench. The trench is sloped and coated to enhance drainage and cleaning to minimize contamination of the facility and the environment. This design approach also minimizes cross-contamination and waste generation.	10.4.6
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation (cont.)	• The CPS is designed to operate in total, partial, and bypass flow modes to meet secondary chemistry requirements. The system is designed with on-line specific conductivity instruments, pH, and other process instrumentation for automated operation with manual initiation, and sampling and analyses to facilitate plant personnel to determine operating requirements. This design approach minimizes cross-contamination and waste generation. • Special internals are provided in the cation vessels to allow backwashing of the top layer of cation resin using service air and demineralized water. The process piping containing contaminated solids is properly sized to facilitate flow and maintain sufficient velocities to prevent settling of solids. The piping is designed to reduce fluid traps, thus reducing the decontamination needs and waste generation. • Utility connections (sluice water and service air) are designed with a minimum of two barriers to prevent cross-contamination.	10.4.6
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual packaged assemblies for easy removal. • The CPS is designed with no embedded or buried piping.	10.4.6

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The CPS polishing operation is designed for automated operations with manual initiation for the different modes of operation. Adequate instrumentation, including conductivity, level, flow, and pressure elements, is provided to monitor and control the operations to prevent undue interruption, thus minimizing the spread of contamination and waste generation. • Operational procedures and maintenance programs are either prepared or to be included as COL Information items. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL Applicant. Documentation requirements are included as a COL	10.4.6
6	Site Radiological Environmental Monitoring	• The CPS is designed with low level of contamination through leakage in the steam generator. Industry experiences demonstrate that the integrity of the CPS is well maintained, resulting minimal contamination of the facility and the environment. Hence Radiological Environmental Monitoring Program is not required for this system.	10.4.6

Table 12.4-10 (56 of 75)

System: Condenser Vacuum System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The condenser vacuum system components are located in an open area at the foundation level inside the turbine generator building. The components are designed to be skid mounted or provided with drip pan and curb. The floors are sloped, coated and provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination to the facility and the environment. • The condenser vacuum system is designed with sufficient capacity and redundancy to support normal operation, including anticipated operational occurrences. The components and piping are fabricated from carbon steel with welded construction for life-cycle planning, as well as minimizing leakage, and unintended contamination of the facility and the environment.	10.4.2
2	Adequate and Early Leak Detection	• The condenser vacuum system is designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level and pressure elements and a radiation monitor, is provided to monitor the operations. Upon receipt of a high radiation signal, the vacuum discharge is diverted to the reactor containment building drain sump, and the vent from that sump is processed through the containment ventilation purge system. This design approach minimizes the spread of contamination and provides early detection.	10.4.2
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The condenser vacuum system components are provided with seal water (demineralized water) for decontamination. Chilled water and other utilities are provided to facilitate operations. The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems.	10.4.2

Table 12.4-10 (57 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. • The SSCs are designed to facilitate decontamination. Design features, such as the welding techniques and surface finishes, are included to minimize the need for decontamination and therefore minimize waste generation. • The condenser vacuum system is designed without any embedded or buried piping, thus preventing unintended contamination to the environment and facilitating eventual decommissioning.	10.4.2
5	Operations and Documentation	• The condenser vacuum system is located in an open area inside the turbine generator building. Adequate space is provided around the equipment to enable prompt assessment and responses when required. • The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control (COL 10.4 (1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations (COL 10.4 (2)). Documentation requirements are included as a COL information item.	10.4.2
6	Site Radiological Environmental Monitoring	• The condenser vacuum system is part of the overall plant and is included in the site radiological environmental monitoring program, for monitoring of the release of non-condensable gases and the potential for environmental contamination. The program includes air sampling and analysis and monitoring of meteorological conditions, hydro-geological parameters, and potential migration pathways of the radioactive contaminants. The site environmental monitoring program is included as a COL information item.	10.4.2

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System: Circulating Water System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The CWS, including all the CWS tubes and associated piping, is designed with corrosion- and erosion-resistant materials for life-cycle planning. The tubes are designed with continuous cleaning features to maintain heat transfer performance and tube integrity, thus minimizing fouling that may result from corrosion. • The CWS is designed in conjunction with the turbine generator building drain system. The sumps in the turbine building generator building drain system are used to collect above-ground piping leakages and are equipped with dual level instruments to ensure control of liquid level, thus minimizing the spread of contamination and waste generation. • CW is sampled and analyzed periodically to monitor radiological contamination in the water. Contamination detected is to be tracked to the source(s) of leakage and corrective actions (repair and/or isolate through tube plugging) are to be initiated. This design approach minimizes leakage and unintended contamination of the facility and the environment. • The facility that houses the system components, including the equipment cubicles that contain radioactively contaminated or potentially contaminated fluid, shall have sloped floors with coating to facilitate the draining of fluid into drain pipes and/or trenches that direct liquid into a local sump. • Sumps that contain contaminated, or potentially contaminated fluid shall be equipped with level switches to initiate pumping when sump levels reach a predetermined setpoint. The concrete sumps shall be coated and shall have seals to prevent unintended infiltration of liquid into the concrete sumps. The sumps shall be designed to facilitate periodic maintenance and inspection of the coating.	10.4.5
2	Adequate and Early Leak Detection	• The CWS leakage is detected by sampling and analysis of the circulating water in the piping, and fluid levels in the DTS sumps, followed by analysis and inspection. This approach is consistent with industry practices and is effective.	10.4.5
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation. • The CWS is designed to operate at a higher pressure than the condenser. This designed approach minimizes infiltration of condensate into the circulating water and is the most effective.	10.4.5

