Design Control Document for the US-APWR

Chapter 12
Radiation Protection

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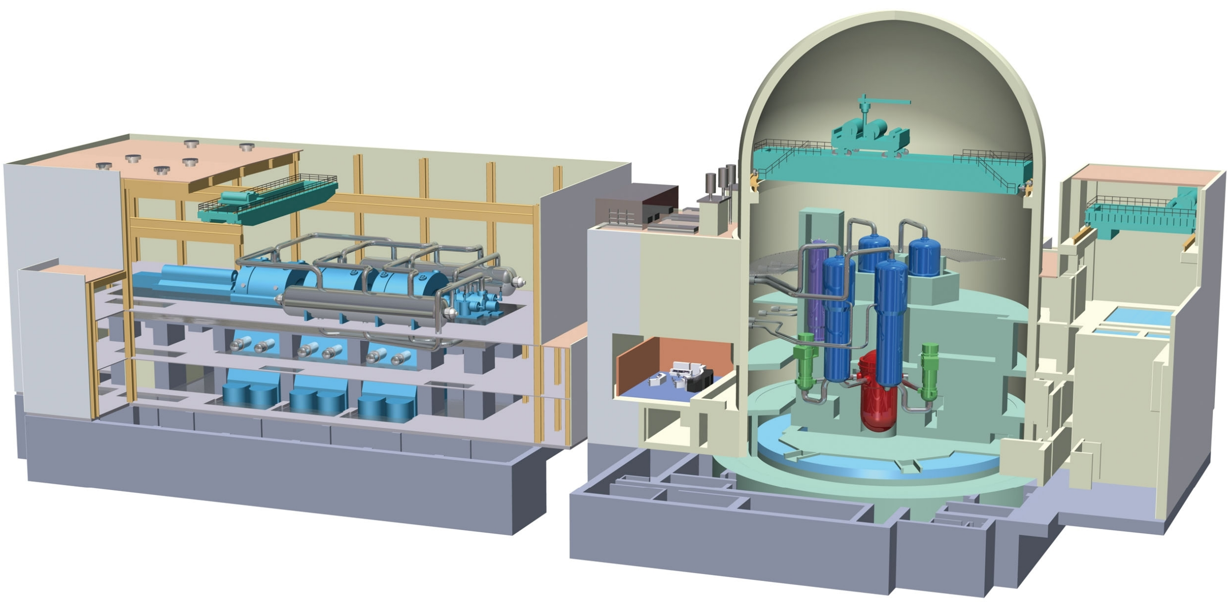

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TABLES

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ACRONYMS AND ABBREVIATIONS

A/B	auxiliary building
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARMS	area radiation monitoring system
B.A.	boric acid
CFR	Code of Federal Regulations
COL	Combined License
C/V	containment vessel
CVCS	chemical and volume control system
DAC	derived air concentration
DBA	design-basis accident
GDC	General Design Criteria
GWMS	gaseous waste management system
HEPA	high-efficiency particulate air
IEEE	Institute of Electrical and Electronics Engineers
ICIS	incore instrumentation system
ISI	inservice inspection
LOCA	loss-of-coolant accident
LWMS	liquid waste management system
MCR	main control room
mR/h	milliRoentgen per hour
mrem/h	millirem per hour
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PAM	post accident monitoring
PWR	pressurized-water reactor
R/B	reactor building
RCA	radiological controlled area
RCP	reactor coolant pump
RCS	reactor coolant system
rem/h	roentgen equivalent man per hour
RG	Regulatory Guide
RHRS	residual heat removal system
SFP	spent fuel pit
SFPCS	spent fuel pit cooling and purification system
SG	steam generator

ACRONYMS AND ABBREVIATIONS (CONTINUED)

SGBDS	steam generator blowdown system
VHRA	very high radiation area
WMS	waste management system

12.0 RADIATION PROTECTION

12.1 Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable

The US-APWR design keeps all personnel radiation exposures within the limits defined by Title 10, Code of Federal Regulations, Part 20 (Reference 12.1-1). Administrative procedures and practices used in the US-APWR for maintaining personnel radiation exposure as low as reasonably achievable (ALARA) are described below, referencing NEI 07-08A (Reference 12.1-2) submitted in October 2009 to the U.S. Nuclear Regulatory Commission (NRC).

12.1.1 Policy Considerations

The facility design, administrative programs and procedures ensure that occupational radiation exposure is kept ALARA. The design and the operation of the US-APWR achieve ALARA occupational radiation exposures.

12.1.1.1 Design Policies

The US-APWR design takes into account the ALARA philosophy to reduce occupational radiation exposure during normal operation and accident conditions. The ALARA philosophy was applied during the initial design of the plant and implemented through internal design reviews. The design has been reviewed in detail for ALARA considerations, and will be reviewed, updated, and modified, as necessary, during the detail design phase, and as experience is obtained from operating plants. Nuclear engineers with extensive experience in ALARA design and operation reviewed the plant design, integrated the layout, shielding, ventilation, and monitoring instrument designs with the traffic control, security, access control, and health physics aspects of the design and operation to ensure that the overall design is conducive to maintaining exposures ALARA.

All pipe routing containing radioactive fluids is reviewed as part of the engineering design effort. This ensures that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize personnel exposure.

Lessons learned from operating plants are continuously integrated into the design of the US-APWR.

ALARA is incorporated into the radiation protection design and integrated into the design in the development of the overall design.

12.1.1.2 Operation Policies

Operation policies comply with 10 CFR 20 (Reference 12.1-1) and Regulatory Guides (RG) 1.8, 8.8 and 8.10 (Reference 12.1-3, 12.1-4 and 12.1-5) to ensure that occupational radiation exposures are ALARA.

The activities conducted by management personnel who have plant operational responsibility for radiation protection is described in Subsection 12.1.3.

12.1.1.3 Compliance with Title 10, Code of Federal Regulations, Part 20 and Regulatory Guides 1.8, 8.8, and 8.10

The US-APWR design and facility operation complies with 10 CFR 20 (Reference 12.1-1) and the guidelines of Regulatory Guides (RG) 1.8, 8.8, and 8.10 (Reference 12.1-3, 12.1-4, and 12.1-5).

12.1.1.3.1 Compliance with Regulatory Guide 1.8

The policy considerations regarding plant operations contained in RG 1.8 (Reference 12.1-3) are the responsibility of the COL Applicant. See Subsection 12.1.4 for the COL information.

12.1.1.3.2 Compliance with Regulatory Guide 8.8

The design of the US-APWR plant meets the guidelines of RG 8.8, Sections C.2, and C.4 (Reference 12.1-4) regarding facility, equipment, and instrumentation design features. Examples of the features of the plant that demonstrate compliance with RG 8.8 (Reference 12.1-4) are delineated in Section 12.3.

The policy considerations regarding plant operations contained in RG 8.8 (Reference 12.1-4) are the responsibility of the COL Applicant. See Subsection 12.1.4 for COL information.

12.1.1.3.3 Compliance with Regulatory Guide 8.10

The policy considerations regarding plant operations contained in RG 8.10 (Reference 12.1-5) are the responsibility of the COL Applicant. See Subsection 12.1.4 for the COL information.

12.1.2 Design Considerations

This subsection discusses the methods and features by which the policy considerations of Subsection 12.1.1 are applied. Operating experience from other nuclear plants was used in the design of the US-APWR. ALARA design requirements were prepared and distributed to every related design section, and engineers in those sections took into account the requirements for ALARA in their activities. The design and other provisions for maintaining personnel exposures ALARA are presented in detail in Subsections 12.3.1 and 12.3.2.

12.1.2.1 General Design Considerations for Keeping Exposures ALARA

General design considerations and methods employed to maintain in-plant radiation exposures ALARA, consistent with the recommendations of RG 8.8 (Reference 12.1-4), have the following two objectives:

- Minimizing the amount of personnel time spent in radiation areas
- Minimizing radiation levels in routinely occupied plant areas near plant equipment expected to require personnel attention

Both the equipment and the facility designs are considered in maintaining exposures ALARA during plant operations. The events considered include normal operation, maintenance, repairs, refueling operations, fuel storage, in-service inspection (ISI), calibrations, and radioactive waste handling and disposal.

The features of the plant design that ensure that the plant can be operated and maintained with exposures ALARA also apply during the decommissioning process and include the following:

- Provisions for draining, flushing, and decontaminating equipment and piping
- Design of equipment to minimize the buildup of radioactive material and to facilitate flushing crud traps
- Shielding which provides protection during maintenance or repair operations
- Provision for means and adequate space to use movable shielding
- Separation of more highly radioactive equipment from less radioactive equipment
- Provision for separate shielded compartments for adjacent items of radioactive equipment
- Provision for access to hatches to install or remove plant components
- Provision for design features to minimize crud buildup
- Countermeasures of design and water chemistry control to reduce radiation exposure such as:
  - Low Cobalt material
  - Low corrosive material
  - Zircaloy grid fuel
  - Modified pH control
  - Zinc injection
  - Increase of CVCS purification rate during shutdown
- Improved hot function test chemistry (dissolved Hydrogen and Lithium addition)

The design incorporates almost forty years of research and analysis and the operating histories of twenty three (23) Japanese PWR power plants, and includes various improvements over prior designs. The examples of these improvements are: (1) mechanisms for minimizing crud source generation; (2) greater understanding of crud behavior and buildup; (3) crud reduction methods in normal operation; (4) dose rate distribution during operation and shut down; (5) revised radiation streaming behavior; (6) reduction of radiation streaming; (7) reduced maintenance and inspection time; and (8) data collection and validation experiments for each of the preceding items.

Whenever new data were found or new situations occurred, they were reported to the engineering and/or research departments so that these phenomena, their mechanisms, reason for occurrence and preventive measures are investigated and resolved. The results of these investigations were reflected in the latest designs.

Practically these actions are used to be held in the case that unexpected high dose rate or radioactivity will be appeared or clarification with dose measurement will be carried out to the design to be expected to lead high dose rate or radioactivity. The measurement data is provided to the engineering and/or research departments in charge of investigation for countermeasure. The responsible departments will investigate the design or operation procedure for countermeasures and review them with related departments. The approved new design or operation procedure will be applied to plant operation or planning, and verified in the field with feedback to the latest plant design or operation procedure.

12.1.2.2 Equipment Design Considerations for Keeping Exposures ALARA

12.1.2.2.1 General Design Criteria

The component designers and engineers have been instructed to comply with ALARA. The design procedures require that the component design engineer consider the applicable RGs (including RG 8.8 [Reference 12.1-4]) as a part of the design criteria. Thus, the radiation protection issues of a component or system are taken into account for each component design or modification. The following paragraphs provide some examples of design considerations made to implement ALARA.

Consistent with the requirements of 10 CFR 20.1406 (Reference 12.1-6), the design minimizes the possibility for contamination of the facility or the environment, facilitates decommissioning, and reduces the generation of radwaste.

Examples of the system design to minimize the possibility for contamination are described below.

- The basic plant layout is planned to minimize the spread of contamination.
- Radioactive and potentially radioactive drains are separated from non-radioactive drains.
- The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.