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	(cont.)	• The areas, in which the CWS SSCs are housed, are designed with sloping floors, epoxy coating to provide drainage and cleanable surfaces, and local sumps to collect leakages and overflows. Cubicle curbs are also provided to reduce cross-contamination and spread of contamination to other areas.	
4	Decommissioning Planning	• The CWS is designed with some buried piping in the yard because of the piping size. CW is sampled and analyzed periodically to monitor radiological contamination in the water. Contamination detected is to be tracked to the source of leakage and corrective actions (repair and/or isolate through tube plugging) are to be initiated. This design approach minimizes unintended contamination of the soil surrounding the buried pipe.	10.4.5
5	Operations and Documentation	• The CWS is designed for automated operations with manual initiation for the different modes of operation. • Adequate ingress and egress spaces are provided for prompt assessments and appropriate responses to tube leaks when and where they are needed. • The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control (COL 10.4(1)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of the system design, construction, design modifications, field changes, and operations (COL 10.4(2)). Documentation requirements are included as a COL information item.	10.4.5
6	Site Radiological Environmental Monitoring	• The CWS is designed to be a non-radiological system but with the potential of a low level of contamination through leakage in the condenser tubes. Through monitoring, inspection, and lessons learned from industry experiences, the integrity of the CWS is well maintained, with no noticeable contamination, or to a very low level of contamination of the facility and the environment that is below the detectable limit. Hence the radiological environmental monitoring program is not required for this system.	10.4.5

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System: Turbine Generator Building Open Cooling Water System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The TGBOCWS is designed to exchange heat with TGBCCW heat exchangers, and is not to be in close contact with radioactively contaminated components. The TGBOCWS system supplies cooling water from CW system and returns the water back to the CW discharge header. • The heat exchangers that the WH System services, are not expected to contain radioactive fluid. • The heat exchanger seals are designed to leak toward the outside of the heat exchangers and the leaked drains are collected in the TGB floor drain system for treatment and release.	9.2.9
2	Adequate and Early Leak Detection	• Radiation monitoring is not provided for the TGBOCWS system due to the low risk of radioactive contamination. Grab sampling is taken periodically for analysis to confirm that the cooling water return is not radioactively contaminated.	9.2.9
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The heat exchangers are constructed with titanium material to minimize pinhole leaks. • The heat exchanger seals are designed to leak toward the outside of the heat exchangers where leakage is collected in the building sump. • A sump is provided for collection of any leakage. The sump is designed with steel liners and is equipped with level instruments to initiate an alarm signal for operator actions.	9.2.9
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. • The TGBOCWS is designed with minimal embedded or buried piping.	9.2.9

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The TGBOCWS is designed for automated operations with manual initiation for the different modes of operation in conjunction with the WT System. • Adequate ingress and egress spaces are provided for prompt assessments and appropriate responses when and where they are needed. • The COL applicant is either to prepare or to include operational procedures and maintenance programs (COL 9.2(34)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item.	9.2.9
6	Site Radiological Environmental Monitoring	• The TGBOCWS is designed to prevent contamination through leakage in the heat exchangers. The integrity of the TGBCCW heat exchangers is expected to be well maintained, resulting in no contamination or to a very low level of contamination of the system. Leakage from the system to the facility and the environment is captured by the design. Any residual contamination of the hydrogeology is not likely to be distinguishable from other contamination sources. Hence, the TGBOCWS has low risk and low radiological consequence, and radiological environmental monitoring for the TGBOCWS is not considered effective. The COL applicant is to include a site-wide radiological environmental monitoring program to monitor both the horizontal and vertical variability of the onsite hydrogeology and the potential effects of the construction and operation of the plant (COL 9.2(34)).	9.2.9

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System: Turbine Generator Building Closed Cooling Water System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The TGBCCW system is designed to exchange heat with TGBCCW heat exchangers, and is not to be in close contact with radioactively contaminated components. The TGBCCW system then supplies cooling water from surge tank to cool TGB equipment (generator hydrogen cooler, ISO phase bus duct cooler, generator stator water cooler, feedwater pump turbine lube oil coolers, etc.) and returns the water back to the surge tank. • The equipment heat exchangers that the TGBCCW system services, are not expected to contain radioactive fluid, except the process sampling coolers, which may have low levels of contamination. Due to the infrequent sampling activity, and because the sampling coolers are small, the risk and consequence for WT contamination is considered low. • The TGBCCW heat exchanger seals are designed to leak towards the outside of the heat exchangers and the leaked drains are collected in the TGB floor drain system for treatment and release. • Demineralized water is used as makeup to the surge tank. This design approach, working in conjunction with the TGBOCWS, prevents unintended contamination of the WT System.	9.2.8
2	Adequate and Early Leak Detection	• The surge tank is monitored by a pressure and level instruments. A high/low level alarm for the TGBCCW surge tank is provided in the MCR and RSR for operator actions. • Radiation monitoring is not provided for the TGBCCW system due to the low risk of radioactive contamination. Grab sampling is taken periodically for analysis to confirm that the cooling water return is not radioactively contaminated.	9.2.8
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• A sump is provided for collection of any leakage. The sump is designed with steel liners and is equipped with level instruments to initiate an alarm signal for operator actions. • Boron nitrite is added to control the pH of the system for corrosion inhibition. • Nitrogen gas is used to blanket the surge tank to minimize corrosion, thus minimizing waste generation.	9.2.8
4	Decommissioning Planning	• The surge tank is a packaged unit for the full service life and is fabricated as an individual assembly for easy removal. • The TGBCCW system is designed with minimal embedded or buried piping.	9.2.8

Table 12.4-10 (63 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The TGBCCW system is designed for automated operations with manual initiation for the different modes of operation. • Adequate ingress and egress spaces are provided for prompt assessments and appropriate responses when and where they are needed. • The COL applicant is either to prepare or to include operational procedures and maintenance programs (COL 9.2(34)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations (COL 9.2(35)). Documentation requirements are included as a COL information item.	9.2.8
6	Site Radiological Environmental Monitoring	• The TGBCCW system is designed to prevent contamination through leakage in the heat exchangers. The integrity of the equipment heat exchanger and the TGBCCW heat exchangers is expected to be well maintained through surveillance, resulting in no contamination or to a very low level of contamination of the system. Leakage from the system to the facility and the environment is captured by the design. The TGBCCW surge tank is located at a high elevation. Leakage from the tank is likely be collected in the drain system inside the TGB. Hence, the TGBCCW system has low risk and low radiological consequence, and radiological environmental monitoring for the TGBCCW is not considered effective. The COL applicant is to include a site-wide radiological environmental monitoring program to monitor both the horizontal and vertical variability of the onsite hydrogeology and the potential effects of the construction and operation of the plant (COL 9.2(36)).	9.2.8

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System: Auxiliary Steam System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The auxiliary steam system condensate receiver tank, vent condenser, and pumps are located in an enclosed area at the foundation level inside the auxiliary building. The boiler is located in its auxiliary boiling building outside the auxiliary building. The components are designed to be skid-mounted and vendor-packaged units. The floors are sloped, coated and drains are directed to the local drain hubs and then sump. This design approach prevents unintended contamination of the facility and the environment. • The auxiliary steam system is designed with sufficient capacities and redundancy to support plant operation when main steam is not available. The components are designed in accordance with ASME Section VIII and other applicable codes for life-cycle planning, thus minimizing unintended contamination and waste generation. • The facility area that houses the system components, including the equipment cubicles that contain radioactively contaminated or potentially contaminated fluid, shall have sloped floors with coating to facilitate the draining of fluid into drain pipes and/or trenches that direct liquid into a local sump. Tank cubicles shall have coated walls to a height that can provide temporary barriers of the contained fluids. In addition, the coated walls provide smooth surfaces for cleaning and to facilitate liquid draining. To the extent practicable, the cubicles shall also have early leak detection capabilities to detect component leakage, overflow, and/or tank rupture, and shall have provisions to initiate alarm signals for operator actions. The facility layout shall facilitate the operators' prompt assessment and fast response when needed. • Sumps that contain contaminated or potentially contaminated fluid shall be equipped with level switches to initiate pumping when sump levels reach a predetermined setpoint. The concrete sumps shall be coated and shall have seals to prevent unintended infiltration of liquid into the concrete sumps. The sumps shall be designed to facilitate periodic maintenance and inspection of the coating. • Piping embedment shall be minimized to the extent practicable. Where embedment cannot be avoided, consideration shall be given to minimizing embedded piping lengths and using double-walled piping with leak detection capabilities on the outer piping. • Buried piping in the yard and between buildings and facilities shall be minimized to the extent practicable. Piping tunnels with leak detection capability and accessibility shall be used for piping that contains radioactive or potentially radioactive fluids.	10.4.10

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	• The auxiliary steam system is designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level and pressure elements, and a radiation monitor, is provided to monitor and control the operations. Upon a high radiation signal, the condensate is diverted to the LWMS for treatment and release. This design approach thus provides early detection and minimizes spread of contamination and waste generation.	10.4.10
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation. • The auxiliary steam system is equipped with a radiation monitor to continuously check the contamination level in the condensate, and sampling and analysis for confirmation and calibration of the radiation monitor, if necessary. If contamination is detected at or above a setpoint, an alarm is initiated for operator actions, and a signal is sent to open the condensate transfer valves to the LWMS for treatment and release. This design approach minimizes cross-contamination to other components.	10.4.10
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities. Demineralized water is provided for makeup as well as for decontamination. Design features such as welding techniques used and surface finishes are intended to minimize the need for decontamination, and hence reduce waste generation. • The auxiliary steam system is designed with minimum embedded or buried piping, thus minimizing unintended contamination due to leaking of buried or embedded piping. Yard piping is routed in an underground concrete tunnel that is designed with a leakage collection sump, level switch, and pump. An alarm is also provided in the MCR for operator actions in the event of detection of liquid.	10.4.10

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The auxiliary steam system is designed with adequate instrumentation for automatic operation with manual initiation. The boiler is a self-contained vendor package complete with its own instrumentation. The operation of the boiler operation is controlled from a local panel but with remote shutdown at low fuel oil level, or at operator initiation from the MCR. • The auxiliary steam system condensate receiver tank, the vent condenser, and the pumps are located in an enclosed cubicle inside the auxiliary building, and the boiler is located in an independent auxiliary boiler building for separation purposes. Adequate ingress and egress spaces are provided for prompt assessments and appropriate responses when and where they are needed. • The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control in accordance with NRC RG 4.21 (COL 10.4(1)). • The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations (COL 10.4(2)).	10.4.10
6	Site Radiological Environmental Monitoring	• The auxiliary steam system is designed with a low level of contamination through leakage in the condensate receiver tank, the vent condenser, pumps, boiler, and the associated piping and instruments. Industry experiences demonstrate that the integrity of the auxiliary steam system is well maintained, and the control methods included in the current design further minimize and/or prevent contamination of the facility and the environment. Hence, a radiological environmental monitoring program is not required for this system.	10.4.10