- Ventilation systems are designed for minimizing the spread of airborne contamination
- In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than the air intake into that area.
- Overflow lines of tanks are directed to the waste collection system to control any contamination within plant structures.
- Tank vents are hard-piped to heating, ventilation, and air conditioning (HVAC) ducts, not to open room spaces.
- Equipment vents and drains from highly radioactive systems are piped directly to the collection system.
- All-welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at the joints.
- The valves in some radioactive systems are provided with leak-off connections piped directly to the collection system.
- Floor drains are provided to recover radioactive leakage.
- Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.
- Refueling tools have smooth surfaces to reduce contamination.

12.1.2.2.2 Considerations to Limit Time Spent in Radiation Areas

The equipment is designed such that access to instrumentation and controls is easy during normal and abnormal operating conditions, or for remote operation.

The equipment is selected to minimize the potential dose to personnel during maintenance.

The equipment is designed with specific drainage to facilitate maintenance.

The equipment is designed with smooth surfaces to reduce potential contamination during use and operation.

The vessel and piping insulation is designed for easy removal.

12.1.2.2.3 Equipment Design Considerations to Limit Component Radiation Levels

The materials selected for the equipment chosen to meet environmental requirements and to avoid Stellite-containing materials coming in contact with the primary coolant system.

The primary system cleanup and filtration systems collect corrosion products to reduce distribution in piping systems, thereby reducing the equipment component radiation levels.

The equipment and piping are designed to reduce the accumulation of radioactive materials in the equipment. The piping, where possible, is constructed of seamless pipe as a means to reduce possible radiation accumulation on seams.

The design of the equipment includes provisions to limit leaks or to control fluid leaks. These provisions include piping for the released fluid to the sumps and using drip pans with drainage piped to the floor drains.

12.1.2.3 Facility Layout General Design Considerations for Keeping Exposures ALARA

12.1.2.3.1 Minimizing Personnel Time Spent in Radiation Areas

The general design considerations used in the design of the facility to minimize the length of time spent in radiation areas include:

- Locating equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection so that they are easily accessible.
- Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment
- Where practicable, transporting equipment or components requiring service to a lower radiation area

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

The general design considerations used in the design of the facility to minimize radiation levels in plant access areas and near equipment requiring personnel attention include:

- Separating radiation sources and occupied areas where practicable (e.g., keeping pipes or ducts containing fluids with the potential for high radiation levels away from occupied areas)
- Providing adequate shielding between radiation sources and access and service areas

- Locating equipment, instruments, and sampling stations in the lowest practicable radiation zone
- Providing central control panels to remotely operate all essential instrumentation and controls in the lowest radiation zone practicable
- Where practicable, separating highly radioactive components from less radioactive components such as instruments and controls
- Providing means and adequate space for utilizing moveable shielding for sources within the service area when required
- Providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable as required by 10 CFR 20.1406 (Reference 12.1-6)
- Providing means to decontaminate service areas
- Providing space for pumps and valves outside of highly radioactive areas
- Providing for remotely-operated filter exchange for radwaste and cleanup systems
- Providing labyrinth entrances to radioactive equipment and valve rooms
- Providing adequate space in labyrinth entrances for easy access
- Maintaining ventilation airflow patterns from areas of lower potential airborne radioactivity to areas of higher potential airborne radioactivity

12.1.3 Operational Considerations

The COL Applicant is to describe how the plant complies with the guidance of RG 8.2 (Reference 12.1-7), 8.4 (Reference 12.1-8), 8.7 (Reference 12.1-10), 8.9 (Reference 12.1-11), 8.13 (Reference 12.1-12), 8.15 (Reference 12.1-13), 8.25 (Reference 12.1-15), 8.27 (Reference 12.1-17), 8.28 (Reference 12.1-18), 8.29 (Reference 12.1-19), 8.34 (Reference 12.1-21), 8.35 (Reference 12.1-22), 8.36 (Reference 12.1-23), and 8.38 (Reference 12.1-24).

In addition, the COL Applicant will describe the operational radiation protection program for ensuring that occupational radiation exposures are ALARA. This program is to be developed, implemented and maintained as described in the Nuclear Energy Institute Technical Report, NEI 07-03A (Reference 12.1-25), including compliance with the relevant quality assurance guidance provided in RG 1.33 (Reference 12.1-26). The specific CFR criteria referenced in NEI 07-03A shall be met and strictly adhered to. All recommendations and guidance referenced in NEI 07-03A are to be addressed and implemented where applicable to the US-APWR and the plant site.

Operational procedures will be developed following the guidance of RG 4.21 (Reference 12.1-27), for the operation and handling of all structures, systems, and components

(SSC) which could be potential sources of contamination within the plant. These procedures will be developed according to the objective of limiting leakage and the spread of contamination within the plant. See Subsection 12.1.4 for COL information.

12.1.4 Combined License Information

COL 12.1(1) *The COL Applicant will demonstrate that the policy considerations regarding plant operations comply with RG 1.8, 8.8 and 8.10 (Subsection 12.1.1.3).*

COL 12.1(2) *Deleted.*

COL 12.1(3) *The COL Applicant will describe how the plant complies with the guidance of RG 8.2, 8.4, 8.7, 8.9, 8.13, 8.15, 8.25, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36 and 8.38.*

COL 12.1(4) *Deleted.*

COL 12.1(5) *The COL Applicant will describe the operational radiation protection program that ensures occupational radiation exposures are ALARA.*

COL 12.1(6) *The COL Applicant will describe how the periodic review of operational practices to ensure configuration management, personnel training and qualification update, and procedure adherence.*

COL 12.1(7) *The COL Applicant will describe how implementation of requirements for record retention are tracked according to 10 CFR 50.75(g) and 10 CFR 70.25(g) as applicable is implemented.*

COL 12.1(8) *The COL Applicant is responsible for the development of the operational procedures following the guidance of RG 4.21 (Reference 12.1-27) for the operation and handling of all structures, systems, and components (SSC) which could be potential sources of contamination within the plant. These procedures will be developed according to the objective of limiting leakage and the spread of contamination within the plant.*

12.1.5 References

12.1-1 "Standards for Protection Against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20, U.S. Nuclear Regulatory Commission, Washington, DC, May 1991.

12.1-2 Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA). NEI Technical Report 07-08A, Revision 0, Oct. 2009.

12.1-3 Qualification and Training of Personnel for Nuclear Power Plants. RG 1.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, May 2000.

12.1-4 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. RG 8.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

12.1-5 Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable. RG 8.10, Rev. 1-R, U.S. Nuclear Regulatory Commission, Washington, DC, May 1977.

12.1-6 "Minimization of Contamination." Energy. Title 10 Code of Federal Regulations, Part 20.1406, U.S. Nuclear Regulatory Commission, Washington, DC.

12.1-7 Guide for Administrative Practices in Radiation Monitoring. RG 8.2, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973.

12.1-8 Direct-Reading and Indirect-Reading Pocket Dosimeters. RG 8.4, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973.

12.1-9 Deleted.

12.1-10 Instructions for Recording and Reporting Occupational Radiation Exposure Data. RG 8.7, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2005.

12.1-11 Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program RG 8.9, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, July 1993.

12.1-12 Instruction Concerning Prenatal Radiation Exposure. RG 8.13, Rev.3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1999.

12.1-13 Acceptable Programs for Respiratory Protection. RG 8.15, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, October 1999.

12.1-14 Deleted.

12.1-15 Air Sampling in the Workplace. RG 8.25, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1992.

12.1-16 Deleted.

12.1-17 Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants. RG 8.27, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981.

12.1-18 Audible-Alarm Dosimeters. RG 8.28, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, August 1981.

12.1-19 Instruction Concerning Risks from Occupational Radiation Exposure. RG 8.29, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, February 1996.

12.1-20 Deleted.

12.1-21 Monitoring Criteria and Methods To Calculate Occupational Radiation Doses. RG 8.34, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, July 1992.

12.1-22 Planned Special Exposures. RG 8.35, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 1992.

12.1-23 Radiation Dose to the Embryo/Fetus. RG 8.36, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, July 1992.

12.1-24 Control of Access to High and Very High Radiation Areas of Nuclear Plants. RG 8.38, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, May 2006.

12.1-25 Generic FSAR Template Guidance for Radiation Protection Program Description. NEI Technical Report 07-03A Revision 0, May. 2009.

12.1-26 Quality Assurance Program Requirements (Operation). RG 1.33, Rev.2, U.S. Nuclear Regulatory Commission, Washington, DC, February 1978.

12.1-27 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

12.1-28 Generic FSAR Template Guidance for Life Cycle Minimization of Contamination. NEI Technical Report 08-08A, Revision 0, October 2009.

12.2 Radiation Sources

This section discusses and identifies the sources of radiation that form the basis for the shielding design calculations, and the sources of airborne radioactivity that are used to establish personnel protection measures and perform dose assessments.

12.2.1 Contained Sources

The basis of the shielding design source terms are the plant conditions: (1) normal full-power operation; (2) shutdown; and (3) design-basis accident (DBA) events.

12.2.1.1 Sources for Full-Power Operation

The sources of radiation during normal full-power operation are direct core radiation, coolant activation processes, the leakage of fission products from defects in the fuel rod cladding, and the activation of the reactor coolant corrosion products. The design basis for the shielding source terms for the fission products for full-power operation is cladding defects in the fuel rods producing 1% of the core thermal power. The design basis for the activation of coolant is calculated independently of the fuel defect level. The design basis for the activation of corrosion products is derived from measurements at operating plants and is independent of the fuel defect level. The design basis of 1% fuel defects for the shielding source terms is used to establish shielding provisions for the auxiliary building (A/B). The outline of sources and estimation methods of source strength are described below, and source characteristics are tabulated in Table 12.2-1.

12.2.1.1.1 Reactor Core

The primary radiation from the reactor core during normal operation is neutrons and gamma rays. Figures 12.2-1 and 12.2-2 show distribution of neutron and gamma ray multigroup fluxes incident on the primary shield wall at the core centerline.

Figure 12.2-3 shows distribution of gamma ray dose rate incident on the primary shield wall at the core midplane. These figures are based on the same core power distribution used for the vessel irradiation estimation in Chapter 4, Subsection 4.3.2.8. Dose rate from the reactor core on the outer surface of the primary shield wall is less than 1 mrem/h, when radiation penetrates the bulk shielding, and less than 100 rem/h by the streaming through the penetration of the reactor coolant pipe in the primary shield wall.

Table 12.2-2 lists the core gamma ray sources after shutdown for the estimation of radiation levels within and around the shutdown reactor.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the reactor coolant system (RCS) are fission products released from fuel and activation and corrosion products that circulate in the reactor coolant. These sources and their bases are discussed in Chapter 11, Section 11.1. Subsection 11.1 estimates the design basis as well as the realistic source terms in the reactor coolant. In shielding design, only the design basis reactor coolant source terms are considered, and are calculated using the ORIGEN code, but without using methods described in ANSI/ANS-18.1-1999.

The activation product, N-16, is the predominant contributor to the activity in the reactor coolant pumps (RCPs), steam generators (SGs), and reactor coolant piping during operation. The N-16 activity in each of the components depends on the total transit time to the component and the average residence time in the core.