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System: HVAC Systems for Potentially Contaminated Building Areas (Fueling Handling Area HVAC System, Auxiliary Building Controlled Area HVAC System, Reactor Containment Building HVAC System, Reactor Containment Building Purge System, Compound Building HVAC System)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The system components, including ACUs, the reactor cavity AHU, low volume purge supply fans, cubicle coolers, and reactor containment fan coolers (RCFCs) in radiologically controlled areas, are fabricated from materials such as carbon and stainless steels that are compatible with the chemical, physical, and radiological environment and are of welded construction for life-cycle planning. The design thus minimizes unintended leakage and unintended contamination of the facility and the environment. • The low volume purge subsystem of the reactor containment building purge system normally operates in an exhaust mode but can be placed in a recirculation mode. Upon detection of high level of radioactivity at the common discharge duct, the subsystem is placed in the recirculation mode. The discharged air filtered from the low volume purge exhaust ACU flows back into the containment building through the low volume purge supply fan until the level of radioactivity returns back to below the preset value. • The chilled water side of the reactor cavity AHU, cubicle coolers, and RCFCs is operated at a higher pressure than the atmospheric pressure in the equipment area to prevent the contamination of the chilled water. Corrosion inhibitor chemicals are also added to the chilled water system to minimize the potential of leakage due to corrosion. • The mounting frames of the HEPA filter and carbon adsorber are stainless steel and of welded construction for life-cycle planning and to prevent unfiltered in-leakage. • Doors and door frames of all ACUs are the marine bulkhead type or an equivalent airtight construction to minimize leakage and unintended contamination of the facility and the environment. • All ACU housing penetrations are sealed by welds or adjustable compression-gland type seals to prevent unfiltered in-leakage. • All drain lines from ACUs, the reactor cavity AHU, cubicle coolers, and RCFCs are sloped and routed to the protective equipment drain. The mounting frames of the high efficient particulate air filters and carbon adsorber are stainless steel and of welded construction for life-cycle planning and to prevent unfiltered leakage.	9.4.2/9.4.5 9.4.6/9.4.7

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	• The ACU housing leak test is performed in the manufacturer's shop prior to shipment and in the field following the recommended test frequency described in ASME AG-1 to prevent unfiltered in-leakage. • Mounting frame leak tests of HEPA filters and carbon adsorbers are performed in the manufacturer's shop prior to shipment and are performed in accordance with ASME AG-1 recommended test frequencies at the site to prevent unfiltered in-leakage. • Radiation monitors are provided at the outlet of ACUs to monitor the radiological contamination levels to provide reasonable assurance that the effluent release limit will not be exceeded.	9.4.2/9.4.5 9.4.6/9.4.7
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The ACU is designed to operate slightly below atmospheric pressure through the use of an induced fan. This design approach minimizes air leakage from the ACU to its surroundings. All ducts are of welded construction to minimize cross-contamination. • HVAC component housing surfaces are painted smooth for corrosion protection, ease of decontamination, and cleaning.	9.4.2/9.4.5 9.4.6/9.4.7
		• HEPA filters capture radioactive particles and carbon adsorbers remove organic and inorganic forms of iodine from the air stream, thus minimizing the spread of contamination and the resultant waste generation. • Adequate space is provided around the HVAC equipment to enable prompt assessment and responses when required.	
4	Decommissioning Planning	• The ACUs, reactor cavity AHU, cubicle coolers, low volume purge supply fans, and RCFCs are designed and fabricated as modular units and compartments for easy decommissioning.	9.4.2/9.4.5 9.4.6/9.4.7

Table 12.4-10 (69 of 75)

	Objective	SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• Adequate work space between mounting frames for the ACUs is provided to facilitate maintenance and minimize the potential for the spread of contamination. • Carbon removal from the carbon adsorber is facilitated by the use of portable pneumatic carbon removal equipment to minimize spillage. • ACUs, the reactor cavity AHU, cubicle coolers, low volume purge supply fans, and RCFCs are designed with adequate instrumentation to be remotely operated with manual initiation and stopped from the MCR and the RSR. • The COL applicant is to establish operational procedures and maintenance programs as related to leak detection and contamination control (COL 9.4(4)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.	9.4.2/9.4.5 9.4.6/9.4.7
6	Site Radiological Environmental Monitoring	• The air quality around the plant is sampled and analyzed routinely for contamination levels and migration pathways as part of the Site Radiological Environmental Monitoring Program. Potentially contaminated HVAC systems are expected to be included in this program.	9.4.2/9.4.5 9.4.6/9.4.7

Table 12.4-10 (70 of 75)

System: Miscellaneous Building Drain System and Turbine Generator Building Drain System

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• The DT/DM Systems contain drains that originate from the TGB. These systems consist of sumps and the associated discharge piping for pumping nonradioactive floor and equipment drainage and condensate overflow to the WWT. The sumps are equipped with sump liners with pumps and level instruments to forward drains for treatment and release. • The discharge from these pumps is monitored for radiation contamination level. When contamination level is detected at or exceeding a predetermined setpoint, the drains are routed to the LWMS for processing and release. This approach prevent unintended contamination of the WWT.	9.3.3
2	Adequate and Early Leak Detection	• Radiation monitors are provided each of the discharge piping from the TGB north and south pit sumps and the condensate polishing area sump to detection contamination levels of the drains.	9.3.3
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The sump is designed with stainless steel sump liners that fit into the concrete sump pits. The concrete sump pits is coated to minimize cross-contamination of the TGB basemat and the soil and groundwater below. • The TGB drains are monitored for contamination. Only drains that are radioactively contaminated are routed to LWMS for processing. This design approach minimizes the wastewater to the LWMS, as the wastewater is likely to contain oily waste that can be harmful to the LWMS treatment system. The oily waste is either separated in the DM/DT sumps or in the LWMS normal sump to minimize waste generation. • The piping for the DM/DT Systems are constructed with stainless stain material and is routed via pipe chase to minimize embedded piping.	9.3.3
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal.	9.3.3