Table 12.2-3 presents the RCS N-16 activity as a function of transport time in a reactor coolant loop. The N-16 source strength for the pressurizer is tabulated in Table 12.2-4.

N-16 activity is not a factor in the radiation source term for systems and components located outside the containment due to its short, 7.35 seconds, half-life, and a transport time of greater than 1 minute before the primary coolant goes out of the containment.

Fission and corrosion product activities circulating in the RCS and out-of-core corrosion products comprise the remaining significant radiation sources during full-power operation. The fission and corrosion product activities circulating in the reactor coolant are given in Chapter 11, Subsection 11.1.1. The fission and corrosion product source strength in the reactor coolant pressurizer liquid phase are tabulated in Tables 12.2-5 and 12.2-6. The fission and corrosion product source strength and activity in the vapor phase are tabulated in Tables 12.2-7 and 12.2-8. The isotopic composition and specific activity of typical out-of-core corrosion products in the primary coolant are tabulated in Table 12.2-9. Crud trap areas may contain significantly higher activity levels than smooth surface areas.

Crud traps generally appear in the following areas:

- Locations of high turbulence
- Areas of high momentum change
- Gravitational sedimentation areas
- High affinity material areas
- Thin boundary layer regions

12.2.1.1.3 Chemical and Volume Control System

Radiation sources in the chemical and volume control system (CVCS) are derived from radionuclides carried in the reactor coolant. The design of the CVCS ensures that most of the N-16 decays before the letdown stream leaves the containment by the long letdown flowpath. The CVCS heat exchangers, except for the regenerative heat exchanger, letdown heat exchanger, and excess letdown heat exchanger are located in the R/B.

The shielding design is based on the maximum activity in each component. These sources are tabulated in Tables 12.2-10 through 12.2-29.

A. CVCS heat exchangers

The regenerative, letdown, and excess letdown heat exchangers are located in the containment. These components provide the primary-stage cooling for the reactor coolant letdown. The radiation sources for these components include N-16.

The magnitude of the N-16 source strength is highly sensitive to the location of these heat exchangers with respect to the RCS loop piping. Therefore, the N-16 source strengths for these heat exchangers are based on the coolant travel time from the reactor to each heat exchanger.

The letdown heat exchanger provides second-stage cooling for reactor coolant prior to entering the demineralizers. The seal water heat exchanger cools water from several sources, including reactor coolant discharged from the excess letdown heat exchangers.

Source strengths for the shell side of the regenerative heat exchanger are derived from the radionuclides contained in the liquid phase of the volume control tank.

B. CVCS demineralizer

The mixed bed demineralizer is in continuous use and removes fission products in cation and anion forms. It is highly effective in removing corrosion products. The cation bed demineralizer is used intermittently to remove lithium for pH control. It also is highly effective in removing the monovalent cations, cesium, and rubidium. The short-lived isotopes are assumed to build up to saturation activities on both beds. Radiation sources of these demineralizers are based on the accumulation of various ions in the coolant during the operation period.

The B. A. evaporator feed demineralizer is a mixed-bed style and removes ionic impurities from the reactor coolant.

C. CVCS filters

The design criterion for CVCS filter shielding is based primarily on operating experience.

The source strength for the reactor coolant filter corresponds to a dose rate of 500 rem/h at contact. The source strength for the remaining filters corresponds to a dose rate of 100 rem/h at contact except for the boric acid filter (10 rem/h at contact). These dose rates are calculated by assuming that impurities are distributed in the annular cylindrical shaped filter elements. Cobalt-60 is adopted as the representative nuclide for this conservative calculation.

D. Tanks

- Volume control tank

  The radiation sources in the volume control tank are based on a nominal operating level in the tank of 400 ft^3 in the liquid phase, 270 ft^3 in the vapor phase, and on the stripping fractions tabulated in Table 12.2-30, assuming no purge of the volume control tank. The values of the stripping fraction with assumption of purging are given in Table 12.2-31.

- Holdup tank

  The radiation sources in the holdup tank are based on the maximum activity for both the liquid phase and vapor phase considering continuous inflow of the coolant and the decay during storage.

E. B. A. evaporator

The B. A. evaporator is used to remove nitrogen, hydrogen, and gaseous fission products from the reactor coolant and to concentrate the remaining borated water for reuse in the RCS. Effluent from the holdup tanks is processed by the B.A evaporator feed demineralizer, and the primary coolant that has been processed is received by the boron recycle system using the B.A. evaporator feed pumps. The primary coolant is condensed by the B.A. evaporator, and the coolant, after condensation, is sent to the boric acid transfer pumps; separated non-condensible and noble gases pass through the B.A. evaporator vent condenser to be disposed of by the GWMS. The source term in the B. A. evaporator is based on the intermittent processing of the coolant. The source terms for the B. A. evaporator and B. A. evaporator vent condenser are tabulated in Tables 12.2-66 through 12.2-69.

F. Boric acid tank

Boric acid tanks receive the concentrate processed by the B.A. evaporator intermittently. Boric acid in the boric acid tanks is reused as primary coolant after adjustment of the concentration by the boric acid blender.

12.2.1.1.4 Essential Service Water System and Component Cooling Water System

The essential service water system and the component cooling water system are normally non-radioactive or, because of inleakage, have very low activity. Radiation monitoring for these systems is described in Chapter 11, Subsection 11.5.2. For shielding and dose assessment purposes, the essential service water system and component cooling water system do not yield substantive doses.

12.2.1.1.5 Spent Fuel Pit Cooling and Purification System

Sources in the spent fuel pit (SFP) cooling and purification system (SFPCS) result from the transfer of radioactive isotopes from the reactor coolant into the SFP during refueling operations.

The reactor coolant activities for fission, corrosion, and activation products decay during the time required to remove the reactor vessel head following shutdown. Activity is reduced by operation of the CVCS purification demineralizers, and diluted by the total volume of the water in the reactor vessel, refueling water storage pit, and the SFP. This activity then undergoes subsequent decay and accumulation on the SFP cooling and purification system filters and demineralizer.

In the shielding design, the activity of SFP water is determined assuming the presence of only Cobalt-60 which generates a dose rate at the pit surface of up to 15 mrem/h (Zone IV levels). Fission products in the reactor coolant are negligible today due to technological improvements in nuclear fuel integrity resulting in a reduced fuel defect fraction. Activities of corrosion products are estimated as Cobalt-60 considering gamma emission energy for each nuclide. The dose rate due to the radiation from both the spent fuel assembly during fuel handling and the contaminated water in the SFP is 15 mrem/h at the SFP water surface.

The activities in the SFP are tabulated in Table 12.2-32. The source terms for the SFP demineralizers and filters are provided in Tables 12.2-33 through 12.2-34.

12.2.1.1.6 Main Steam System

Potential radioactivity in the main steam system is a result of the SG tube leaks and fuel defects.

This radioactivity is sufficiently low that no radiation shielding is needed for equipment in secondary systems, other than portions of the steam generator blowdown system (SGBDS) where it is required to meet radiation zone requirements.

For the purpose of evaluating SGBDS, the radioactivity in the main steam system is based on a SG tube leakage rate of 150 gallons per day concurrent with a 1% failed fuel. Continuous operation with primary-to-secondary leakage is assumed. The RCS radionuclide concentrations used are tabulated in Table 11.1-2. The treatment of SG secondary side water and steam is discussed in Chapter 10, Subsection 10.4.8.

The source terms for the steam generator blowdown demineralizer are tabulated in Tables 12.2-35 and 12.2-36.

12.2.1.1.7 Liquid Waste Management System

Radioactive inputs to the liquid waste management system (LWMS) sources include fission and activation product radionuclides produced in the core and reactor coolant. The components of the radwaste systems contain varying degrees of activity.

The concentrations of radionuclides present in the process fluids at various locations in the radwaste systems, such as pipes, tanks, filters, and demineralizers are based on system activities discussed in Chapter 11, Section 11.1 and 11.2. Shielding for each component of the LWMS is based on the maximum activity conditions shown in Tables 12.2-37 through 12.2-43. Radiation sources in the various pumps in the system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

12.2.1.1.8 Gaseous Waste Management System

The gaseous waste management system (GWMS) consists of the gaseous surge tank subsystem and the charcoal bed delay subsystem.

The radiation source for each component of the GWMS is based on operating conditions as given in Chapter 11, Sections 11.1 and 11.3.

Tables 12.2-44 and 12.2-45 show the distribution of the radioactive gas inventory and the gamma ray source strength associated with operation of the GWMS is conservative. The purge of the volume control tank is not taken into consideration. The calculated values represent the design activity distribution with 1% fuel defect.

The volume control tank stripping fractions used in establishing the activity distributions are tabulated in Table 12.2-30.

The radioactive gases removed from the RCS at the volume control tank are continuously re-circulated through a waste gas surge tank and other GWMS equipment, including the waste gas compressors. The gamma ray source strengths for the waste gas surge tank are derived from refueling shutdown procedures during which the radioactive gases are stripped from the RCS. Tables 12.2-46 and 12.2-47 tabulate the activities and gamma ray source strengths for the waste gas surge tank.

12.2.1.1.9 Solid Waste Management System

The only fixed component with radiation sources in the solid waste management system (SWMS) is the spent resin storage tank. The spent resin storage tank receives the spent resin from each demineralizer. The radiation source of the spent resin storage tank is based on the stored spent resin from the CVCS demineralizers because radioactivities of the spent resin of the other demineralizers are lower than that of the CVCS demineralizers. Tables 12.2-48 and 12.2-49 tabulate the activities for the spent resin storage tank.

12.2.1.1.10 Miscellaneous Sources

The principal sources of activity outside the buildings but inside the tank house include the following:

- The refueling water storage auxiliary tank
- The primary makeup water tank

The content of the water tanks is processed by the SFP purification system, or the boron recycle system until the activity in the fluids is sufficiently low to result in dose rates less than 0.25 mrem/h at 2 meters from the surface of the tank.

Radionuclide inventories of the refueling water storage auxiliary tank and primary makeup water tank are presented in Tables 12.2-50 and 12.2-51. There are no other significant amounts of radioactive fluids permanently stored outside the buildings.

Spent fuels are stored in the SFP. When the fuel is to be moved away from the SFP, it is placed in a spent fuel shipping cask for transport.

Storage space is allocated in the radwaste processing facility for storage of spent filter cartridges and packaged spent resins.

Radioactive wastes stored inside the plant structures are shielded so that areas outside the structures meet Radiation Zone I criteria. Additional storage space for radwaste is to be provided in the detailed design by the COL Applicant. If it becomes necessary to temporarily store radioactive wastes/materials outside the plant structures, radiation protection measures are to be taken by the radiation protection staff to ensure compliance with 10 CFR 20 (Reference 12.2-1), 40 CFR 190 (Reference 12.2-6) and to be consistent with the recommendations of RG 8.8 (Reference 12.2-2).

The SWMS facilities process and store dry active waste. If it becomes necessary to install additional radwaste facilities for dry active waste, it is to be provided by the COL Applicant. Radiation shielding is provided such that the dose rates comply with the requirements of 10 CFR 20 (Reference 12.2-1) and 40 CFR 190 (Reference 12.2-6). Interior concrete shielding is provided to limit exposure to personnel during waste processing. The ALARA methodology of RGs 8.8 (Reference 12.2-2) and 8.10 (Reference 12.2-3) has been used in the design of this facility.

Any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography, are described by the COL Applicant.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent fuel, the residual heat removal system (RHRS), and the incore instrumentation system (ICIS). Individual components may require shielding during shutdown due to fission and activation products in spent fuel, deposited crud material and the induced activity of the ICIS. The estimates of accumulated crud are given in Subsection 12.2.1.1.2. The radiation sources in the RCS and other systems addressed in Subsection 12.2.1.1 are bounded by the sources given for the full power operation with the exception of a short time period (i.e., less than 24 hours) following shutdown, during which the fission product spiking phenomenon and crud bursts can result in increased radiation sources. The spiking phenomenon involves the release of a portion of the accumulated water-soluble salts from the interior cladding surface (e.g., iodine, cesium, and gases [e.g., xenon and krypton]) of defective fuel rods during the shutdown and coolant depressurization.

Crud bursts are the resuspension or solubilization of a portion of the accumulated deposited corrosion products into the RCS during shutdown such as during oxygenization of the reactor coolant. However, special shielding considerations to accommodate these increases should be unnecessary due to several factors, including the following:

- The spike or crud burst release is of short duration (generally less than 6 hours).
- The CVCS is generally in operation at full reactor coolant purification capability during the shutdown.

12.2.1.2.1 Residual Heat Removal System

Radionuclide activities and maximum gamma source strengths in the RHRS at 4 hours after the reactor shutdown are identified in Tables 12.2-52 and 12.2-53. The system may be placed in operation at approximately 4 hours following a shutdown at the maximum cooldown rate. The system removes decay heat from the reactor for the duration of the shutdown. The sources given are the maximum values with credit for 4 hours of fission and corrosion product decay and purification.

12.2.1.2.2 Reactor Core

Core average gamma ray source strengths are tabulated in Table 12.2-2. These source strengths are used in the evaluation of radiation levels within and around the shutdown reactor.

For source strength calculation, it is assumed that the core has two regions and the irradiation time is 28 months to conservatively bound cycle lengths up to 24 months. The specific power is 32.0 MW/MTU as described in Chapter 4, Table 4.4-1. In this calculation, the specific power was rounded up a fraction to 32.1 MW/MTU. These calculation conditions lead to fission and activation products generated in fuel with burnup of about 55 GWD/MTU in two cycles.

12.2.1.2.3 Spent Fuel

The predominant radioactivity sources in the spent fuel storage and transfer areas in the Reactor Building (R/B) and the refueling cavity and fuel handling areas in the pre-stressed concrete containment vessel (PCCV) are the spent fuel assemblies. The source strengths employed to determine the minimum water depth above spent fuel and shielding walls around the SFP, the spent fuel transfer tube, refueling cavity, and fuel handling area are tabulated in Table 12.2-54. For the shielding design, the SFP and refueling cavity containment racks are assumed to contain the maximum number of fuel assemblies. To be conservative, 257 spent fuel assemblies, assumed to be from unloading the full core with only a 24-hour decay period, are assumed to be located in the outer rows of the spent fuel racks. The remaining assemblies, from previous refueling operations, do not significantly affect the shield wall design due to the shielding of the intervening, recently discharged assemblies. To be conservative, six spent fuel assemblies, assumed to be from unloading the core with only a 24-hour decay period, are assumed to fill the containment racks located in the refueling cavity.

The source strengths in Table 12.2-54 are also used in the evaluation of radiation levels for spent fuel handling, storage, and shipping. These sources are calculated using the ORIGEN code, based on a specific power of 32.1 MW/MTU and a burnup of 62 GWD/MTU, which is a limitation for maximum burnup for fuel rod as described in Chapter 4, Subsection 4.2.1. Other calculation parameters are tabulated in Table 12.2-70.

12.2.1.2.4 Control Rods, Primary and Secondary Source Rods

As source material, byproduct material or special nuclear material, there are primary and secondary source rods. As described in Chapter 4, Subsection 4.2.2.3 and 4.2.2.3.3, a primary source rod contains californium-252 source, and a secondary source rod contains antimony-beryllium source. These rods are stored in the SFP after use. Irradiated control rods are also stored in the SFP. Source strengths of these rods are less than that of spent fuel. Therefore, in radiation shielding design, source strengths of spent fuel are used as these rods' source strengths.

12.2.1.2.5 Incore Flux Thimbles

Irradiated incore detector and drive cable maximum gamma ray source strengths are tabulated in Table 12.2-55. These source strengths are used in determining shielding requirements and evaluating occupational radiation exposure when detectors are being moved during or following a flux mapping of the reactor core. These source strengths are given an irradiation period of 20 hours, respectively, and are given in terms of per cubic centimeters (cm^3) of detector and per centimeters of drive cable. The irradiation of the small amount of uranium contained in the fission chamber contributes to the source term of the incore detector; however, this fission product source term is insignificant with respect to the drive cable source term due to the long length of irradiated cable.

Irradiated incore detector drive cable average gamma ray source strengths are tabulated in Table 12.2-56. These source strengths are used in determining shielding requirements when the detectors are not in use and for shipment when the detectors have failed. The values are given in terms of per centimeters of drive cable after an irradiation period of 20 hours.

Irradiated incore flux thimble gamma ray source strengths are tabulated in Table 12.2-57. These source strengths are used in determining shielding requirements during refueling operations when the flux thimbles are withdrawn from the reactor core. The values are given in terms of per cm^3 stainless steel for an irradiation period of 60 years. The flux thimbles are made of type 316 stainless steel with a maximum cobalt impurity content of 0.10 weight percent.

All these activities are calculated using the following equation:

$$A = \frac{1}{3.7 \times 10^4} N \cdot \sigma \cdot \varphi [1 - \exp(-\lambda t_1)] \cdot \exp(-\lambda t_2) \qquad \text{Eq. 12.2-3}$$

Where:

A = activity ($\mu Ci/cm^3$)

N = isotope number density ($1/cm^3$)

σ = activation cross section (cm^2)

φ = neutron flux ($n/cm^2/s$)

λ = decay constant (1/s)

t_1 = irradiation period (s)

t_2 = time after shutdown (s)

Other calculation parameters are tabulated in Table 12.2-71.

12.2.1.3 Sources for the Design-Basis Accident

The radiation sources of importance for the DBA are the containment source and the RHRS and Containment Spray System sources.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in RG 1.183 (Reference 12.2-4). The airborne radioactivity in containment is calculated based on the assumption that all the radioactive material released into containment is airborne. Decreases due to deposition, leakage, spray, or dissolving into the recirculation water are not taken into consideration. The gamma ray source strengths can be calculated with the MicroShield code using the airborne radioactivity concentration in containment. The beta source strengths are calculated by multiplying the airborne radioactivity concentration in containment by the effective energy of beta. The integrated gamma ray and beta particle source strengths for various time-periods following the postulated accident are tabulated in Table 12.2-58.

The RHRS and shielding are designed to allow limited access to the RHR pumps following a DBA. The sources are based on the assumptions given in RG 1.183 (Reference 12.2-4). The radioactivity in the recirculation water is calculated based on the assumption that all the radioactive material released into containment, except for noble gases, is dissolved in the recirculation water. Decreases due to deposition, leakage, or radioactivity being airborne in containment are not taken into consideration. The gamma ray source strengths can be calculated with the MicroShield code using the radioactivity concentration in recirculation water. The beta source strengths are calculated by multiplying the radioactivity concentration in recirculation water by the effective energy of beta. Noble gases formed by the decay of halogens in the sump water are assumed to be retained in the water. Credit has been taken for dilution by the RCS volume plus the contents of the refueling water storage. Gamma ray source strengths for radiation sources circulating in the RHR loop and associated equipment are tabulated in Table 12.2-59.

12.2.2 Airborne Radioactive Material Sources

This section deals with the models, parameters, and sources required to evaluate the airborne concentration of radionuclides during the plant operations in the various plant radiation areas where personnel occupancy is expected.

Radioactive material that becomes airborne may come from the RCS, spent fuel pit, and refueling water storage pit. The calculation of potential airborne radioactivity in equipment cubicles, corridors, or operating areas normally occupied by operating personnel is based on reactor coolant activities given in Chapter 11, Section 11.1.

The assumptions and parameters required to evaluate the isotopic airborne concentrations in the various applicable regions are tabulated in Table 12.2-60 and Table 12.2-72.

The CVCS and the RHRS are designed to provide the capability to purify the reactor coolant through the purification demineralizer after the reactor shutdown and cooldown.

This mode of operation will ensure that the effect of activity spikes does not significantly contribute to the containment airborne activity during refueling operations.

Sources resulting from the removal of the reactor vessel head and the movement of spent fuel are dependent on a number of operating characteristics (e.g., coolant chemistry, fuel performance) and operating procedures followed during and after shutdown. The permissible coolant activity levels following de-pressurization are based on the noble gases evolved from the RCS water upon the removal of the reactor vessel head. The endpoint limit for coolant cleanup and degasification is established based on the maximum permissible concentration considerations and containment ventilation system capabilities of the plant.

The exposure rates at the surface of the refueling cavity and spent fuel pit water are dependent on the purification capabilities of the refueling cavity and spent fuel pit cleanup systems. A water total activity level of less than 0.005 μCi/g for the dominant gamma-emitting isotopes at the time of refueling leads to a dose rate at the water surface less than 2.5 mrem/h.

The detailed listing of the expected airborne isotopic concentrations in all the various plant regions is presented in Table 12.2-61. The final design of the plant ensures that all the expected airborne isotopic concentrations in all normally occupied areas are well below the derived air concentration (10 CFR 20 Appendix B [Reference 12.2-5]). If entry is needed in areas where airborne concentrations exceed the limit (such as containment during normal operation), appropriate personnel protection equipment and radiological controls will be implemented to ensure that personnel doses are in compliance with 10 CFR 20 (Reference 12.2-1).

12.2.2.1 Containment Vessel Atmosphere

The expected airborne isotopic concentrations in the containment vessel atmosphere is presented in Table 12.2-61.

12.2.2.2 Reactor Building Atmosphere

The expected airborne isotopic concentrations in the R/B atmosphere is presented in Table 12.2-61.

12.2.2.3 Fuel-Handling Area Atmosphere

The expected airborne isotopic concentrations in the fuel handling area atmosphere is presented in Table 12.2-61.

12.2.2.4 Auxiliary Building Atmosphere

The expected airborne isotopic concentrations in the A/B is presented in Table 12.2-61.