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The DM/DT Systems are designed for automated operations with level instruments. The radiation monitors are interlocked with the discharge valves to provide the proper routing of the contaminated fluid. • Adequate ingress and egress spaces are provided for prompt assessments and appropriate responses when and where are needed. • Operational procedures and maintenance programs are either prepared or to be included as COL Information items. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL Applicant. Documentation requirements are included as a COL Information item.	9.3.3
6	Site Radiological Environmental Monitoring	• Drains and leakage collected in the TGB sumps are generally low in potential that the drain is radioactively contaminated as condensate and feedwater are continuously polished by ion exchange to keep radiation contamination low. The DM/DT Systems are designed to provide continuous monitoring of drain routing and to prevent spread of contamination. As the contamination level is low, environmental radiation monitoring for the DM/DT Systems is not effective. A site-wide Radiological Environmental Monitoring Program is to be included to monitor the onsite hydrogeology and the potential effects of operation of the plant by the COL Applicant.	9.3.3

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System: Radiation Monitoring Systems

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• PERMS use both off-line and in-line type detectors. Process and effluent radiation monitors that come in contact with process fluids are fabricated with stainless steel to prevent unintended contamination. Readout and alarm units are installed in low-radiation areas. • The piping is stainless steel material and of welded construction, thus minimizing leakage and unintended contamination of the facility and the environment. • Area radiation monitors are installed in normally accessible locations. In the event that a detector is located in a high-radiation area, the local readout and alarm unit are located in low-radiation areas for remote readout from the detector.	11.5.2
2	Adequate and Early Leak Detection	• The RMS is designed for automated operation to provide early alarms for process and area monitoring. Radiation levels and alarms are displayed in the MCR, with the exception of the compound building radiation levels and alarms which are displayed in the radwaste control room.	11.5.2
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The monitors are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation. • PERMS monitors are provided with means for the purging or flushing the detector assemblies using nitrogen gas or demineralized water. • PERMS returns process fluids back to the system to minimize waste generation.	11.5.2
4	Decommissioning Planning	• The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. • The SSCs are designed with decontamination capabilities. Design features such as welding techniques used and surface finishes are intended to minimize the need for decontamination, and hence reduce waste generation. • The RMS system is designed with no embedded or buried piping, thus preventing unintended contamination due to the leaking of buried or embedded piping.	11.5.2

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Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The RMS system maintains a continuous record of radiation levels for documentation purposes. Plant personnel can access all RMS information from the MCR operator console. • The RMS piping and components are located in the reactor containment building, compound building, and turbine generator building. Adequate ingress and egress spaces are provided for prompt assessment and appropriate responses, when required. • Operational procedures and maintenance programs are to be prepared by the COL applicant (COL 11.5(8)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item.	11.5.2
6	Site Radiological Environmental Monitoring	• The RMS is designed to provide data for plant operation. The RMS components do not process or generate wastes that can impact contamination of facility and the environment. The RMS is designed to provide data for plant operation and the data can be used to support the radiological environmental monitoring program The program is included as a COL information item (COL 11.5(9)).	11.5.2

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System: Containment Monitoring System

	Objective	SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	• All temperature and containment water level instruments are located inside the containment with measurement readout in the MCR for continuous indication of the RCB environment and containment integrity. The monitoring provides reasonable assurance during normal power operation and minimize the potential for interruption or mishap, thus preventing unintended contamination of the facility and the environment. • Four channel of pressure instruments are located close to the containment for more accurate measurements. Readouts are provided in the MCR. This design approach minimizes unintended contamination of the facility and the environment. • Two channels of continuous monitoring of containment atmosphere hydrogen concentration are provided. After the analysis, the air samples are sent back inside the containment. • The same two channels are also use for continuous monitoring of IRWST hydrogen concentration. After the analysis, the air samples are sent back inside the containment.	6.2.5
2	Adequate and Early Leak Detection	• The CM system is designed to provide continuous indication of containment environment including pressure, containment water level, hydrogen concentration in the containment and the IRWST, and containment temperature. The instruments monitors the containment integrity for safe operation. The instruments do not process or generate any waste that require leak detection.	6.2.5
3	Reduction of Cross-Contamination, Decontamination, and Waste Generation	• The air samples from the containment and the IRWST using for hydrogen monitoring are returned back to the containment. The air is then process by the low volume purge system. This design approach minimizes the potential for cross-contamination and waste generation. • The temperature and water level instruments are located inside the containment for direct measurement. This design approach also minimizes cross-contamination.	6.2.5
4	Decommissioning Planning	• The instruments are designed for the full service life, including preventive and periodic replacement of components required for accurate indication. • The CM system piping is designed with minimal embedded or buried piping.	6.2.5

Table 12.4-10 (75 of 75)

Objective		SSC Control Measures / Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	• The CM system is designed to provide containment environmental data for monitoring of the integrity of the containment. Operational procedures and maintenance programs for these instruments are either prepared or to be included as COL information items. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item.	6.2.5
6	Site Radiological Environmental Monitoring	• The CM system is designed to provide continuous monitoring of containment integrity. The instruments do not process or generate radioactive wastes that contribute to contamination of the facility or the environment. Hence the CM system is in compliance with NRC RG 4.21. A site-wide radiological environmental monitoring program is to be included to monitor the onsite hydrogeology and the potential effects of operation of the plant by the COL applicant.	6.2.5