12.2.2.5 Airborne Radioactivity Model

For those regions characterized by a constant leak rate of the radioactive source at constant source strength and a constant exhaust rate of the region, the peak or equilibrium airborne concentration of the radioisotope in the regions is calculated using the following equation:

$$C_i(t) = \frac{(LR)_i A_i (PF)_i [1 - \exp(-\lambda_{Ti} t)]}{V \lambda_{Ti}} \qquad \text{Eq. 12.2-1}$$

Where:

$(LR)_i$ = Leak or evaporation rate of the ith radioisotope in the applicable region (g/s)

A_i = Radioactivity concentration of the ith leaking or evaporating radioisotope (μCi/g)

$(PF)_i$ = Partition factor or the fraction of the leaking radioactivity that is airborne for the ith radioisotope

λ_{Ti} = Total removal rate constant for the ith radioisotope from the applicable region (1/s)

λ_{Ti} = $\lambda_{di} + \lambda_e$, the removal rate constants in 1/s due to radioactive decay for the ith radioisotope and the exhaust from the applicable region, respectively

λ_e = the exhaust removal rate in 1/s defined as Q/V

λ_{di} = the radioactive decay rate in 1/s for the i^{th} radioisotope

t = Time elapsed from the start of the leak and the time at which the concentration is evaluated (s)

V = Free volume of the region in which the leak occurs (cm^3)

Q = Ventilation flow rate (cm^3/s)

$C_i(t)$ = Airborne concentration of the ith radioisotope at time t in the applicable region ($\mu Ci/cm^3$)

From the above equation, it is evident that the peak or equilibrium concentration, C_i, of the ith radioisotope in the applicable region will be given by the following expression:

$$C_i(t) = \frac{(LR)_i A_i (PF)_i}{V \lambda_{Ti}} \qquad \text{Eq. 12.2-2}$$

With high exhaust rates, this peak concentration will be reached within a few hours.

As a conservative assumption, radioactive decay of the i[th] radioisotope is ignored. Using this assumption, Eq. 12.2-2 can be simplified as represented below:

$$C_i = \frac{(LR)iAi(PF)i}{V\left(\frac{Q}{V}\right)} = \frac{(LR)iAi(PF)i}{Q} \qquad \text{Eq. 12.2-4}$$

12.2.2.6 Sources Resulting from Design-Basis Accidents

The radiation sources from DBAs include the design basis inventory of radioactive isotopes in the reactor coolant, plus the postulated fission product released from the fuel. DBA parameters and sources are discussed and evaluated in Chapter 15, Subsection 15.6.5.5.

12.2.3 Combined License Information

COL 12.2(1) *The COL Applicant will list any additional contained radiation sources that are not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography.*

COL 12.2(2) *The COL Applicant will address the radiation protection aspects associated with additional storage space for radwaste and/or additional radwaste facilities for dry active waste.*

COL 12.2(3) *The COL Applicant will include the conduct of regular surveillance activities and provisions to maintain the dose rate at 2 meters from the surface of both the RWSAT and the PMWTs under 0.25 mrem/h in the Radiation Protection Program.*

COL 12.2(4) *The COL Applicant will implement a method of ensuring that the radioactivity concentration in both the RWSAT and the PMWTs remain under the specified concentration level described in the DCD.*

12.2.4 References

12.2-1 "Standards for Protection Against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20, U.S. Nuclear Regulatory Commission, Washington, DC, May 1991.

12.2-2 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. RG 8.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

12.2-3 Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable. RG 8.10, Rev. 1-R, U.S. Nuclear Regulatory Commission, Washington, DC, May 1977.

12.2-4 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. RG 1.183, Rev. 0, U.S. Nuclear Regulatory Commission, Washington, DC, July 2000.

12.2-5 "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," Energy. Title 10, Code of Federal Regulations, Part 20, Appendix B, U.S. Nuclear Regulatory Commission, Washington, DC.

12.2-6 "Environmental Radiation Protection Standards for Nuclear Power Operations," Protection of Environment. Title 40, Code of Federal Regulations, Part 190, U.S. Environmental Protection Agency, Washington DC, January 1977.

Table 12.2-1 Radiation Sources Parameters (Sheet 1 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density** (lb/ft^3)	Equipment Self-Shielding (in.)	
Inside the containment vessel							
Steam generator Plenum Side Shell Side	66.9 65.9	63.0 434.2	Homogeneous Homogeneous	Source Water Source Water 22 wt%+ Secondary Water 9wt%+ Steel 69wt%	44.5 69.2	6.1 3.4	4
Regenerative heat exchanger * Plenum Side Shell Side	8.3	23.2 132.4	Homogenous Homogenous	Water (Charging Line) Water (Letdown Line) 26 wt%+ Water (Charging Line) 6 wt%+ Steel 68 wt%	62.4 121.5	2.0	2
Letdown heat exchanger Plenum Side Shell Side	16.7	24.4 189.8	Homogenous Homogenous	Source Water Source Water 13 wt%+ Cooling water 57 wt%+ Steel 31 wt%	62.4 85.2	ignored	1
Excess letdown heat exchanger Plenum Side Shell Side	5.9	21.7 130.2	Homogenous Homogenous	Source Water Source Water 5 wt%+ Cooling water 54 wt%+ Steel 41wt%	62.4 96.7	1.8 ignored	1

* The regenerative heat exchanger consists of two shells.
**Gamma shielding calculations utilize a point kernel method, as described in Subsection 12.3.2.3. This method requires the use of a single mixed density for modeling complex geometries, such as the shell side of a heat exchanger. The shell side density values provided here are based on a weighted combination of the individual densities of the component cooling water outside the U-tubes, the stainless steel of the U-tubes, and the source water inside the tubes.

Table 12.2-1 Radiation Sources Parameters (Sheet 2 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density* (lb/ft^3)	Equipment Self-Shielding (in.)	
Outside the containment vessel (Reactor Building)							
Containment spray/residual heat removal heat exchanger	31.5						
Plenum Side		56.7	Homogenous	Source Water	62.4	1.8	4
Shell Side		264.4	Homogenous	Source Water 15 wt%+ Cooling water 48 wt%+ Steel 37 wt%	92.2	1.2	
Seal water heat exchanger	8.4					ignored	
Plenum Side		22.3	Homogenous	Source Water	62.4		
Shell Side		144.6	Homogenous	Source Water 10 wt%+ Cooling water 48 wt%+ Steel 42 wt%	98.2		1
Volume control tank	47.2	179.2				ignored	1
Liquid Phase		107.5	Homogenous	Air	7.6E-02		
Vapor Phase		71.7	Homogenous	Water	62.4		

* Gamma shielding calculations utilize a point kernel method, as described in Subsection 12.3.2.3. This method requires the use of a single mixed density for modeling complex geometries, such as the shell side of a heat exchanger. The shell side density values provided here are based on a weighted combination of the individual densities of the component cooling water outside the U-tubes, the stainless steel of the U-tubes, and the source water inside the tubes.

Table 12.2-1 Radiation Sources Parameters (Sheet 3 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Auxiliary Building							
Mix bed demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	2
Cation-bed demineralizer*	15.9	65.6	Homogeneous	Water	62.4	ignored	1
Deborating demineralizer*	23.7	68.9	Homogenous	Water	62.4	ignored	2
Holdup tank Liquid Phase Vapor Phase	147.6	410.0 229.7 180.3	 Homogenous Homogenous	 Water Air	 62.4 7.6E-02	ignored	3
B.A. evaporator feed demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	1
Spent fuel pit demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	2
Steam generator blowdown demineralizer*	44.3	63.4	Homogeneous	Water	62.4	ignored	4
Waste holdup tank	128.0	138.6	Homogeneous	Water	62.4	ignored	4
Waste demineralizer*	23.7	68.9	Homogeneous	Water	62.4	ignored	4
Charcoal bed Charcoal Phase Vapor Phase	23.7	126.0 68.8 57.2	 Homogenous Homogenous	 Charcoal Air	 34.4 7.6E-02	ignored	4
Waste gas surge tank	74.8	167.0	Homogeneous	Air	7.6E-02	1.0	4
Spent resin storage tank	59.1	131.2	Homogeneous	Water	62.4	ignored	2
B.A. evaporator	26.9	188.5	Homogeneous	Water	62.4	ignored	1
B.A. evaporator vent condenser	5.0	78.1	Homogeneous	Air	7.6E-02	ignored	1
Boric acid tank	118.1	361.5	Homogeneous	Water	62.4	ignored	2

* Parameters from the US-APWR demineralizers are tabulated in Table 12.2-73.

Table 12.2-1 Radiation Sources Parameters (Sheet 4 of 6)

Components	Assumed Shielding Sources						
	Source Approximate Geometry as Cylinder Volume		Source Characteristics				Quantity
	Radius (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Plant Yard Area (Outside the Power Block)							
Refueling water storage auxiliary tank	196.9	446.3	Homogeneous	Water	62.4	ignored	1
Primary makeup water tank	183.1	316.9	Homogeneous	Water	62.4	ignored	2

Table 12.2-1 Radiation Sources Parameters (Sheet 5 of 6)

Components	Assumed Shielding Sources							
	Source Approximate Geometry as rectangular parallelepiped Volume			Source Characteristics				Quantity
	Width (in.)	Depth (in.)	Length (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	
Outside the Containment Vessel (Reactor Building)								
Spent fuel pit heat exchanger *	29.5	47.7	88.6	Homogeneous	Source Water 25.5 wt%+ Cooling water 25.5 wt%+ Steel 49 wt%	109.3	ignored	2

* Spent fuel pit exchanger is a plate heat exchanger.

Table 12.2-1 Radiation Sources Parameters (Sheet 6 of 6)

Components	Assumed Shielding Sources								
	Source Approximate Geometry as Annular Cylinder Volume			Source Characteristics					Quantity
	Outer Radius (in.)	Inner Radius (in.)	Height (in.)	Type	Material	Density (lb/ft^3)	Equipment Self-Shielding (in.)	Designed Upper limit dose rate (mrem/h)	
Auxiliary Building									
Reactor coolant Filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	500	2
Mixed bed demineralizer inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	500	3
B.A. evaporator feed demineralizer filter	3.4	2.7	19.7	Homogeneous	Water	62.4	Ignored	100	1
Boric acid filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	10	1
Seal water injection filter	1.7	1.6	19.9	Homogeneous	Water	62.4	Ignored	100	2
Waste effluent inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	100	2
SFP filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	100	2
SG blowdown demineralizer inlet filter	6.4	5.2	19.7	Homogeneous	Water	62.4	Ignored	10	2

Table 12.2-2 Core Average Gamma Ray Source Strengths at Various Times after Shutdown (Sheet 1 of 2)

Gamma Ray Energy (MeV)	Source Strength at Various Times after Shutdown (MeV/watt-sec)			
	12 hours	24 hours	100 hours	7 days
0.01	2.8E+08	2.3E+08	1.2E+08	7.4E+07
0.025	9.0E+07	7.5E+07	4.7E+07	3.7E+07
0.0375	1.5E+08	1.3E+08	8.3E+07	6.4E+07
0.0575	1.7E+08	1.4E+08	8.5E+07	6.6E+07
0.085	4.5E+08	3.9E+08	2.1E+08	1.3E+08
0.125	1.6E+09	1.4E+09	6.6E+08	3.9E+08
0.225	2.3E+09	1.9E+09	8.0E+08	4.3E+08
0.375	1.0E+09	8.9E+08	5.7E+08	4.3E+08
0.575	5.2E+09	4.2E+09	2.5E+09	2.0E+09
0.85	6.6E+09	5.8E+09	4.2E+09	3.7E+09
1.25	2.0E+09	1.2E+09	5.5E+08	4.0E+08
1.75	3.2E+09	2.9E+09	2.4E+09	2.1E+09
2.25	3.0E+08	2.1E+08	1.6E+08	1.3E+08
2.75	1.7E+08	1.6E+08	1.4E+08	1.2E+08
3.5	5.0E+06	1.8E+06	1.5E+06	1.3E+06
5.0	3.2E+05	1.7E+04	7.2E+00	7.1E+00
7.0	1.2E+00	1.2E+00	1.2E+00	1.1E+00
9.5	1.8E-01	1.8E-01	1.8E-01	1.8E-01