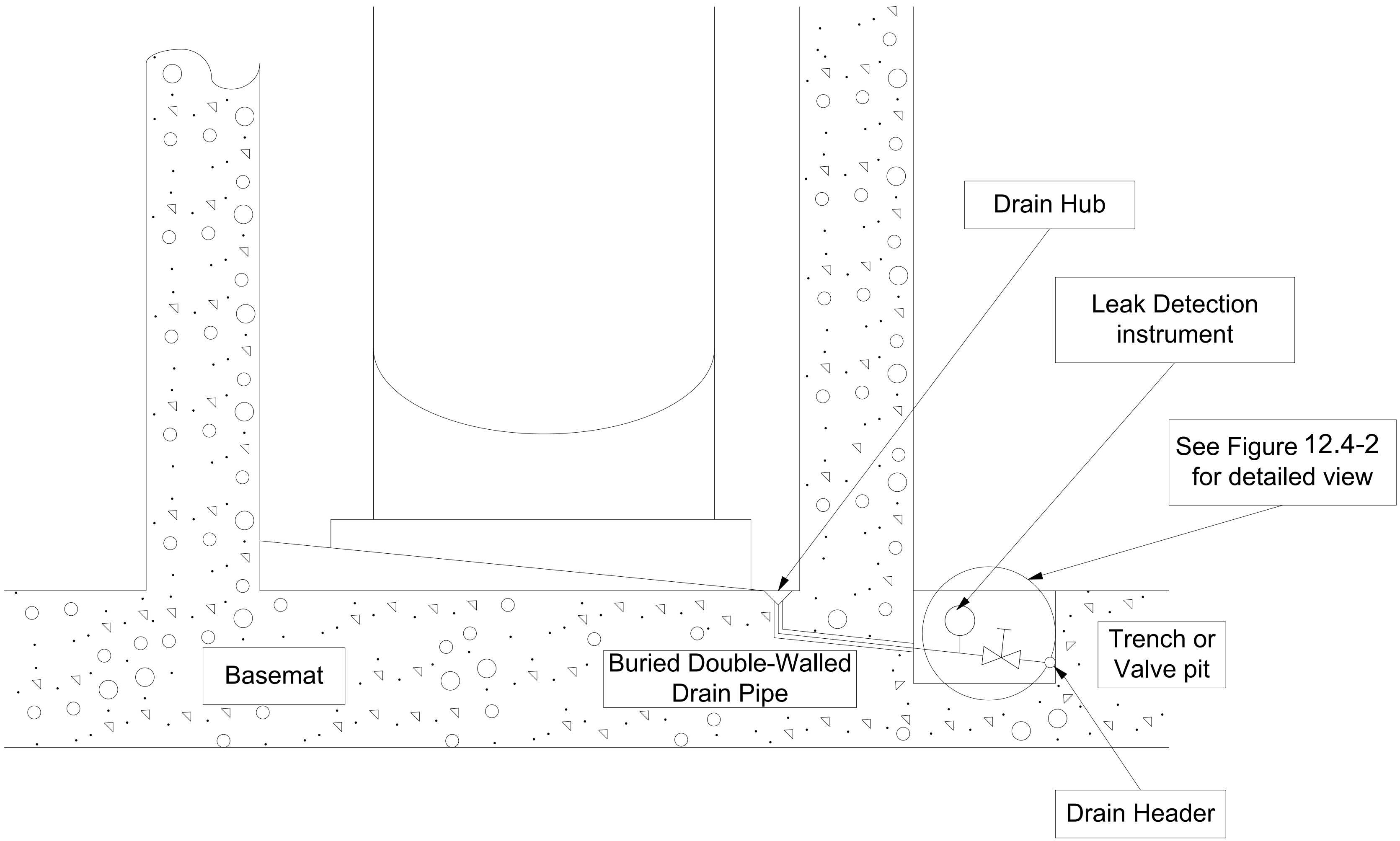

Figure 12.4-1 Typical Design Details of Drainage and Leak Detection Configuration

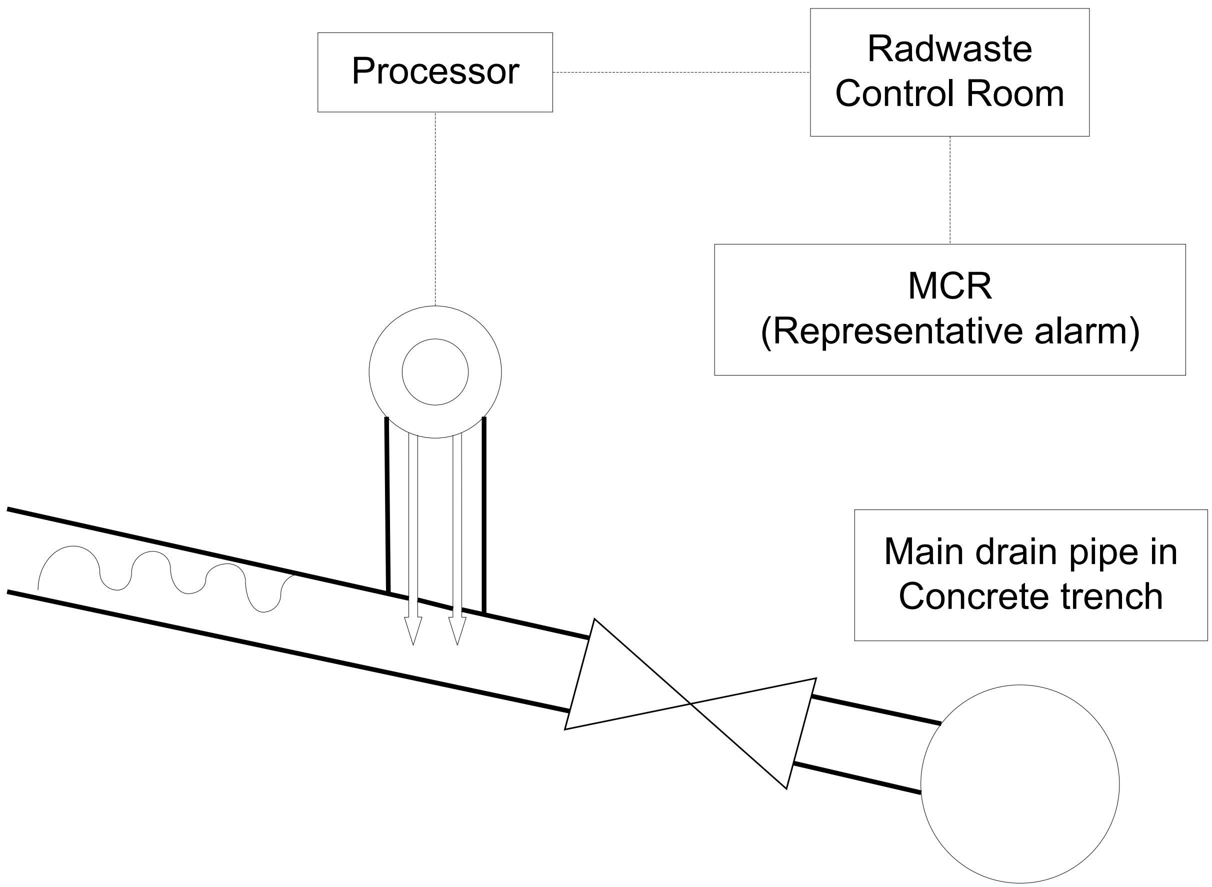

Figure 12.4-2 Typical Design Details of Leak Detection Instrumentation

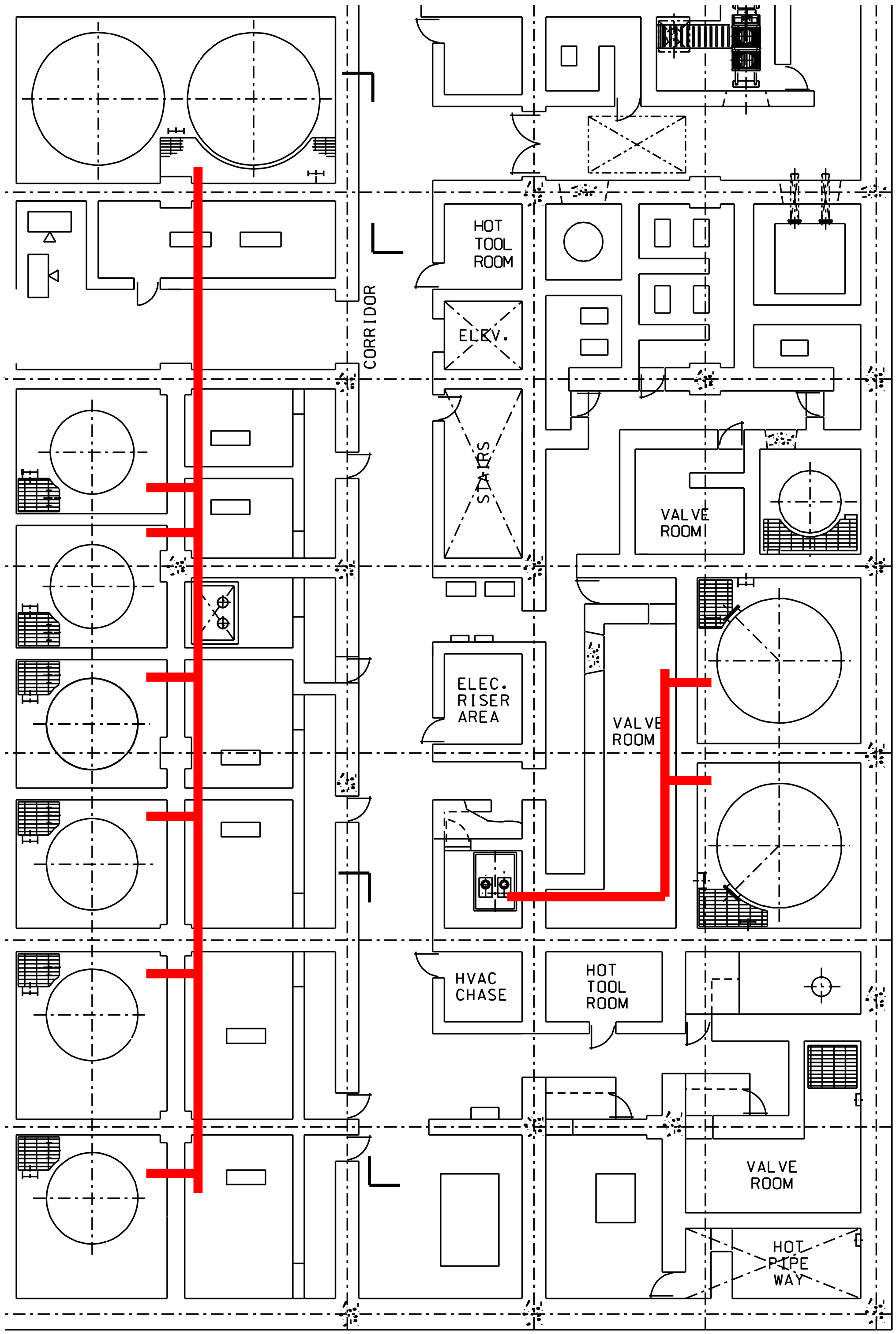

Figure 12.4-3 Typical Compound Building Trench Location Schematic

- Install piping sleeve with flexible joint between building walls
- Slope sleeve slightly to facilitate drainage
- Provide caulking on sleeve to seal and flexibility to sleeve expansion
- Provide sealing material between sleeve and inside pipe to maintain pressure boundary and piping movement
- Provide drainage