Table 12.2-2 Core Average Gamma Ray Source Strengths at Various Times after Shutdown (Sheet 2 of 2)

Gamma Ray Energy (MeV)	Source Strength at Various Times after Shutdown (MeV/watt-sec)			
	30 days	100 days	180 days	365 days
0.01	2.7E+07	1.7E+07	1.3E+07	8.0E+06
0.025	1.7E+07	9.6E+06	7.1E+06	4.5E+06
0.0375	2.8E+07	1.5E+07	1.1E+07	7.0E+06
0.0575	3.1E+07	1.9E+07	1.5E+07	9.6E+06
0.085	3.5E+07	2.0E+07	1.5E+07	1.0E+07
0.125	1.1E+08	4.7E+07	2.9E+07	1.7E+07
0.225	7.6E+07	4.5E+07	3.5E+07	2.3E+07
0.375	1.2E+08	4.0E+07	3.1E+07	2.1E+07
0.575	1.0E+09	5.3E+08	3.9E+08	3.0E+08
0.85	2.6E+09	1.4E+09	7.4E+08	2.4E+08
1.25	1.3E+08	6.9E+07	5.9E+07	4.6E+07
1.75	6.2E+08	2.2E+07	6.6E+06	4.3E+06
2.25	5.4E+07	2.1E+07	1.6E+07	1.0E+07
2.75	3.6E+07	1.1E+06	2.8E+05	1.9E+05
3.5	4.3E+05	5.7E+04	4.2E+04	3.0E+04
5.0	6.9E+00	6.4E+00	6.0E+00	5.4E+00
7.0	1.1E+00	1.0E+00	9.7E-01	8.7E-01
9.5	1.7E-01	1.6E-01	1.5E-01	1.4E-01

Table 12.2-3 Radiation Sources Reactor Coolant Nitrogen-16 Specific Activity

Position in Loop	Loop Transit Time (s)	Nitrogen-16 Specific Activity (μCi/g)
Leaving core	0.0	330
Leaving reactor vessel	1.3	300
Entering SG	1.7	280
Leaving SG	7.5	160
Entering RCP	8.1	160
Entering reactor vessel	9.3	140
Entering core	11.6	110
Leaving core	12.6	330

Table 12.2-4 Nitrogen-16 Radiation Sources – Pressurizer

Gamma Ray Energy (MeV)	Specific Source Strength (MeV/(g-s))
2.0	2.8E+04
3.0	2.5E+05
6.0	4.5E+07
8.0	4.4E+06

Table 12.2-5 Pressurizer Liquid Phase Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/(g-s))
0.015	9.8E+03
0.02	2.4E+01
0.03	1.7E+05
0.04	8.3E+03
0.05	4.5E+01
0.06	5.7E+01
0.08	3.5E+05
0.1	6.6E+01
0.15	1.1E+04
0.2	8.1E+04
0.3	9.4E+03
0.4	3.8E+04
0.5	8.2E+04
0.6	7.2E+04
0.8	1.4E+05
1.0	8.2E+04
1.5	1.6E+05
2.0	2.8E+05
3.0	5.1E+04
4.0	8.4E+02
5.0	1.2E+03

Table 12.2-6 Pressurizer Liquid Phase Specific Activity

Nuclide	Specific Activity (µCi/g)	Nuclide	Specific Activity (µCi/g)
Kr-83m	4.6E-01	Te-129m	5.9E-03
Kr-85m	1.8E+00	Te-129	7.3E-03
Kr-85	9.3E+01	Te-131m	1.6E-02
Kr-87	1.2E+00	Te-131	8.5E-03
Kr-88	3.3E+00	Te-132	1.7E-01
Xe-131m	4.1E+00	Te-133m	1.6E-02
Xe-133m	4.2E+00	Te-134	2.9E-02
Xe-133	3.2E+02	I-130	6.3E-02
Xe-135m	7.7E-01	I-131	1.6E+00
Xe-135	1.0E+01	I-132	8.6E-01
Xe-138	6.7E-01	I-133	2.8E+00
		I-134	5.9E-01
Br-82	8.6E-03	I-135	1.8E+00
Br-83	7.8E-02	Cs-132	8.3E-04
Br-84	4.2E-02	Cs-134	7.7E-01
Rb-86	7.5E-03	Cs-135m	9.0E-03
Rb-88	4.3E+00	Cs-136	2.0E-01
Rb-89	9.8E-02	Cs-137	4.4E-01
Sr-89	1.9E-03	Cs-138	9.9E-01
Sr-90	1.2E-04	Ba-137m	4.1E-01
Sr-91	1.3E-03	Ba-140	2.3E-03
Sr-92	7.1E-04	La-140	6.0E-04
Y-90	2.8E-05	Ce-141	3.6E-04
Y-91m	6.6E-04	Ce-143	3.0E-04
Y-91	3.0E-04	Ce-144	2.7E-04
Y-92	5.5E-04	Pr-144	2.7E-04
Y-93	2.4E-04	Pm-147	3.0E-05
Zr-95	3.7E-04	Eu-154	2.8E-06
Nb-95	3.7E-04		
Mo-99	4.5E-01	Na-24	3.9E-02
Mo-101	2.0E-02	Cr-51	3.8E-03
Tc-99m	1.8E-01	Mn-54	2.6E-03
Ru-103	3.0E-04	Mn-56	1.3E-01
Ru-106	1.1E-04	Fe-55	2.5E-03
Ag-110m	9.8E-07	Fe-59	4.4E-04
Te-125m	4.4E-04	Co-58	6.1E-03
Te-127m	1.7E-03	Co-60	8.9E-04
		Zn-65	7.3E-04

Table 12.2-7 Pressurizer Vapor Phase Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/(cm^3-s))
0.015	5.1E+04
0.03	9.0E+05
0.04	4.3E+04
0.06	7.0E-03
0.08	1.9E+06
0.1	2.9E+00
0.15	1.3E+04
0.2	3.4E+04
0.3	1.5E+03
0.4	1.8E+03
0.5	3.5E+06
0.6	2.9E+03
0.8	2.6E+03
1.0	1.3E+03
1.5	4.6E+03
2.0	1.9E+04
3.0	1.3E+03

Table 12.2-8 Pressurizer Vapor Phase Activity

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	3.5E-02
Kr-85m	3.4E-01
Kr-85	4.5E+04
Kr-87	6.2E-02
Kr-88	4.0E-01
Xe-131m	5.0E+01
Xe-133m	9.3E+00
Xe-133	1.7E+03
Xe-135m	8.3E-03
Xe-135	3.9E+00
Xe-138	6.7E-03
I-130	4.5E-04
I-131	1.1E-02
I-132	6.1E-03
I-133	2.0E-02
I-134	4.2E-03
I-135	1.3E-02

Table 12.2-9 Isotopic Composition and Specific Activity of Typical Out-of-Core Corrosion Products in the primary coolant

Nuclide	Specific Activity (μCi/g)
Na-24	3.9E-02
Cr-51	3.8E-03
Mn-54	2.6E-03
Mn-56	1.3E-01
Fe-55	2.5E-03
Fe-59	4.4E-04
Co-58	6.1E-03
Co-60	8.9E-04
Zn-65	7.3E-04

Table 12.2-10 Chemical and Volume Control System Radiation Sources Regenerative Heat Exchanger Activity (Letdown Line)

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	4.3E-01	Te-129m	5.6E-03
Kr-85m	1.7E+00	Te-129	6.9E-03
Kr-85	8.8E+01	Te-131m	1.5E-02
Kr-87	1.1E+00	Te-131	8.0E-03
Kr-88	3.2E+00	Te-132	1.6E-01
Xe-131m	3.9E+00	Te-133m	1.5E-02
Xe-133m	4.0E+00	Te-134	2.8E-02
Xe-133	3.0E+02	I-130	5.9E-02
Xe-135m	7.3E-01	I-131	1.5E+00
Xe-135	9.8E+00	I-132	8.1E-01
Xe-138	6.4E-01	I-133	2.6E+00
N-16	7.9E+01	I-134	5.6E-01
Br-82	8.1E-03	I-135	1.7E+00
Br-83	7.4E-02	Cs-132	7.9E-04
Br-84	4.0E-02	Cs-134	7.2E-01
Rb-86	7.1E-03	Cs-135m	8.5E-03
Rb-88	4.0E+00	Cs-136	1.9E-01
Rb-89	9.3E-02	Cs-137	4.1E-01
Sr-89	1.8E-03	Cs-138	9.4E-01
Sr-90	1.2E-04	Ba-137m	3.9E-01
Sr-91	1.2E-03	Ba-140	2.2E-03
Sr-92	6.7E-04	La-140	5.7E-04
Y-90	2.6E-05	Ce-141	3.4E-04
Y-91m	6.2E-04	Ce-143	2.8E-04
Y-91	2.8E-04	Ce-144	2.5E-04
Y-92	5.2E-04	Pr-144	2.5E-04
Y-93	2.3E-04	Pm-147	2.8E-05
Zr-95	3.5E-04	Eu-154	2.6E-06
Nb-95	3.5E-04		
Mo-99	4.2E-01	Na-24	3.7E-02
Mo-101	1.9E-02	Cr-51	3.6E-03
Tc-99m	1.7E-01	Mn-54	2.4E-03
Ru-103	2.9E-04	Mn-56	1.2E-01
Ru-106	1.0E-04	Fe-55	2.4E-03
Ag-110m	9.2E-07	Fe-59	4.1E-04
Te-125m	4.1E-04	Co-58	5.7E-03
Te-127m	1.6E-03	Co-60	8.4E-04
		Zn-65	6.9E-04

Table 12.2-11 Chemical and Volume Control System Radiation Sources Regenerative Heat Exchanger Activity (Charging Line)

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	2.1E-01	Te-129m	1.2E-04
Kr-85m	1.1E+00	Te-129	1.5E-04
Kr-85	9.2E+01	Te-131m	3.1E-04
Kr-87	3.5E-01	Te-131	1.7E-04
Kr-88	1.6E+00	Te-132	3.4E-03
Xe-131m	4.1E+00	Te-133m	3.2E-04
Xe-133m	4.1E+00	Te-134	5.8E-04
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	9.4E-01	I-131	1.6E-02
Xe-135	9.6E+00	I-132	6.5E-02
Xe-138	7.4E-02	I-133	2.7E-02
		I-134	6.1E-03
Br-82	8.6E-05	I-135	1.8E-02
Br-83	7.8E-04	Cs-132	4.1E-04
Br-84	4.2E-04	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	4.5E-03
Rb-88	2.1E+00	Cs-136	1.0E-01
Rb-89	4.9E-02	Cs-137	2.2E-01
Sr-89	3.8E-05	Cs-138	4.9E-01
Sr-90	2.4E-06	Ba-137m	1.1E+03
Sr-91	2.5E-05	Ba-140	4.6E-05
Sr-92	1.4E-05	La-140	3.5E-04
Y-90	4.4E-04	Ce-141	7.1E-06
Y-91m	1.8E-04	Ce-143	6.0E-06
Y-91	6.1E-06	Ce-144	5.3E-06
Y-92	2.2E-05	Pr-144	1.0E-01
Y-93	4.8E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.9E-03	Na-24	7.7E-04
Mo-101	3.9E-04	Cr-51	7.5E-05
Tc-99m	8.7E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	2.6E-03
Ru-106	2.1E-06	Fe-55	5.0E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.7E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

Table 12.2-12 Chemical and Volume Control System Radiation Sources Regenerative Heat Exchanger Source Strengths

Gamma Ray Energy (MeV)	Letdown Line Source Strength ($MeV/cm^3/sec$)	Charging Line Source Strength ($MeV/cm^3/sec$)
0.015	9.2E+03	1.5E+04
0.02	2.2 E+01	5.0E+00
0.03	1.6 E+05	2.4E+05
0.04	7.8 E+03	3.2E+04
0.05	4.3 E+01	9.0E-01
0.06	5.4 E+01	2.8E+01
0.08	3.3 E+05	3.4E+05
0.1	6.2 E+01	1.6E+01
0.15	1.1E+04	7.0E+03
0.2	7.7 E+04	7.0E+04
0.3	8.9 E+03	2.9E+03
0.4	3.6 E+04	5.2E+03
0.5	7.7 E+04	2.6E+04
0.6	6.8 E+04	2.3E+07
0.8	1.3 E+05	3.4E+04
1.0	7.8 E+04	1.7E+04
1.5	1.5 E+05	4.2E+04
2.0	2.8 E+05	1.2E+05
3.0	1.2 E+05	1.9E+04
4.0	8.0 E+02	2.0E+02
5.0	1.1 E+03	6.0E+02
6.0	1.2 E+07	-
8.0	1.2 E+06	-

Table 12.2-13 Chemical and Volume Control System Radiation Sources Letdown Heat Exchanger Activity

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	4.5E-01	Te-129m	5.9E-03
Kr-85m	1.8E+00	Te-129	7.3E-03
Kr-85	9.2E+01	Te-131m	1.6E-02
Kr-87	1.2E+00	Te-131	8.5E-03
Kr-88	3.3E+00	Te-132	1.7E-01
Xe-131m	4.1E+00	Te-133m	1.6E-02
Xe-133m	4.2E+00	Te-134	2.9E-02
Xe-133	3.1E+02	I-130	6.2E-02
Xe-135m	7.7E-01	I-131	1.6E+00
Xe-135	1.0E+01	I-132	8.5E-01
Xe-138	6.7E-01	I-133	2.7E+00
N-16	7.3E+00	I-134	5.8E-01
Br-82	8.6E-03	I-135	1.8E+00
Br-83	7.8E-02	Cs-132	8.3E-04
Br-84	4.2E-02	Cs-134	7.6E-01
Rb-86	7.5E-03	Cs-135m	8.9E-03
Rb-88	4.3E+00	Cs-136	2.0E-01
Rb-89	9.8E-02	Cs-137	4.3E-01
Sr-89	1.9E-03	Cs-138	9.9E-01
Sr-90	1.2E-04	Ba-137m	4.1E-01
Sr-91	1.3E-03	Ba-140	2.3E-03
Sr-92	7.1E-04	La-140	6.0E-04
Y-90	2.8E-05	Ce-141	3.5E-04
Y-91m	6.6E-04	Ce-143	3.0E-04
Y-91	3.0E-04	Ce-144	2.7E-04
Y-92	5.5E-04	Pr-144	2.7E-04
Y-93	2.4E-04	Pm-147	3.0E-05
Zr-95	3.6E-04	Eu-154	2.8E-06
Nb-95	3.7E-04		
Mo-99	4.4E-01	Na-24	3.9E-02
Mo-101	2.0E-02	Cr-51	3.8E-03
Tc-99m	1.8E-01	Mn-54	2.6E-03
Ru-103	3.0E-04	Mn-56	1.3E-01
Ru-106	1.1E-04	Fe-55	2.5E-03
Ag-110m	9.7E-07	Fe-59	4.4E-04
Te-125m	4.3E-04	Co-58	6.0E-03
Te-127m	1.7E-03	Co-60	8.8E-04
		Zn-65	7.2E-04

Table 12.2-14 Chemical and Volume Control System Radiation Sources Letdown Heat Exchanger Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	9.7E+03
0.02	2.4E+01
0.03	1.7E+05
0.04	8.2E+03
0.05	4.5E+01
0.06	5.6E+01
0.08	3.5E+05
0.1	6.5E+01
0.15	1.1E+04
0.2	8.1E+04
0.3	9.4E+03
0.4	3.7E+04
0.5	8.1E+04
0.6	7.1E+04
0.8	1.4E+05
1.0	8.2E+04
1.5	1.6E+05
2.0	2.8E+05
3.0	5.7E+04
4.0	8.4E+02
5.0	1.2E+03
6.0	1.1E+06
8.0	1.1E+05

Table 12.2-15 Chemical and Volume Control System Radiation Sources Excess Letdown Heat Exchanger Activity

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	4.5E-01	Te-129m	5.9E-03
Kr-85m	1.8E+00	Te-129	7.3E-03
Kr-85	9.2E+01	Te-131m	1.6E-02
Kr-87	1.2E+00	Te-131	8.5E-03
Kr-88	3.3E+00	Te-132	1.7E-01
Xe-131m	4.1E+00	Te-133m	1.6E-02
Xe-133m	4.2E+00	Te-134	2.9E-02
Xe-133	3.1E+02	I-130	6.2E-02
Xe-135m	7.7E-01	I-131	1.6E+00
Xe-135	1.0E+01	I-132	8.5E-01
Xe-138	6.7E-01	I-133	2.7E+00
N-16	1.1E+02	I-134	5.8E-01
Br-82	8.6E-03	I-135	1.8E+00
Br-83	7.8E-02	Cs-132	8.3E-04
Br-84	4.2E-02	Cs-134	7.6E-01
Rb-86	7.5E-03	Cs-135m	8.9E-03
Rb-88	4.3E+00	Cs-136	2.0E-01
Rb-89	9.8E-02	Cs-137	4.3E-01
Sr-89	1.9E-03	Cs-138	9.9E-01
Sr-90	1.2E-04	Ba-137m	4.1E-01
Sr-91	1.3E-03	Ba-140	2.3E-03
Sr-92	7.1E-04	La-140	6.0E-04
Y-90	2.8E-05	Ce-141	3.5E-04
Y-91m	6.6E-04	Ce-143	3.0E-04
Y-91	3.0E-04	Ce-144	2.7E-04
Y-92	5.5E-04	Pr-144	2.7E-04
Y-93	2.4E-04	Pm-147	3.0E-05
Zr-95	3.6E-04	Eu-154	2.8E-06
Nb-95	3.7E-04		
Mo-99	4.4E-01	Na-24	3.9E-02
Mo-101	2.0E-02	Cr-51	3.8E-03
Tc-99m	1.8E-01	Mn-54	2.6E-03
Ru-103	3.0E-04	Mn-56	1.3E-01
Ru-106	1.1E-04	Fe-55	2.5E-03
Ag-110m	9.7E-07	Fe-59	4.4E-04
Te-125m	4.3E-04	Co-58	6.0E-03
Te-127m	1.7E-03	Co-60	8.8E-04
		Zn-65	7.2E-04

Table 12.2-16 Chemical and Volume Control System Radiation Sources Excess Letdown Heat Exchanger Source Strength

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	9.7E+03
0.02	2.4E+01
0.03	1.7E+05
0.04	8.2E+03
0.05	4.5E+01
0.06	5.6E+01
0.08	3.5E+05
0.1	6.5E+01
0.15	1.1E+04
0.2	8.1E+04
0.3	9.4E+03
0.4	3.7E+04
0.5	8.1E+04
0.6	7.1E+04
0.8	1.4E+05
1.0	8.2E+04
1.5	1.6E+05
2.0	2.9E+05
3.0	1.5E+05
4.0	8.4E+02
5.0	1.2E+03
6.0	1.7E+07
8.0	1.7E+06

Table 12.2-17 Chemical and Volume Control System Radiation Sources Mixed Bed Demineralizer Activity (70 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	8.9E+00	Te-129m	1.4E+02
Br-83	5.5E+00	Te-129	2.5E-01
Br-84	6.6E-01	Te-131m	1.4E+01
Rb-86	5.0E+01	Te-131	1.0E-01
Rb-88	1.9E+01	Te-132	3.9E+02
Rb-89	3.7E-01	Te-133m	4.3E-01
Sr-89	6.7E+01	Te-134	5.9E-01
Sr-90	4.2E+01	I-130	2.1E+04
Sr-91	3.5E-01	I-131	9.0E+03
Sr-92	5.6E-02	I-132	4.4E+02
Y-90	4.1E+01	I-133	1.7E+03
Y-91m	2.1E-01	I-134	1.6E+01
Y-91	1.3E+01	I-135	3.5E+02
Y-92	1.1E-01	Cs-132	1.3E+02
Y-93	7.1E-02	Cs-134	1.0E+05
Zr-95	1.6E+01	Cs-135m	1.2E-01
Nb-95	2.5E+01	Cs-136	9.4E+02
Mo-99	8.6E+02	Cs-137	7.7E+04
Mo-101	1.4E-01	Cs-138	7.9E+00
Tc-99m	7.7E+02	Ba-137m	7.1E+04
Ru-103	8.3E+00	Ba-140	2.1E+01
Ru-106	2.0E+01	La-140	2.1E+01
Ag-110m	1.5E-01	Ce-141	8.1E+00
Te-125m	1.8E+01	Ce-143	2.9E-01
Te-127m	1.3E+02	Ce-144	4.4E+01
		Pr-144	4.3E+01
		Pm-147	8.2E+00
		Eu-154	9.1E-01
		Na-24	1.7E+01
		Cr-51	7.3E+01
		Mn-54	4.5E+02
		Mn-56	9.7E+00
		Fe-55	6.9E+02
		Fe-59	1.4 E+01
		Co-58	3.0E+02
		Co-60	2.8E+02
		Zn-65	1.1E+02

Table 12.2-18 Chemical and Volume Control System Radiation Sources
Mixed Bed Demineralizer Source Strength (70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	7.5E+05
0.02	6.7E+04
0.03	7.1E+06
0.04	1.8E+06
0.05	1.0E+05
0.06	2.6E+05
0.08	8.8E+05
0.1	7.6E+04
0.15	4.8E+06
0.2	4.4E+06
0.3	1.4E+07
0.4	2.2E+08
0.5	4.6E+08
0.6	4.6E+09
0.8	3.4E+09
1.0	2.7E+08
1.5	2.2E+08
2.0	5.0E+06
3.0	2.2E+06
4.0	1.0E+04
5.0	5.3E+03