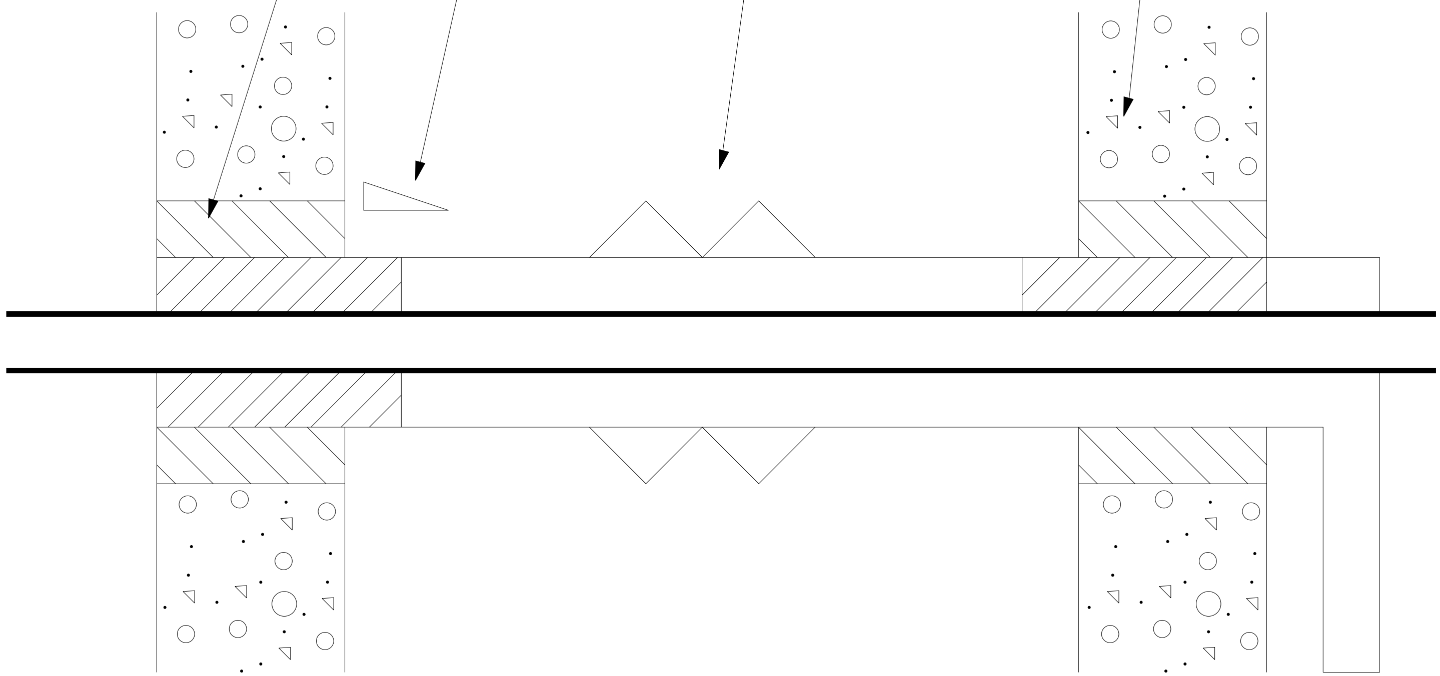

Figure 12.4-4 Typical Design for Building Sleeves Schematic

12.5 Operational Radiation Protection Program

To achieve the goal of maintaining occupational and public doses both below regulatory limits and as low as is reasonably achievable (ALARA), the operational radiation protection program shall be developed and implemented by the COL applicant in such a way that the program is clearly and sufficiently described in accordance with all applicable guidance (COL 12.1(3)).

The COL applicant is to provide the operational radiation protection program, which includes the following: (Reference 1) (COL 12.5(1))

a Organizational positions, functional responsibilities, experience, and qualifications of personnel responsible for the radiation protection program

b Equipment necessary to measure radioactivity as well as radiation fields and exposures including:

  1) The number, type, range, sensitivity, calibration method and frequency, availability

  2) The planned use of portable, fixed, laboratory, and personnel monitoring instrumentation

c. Health physics facilities and associated protective equipment for controlling ORE and contamination

d. Description of the methods for ensuring development of the training, retraining, and indoctrination program and the radiation protection instruction manuals

e. Procedures:

  1) To receive, store, transfer, and dispose of radioactive material

  2) To control exposures

  3) To control and minimize contamination

  4) To facilitate decommissioning

  5) To provide adequate radiation monitoring

6) To conduct program reviews and quality assurance

f. Operational programs including:

1) Radiation protection program

2) ALARA program

3) Groundwater protection program

4) Leakage control program

12.5.1 Combined License Information

COL 12.5(1) The COL applicant is to provide the operational radiation protection program, including the items described in Section 12.5.

12.5.2 References

1. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, June 2007.