Table 12.2-19 Chemical and Volume Control System Radiation Sources Cation-Bed Demineralizer Activity (30 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Rb-86	1.0E+01	Te-129m	5.9E-01
Rb-88	3.9E+00	Te-129	1.1E-03
Rb-89	7.8E-02	Te-131m	5.8E-02
Sr-89	3.6E-01	Te-131	4.4E-04
Sr-90	1.8E-01	Te-132	1.7E+00
Sr-91	1.5E-03	Te-133m	1.9E-03
Sr-92	2.4E-04	Te-134	2.5E-03
Y-90	3.4E-01	Cs-132	2.8E+01
Y-91m	1.7E-03	Cs-134	2.1E+04
Y-91	5.5E-02	Cs-135m	2.5E-02
Y-92	6.9E-04	Cs-136	2.0E+02
Y-93	3.0E-04	Cs-137	1.6E+04
Zr-95	7.0E-02	Cs-138	1.7E+00
Nb-95	1.7E-01	Ba-137m	1.4E+04
Mo-99	3.7E+00	Ba-140	8.8E-02
Mo-101	6.0E-04	La-140	1.7E-01
Tc-99m	6.2E+00	Ce-141	3.5E-02
Ru-103	3.6E-02	Ce-143	1.2E-03
Ru-106	8.8E-02	Ce-144	1.9E-01
Ag-110m	6.3E-04	Pr-144	3.6E-01
Te-125m	7.5E-02	Pm-147	3.5E-02
Te-127m	5.6E-01	Eu-154	3.9E-03
		Na-24	7.2E-02
		Cr-51	3.1E-01
		Mn-54	1.9E+00
		Mn-56	4.1E-02
		Fe-55	2.9E+00
		Fe-59	5.8E-02
		Co-58	1.3E+00
		Co-60	1.2E+00
		Zn-65	4.6E-01

Table 12.2-20 Chemical and Volume Control System Radiation Sources Cation-Bed Demineralizer Source Strength (30 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	8.5E+04
0.02	4.5E+02
0.03	1.2E+06
0.04	3.6E+05
0.05	4.4E+02
0.06	5.5E+04
0.08	3.7E+04
0.1	3.3E+03
0.15	1.7E+05
0.2	2.5E+05
0.3	1.3E+06
0.4	2.5E+03
0.5	5.7E+06
0.6	8.5E+08
0.8	5.9E+08
1.0	2.9E+07
1.5	3.6E+07
2.0	8.8E+04
3.0	3.8E+04
4.0	4.1E+02
5.0	1.1E+03

Table 12.2-21 Chemical and Volume Control System Radiation Sources Reactor Coolant Filter

Reactor Coolant Filter activity	
Nuclide	Activity ($\mu Ci/cm^3$)
Co-60	1.2E+03

Reactor Coolant Filter source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	7.4E+01
0.3	1.0E+03
0.8	2.7E+03
1.0	4.5E+07
1.5	6.7E+07
2.0	9.8E+02
3.0	4.8E+00

Calculation Model	
Source Dimensions	Source region material
Outer radius : 6.4 in.	Water : 100 %
Inner radius : 5.2 in.	
Height : 27.3 in.	

Table 12.2-22 Chemical and Volume Control System Radiation Sources Volume Control Tank Activity (Liquid Phase)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	2.1E-01	Te-129m	1.2E-04
Kr-85m	1.1E+00	Te-129	1.5E-04
Kr-85	9.2E+01	Te-131m	3.1E-04
Kr-87	3.5E-01	Te-131	1.7E-04
Kr-88	1.6E+00	Te-132	3.4E-03
Xe-131m	4.1E+00	Te-133m	3.2E-04
Xe-133m	4.1E+00	Te-134	5.8E-04
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	9.4E-01	I-131	1.6E-02
Xe-135	9.6E+00	I-132	6.5E-02
Xe-138	7.4E-02	I-133	2.7E-02
		I-134	6.1E-03
Br-82	8.6E-05	I-135	1.8E-02
Br-83	7.8E-04	Cs-132	4.1E-04
Br-84	4.2E-04	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	4.5E-03
Rb-88	2.1E+00	Cs-136	1.0E-01
Rb-89	4.9E-02	Cs-137	2.2E-01
Sr-89	3.8E-05	Cs-138	4.9E-01
Sr-90	2.4E-06	Ba-137m	1.1E+03
Sr-91	2.5E-05	Ba-140	4.6E-05
Sr-92	1.4E-05	La-140	3.5E-04
Y-90	4.4E-04	Ce-141	7.1E-06
Y-91m	1.8E-04	Ce-143	6.0E-06
Y-91	6.1E-06	Ce-144	5.3E-06
Y-92	2.2E-05	Pr-144	1.0E-01
Y-93	4.8E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.9E-03	Na-24	7.7E-04
Mo-101	3.9E-04	Cr-51	7.5E-05
Tc-99m	8.7E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	2.6E-03
Ru-106	2.1E-06	Fe-55	5.0E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.7E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

Table 12.2-23 Chemical and Volume Control System Radiation Sources Volume Control Tank Source Strength (Liquid Phase)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.5E+04
0.02	5.0E+00
0.03	2.4E+05
0.04	3.2E+04
0.05	9.0E-01
0.06	2.8E+01
0.08	3.4E+05
0.1	1.6E+01
0.15	7.0E+03
0.2	7.0E+04
0.3	2.9E+03
0.4	5.2E+03
0.5	2.6E+04
0.6	2.3E+07
0.8	3.4E+04
1.0	1.7E+04
1.5	4.2E+04
2.0	1.2E+05
3.0	1.9E+04
4.0	2.0E+02
5.0	6.0E+02

Table 12.2-24 Chemical and Volume Control System Radiation Sources Volume Control Tank Activity (Vapor Phase)

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	4.9E+00
Kr-85m	2.5E+01
Kr-85	2.6E+02
Kr-87	8.0E+00
Kr-88	3.7E+01
Xe-131m	6.2E+01
Xe-133m	6.2E+01
Xe-133	4.6E+03
Xe-135m	1.4E+01
Xe-135	1.4E+02
Xe-138	1.1E+00

Table 12.2-25 Chemical and Volume Control System Radiation Sources
Volume Control Tank Source Strength (Vapor Phase)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.4E+05
0.03	2.4E+06
0.04	1.2E+05
0.06	1.1E+00
0.08	5.1E+06
0.1	2.7E+02
0.15	1.3E+05
0.2	1.0E+06
0.3	4.6E+04
0.4	9.4E+04
0.5	2.3E+05
0.6	9.8E+04
0.8	1.9E+05
1.0	9.2E+04
1.5	3.9E+05
2.0	1.7E+06
3.0	1.6E+05

Table 12.2-26 Chemical and Volume Control System Radiation Sources Holdup Tank Activity (Liquid Phase)

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	2.1E-01	Te-129m	1.2E-04
Kr-85m	1.1E+00	Te-129	3.9E-05
Kr-85	9.2E+01	Te-131m	2.9E-04
Kr-87	3.3E-01	Te-131	1.6E-05
Kr-88	1.7E+00	Te-132	3.3E-03
Xe-131m	4.1E+00	Te-133m	6.8E-05
Xe-133m	4.1E+00	Te-134	9.3E-05
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	4.6E-01	I-131	1.6E-02
Xe-135	9.2E+00	I-132	3.1E-02
Xe-138	3.6E-02	I-133	2.5E-02
		I-134	1.3E-03
Br-82	8.1E-05	I-135	1.3E-02
Br-83	3.6E-04	Cs-132	4.1E-04
Br-84	5.1E-05	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	9.0E-04
Rb-88	1.8E+00	Cs-136	1.0E-01
Rb-89	2.8E-03	Cs-137	2.2E-01
Sr-89	4.7E-05	Cs-138	9.7E-02
Sr-90	2.4E-06	Ba-137m	1.1E+01
Sr-91	2.0E-05	Ba-140	4.6E-05
Sr-92	7.0E-06	La-140	3.4E-04
Y-90	4.3E-04	Ce-141	7.1E-06
Y-91m	4.3E-05	Ce-143	5.6E-06
Y-91	6.2E-06	Ce-144	5.3E-06
Y-92	1.6E-05	Pr-144	6.8E-03
Y-93	3.9E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.6E-03	Na-24	6.7E-04
Mo-101	2.2E-05	Cr-51	7.5E-05
Tc-99m	6.4E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	1.2E-03
Ru-106	2.1E-06	Fe-55	4.9E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.6E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

Table 12.2-27 Chemical and Volume Control System Radiation Sources Holdup Tank Source Strength (Liquid Phase)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	9.4E+03
0.02	3.8E+00
0.03	1.7E+05
0.04	8.3E+03
0.05	8.8E-01
0.06	2.8E+01
0.08	3.4E+05
0.1	1.5E+01
0.15	7.1E+03
0.2	6.8E+04
0.3	2.8E+03
0.4	4.6E+03
0.5	1.6E+04
0.6	2.4E+05
0.8	3.1E+04
1.0	1.0E+04
1.5	2.4E+04
2.0	1.1E+05
3.0	1.3E+04
4.0	6.1E+01
5.0	5.1E+02

Table 12.2-28 Chemical and Volume Control System Radiation Sources Holdup Tank Activity (Vapor Phase)

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	1.8E-01
Kr-85m	9.8E-01
Kr-85	7.9E+01
Kr-87	2.8E-01
Kr-88	1.5E+00
Xe-131m	3.5E+00
Xe-133m	3.5E+00
Xe-133	2.7E+02
Xe-135m	3.9E-01
Xe-135	7.9E+00
Xe-138	3.1E-02

Table 12.2-29 Chemical and Volume Control System Radiation Sources Holdup Tank Source Strength (Vapor Phase)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	8.1E+03
0.03	1.4E+05
0.04	6.9E+03
0.06	3.3E-02
0.08	3.0E+05
0.1	1.1E+01
0.15	5.8E+03
0.2	5.8E+04
0.3	1.8E+03
0.4	3.7E+03
0.5	1.2E+04
0.6	5.5E+03
0.8	7.4E+03
1.0	3.6E+03
1.5	1.5E+04
2.0	6.8E+04
3.0	5.7E+03

Table 12.2-30 Volume Control Tank Noble Gas Stripping Fractions without assumption of purging

Nuclide	Stripping Fraction
Kr-83m	6.2E-01
Kr-85m	4.0E-01
Kr-85	3.2E-05
Kr-87	7.0E-01
Kr-88	5.1E-01
Xe-131m	6.8E-03
Xe-133m	3.6E-02
Xe-133	1.5E-02
Xe-135m	8.8E-01
Xe-135	1.7E-01
Xe-137	9.6E-01
Xe-138	8.9E-01

Table 12.2-31 Volume Control Tank Noble Gas Stripping Fraction with assumption of purging*

Nuclide	Stripping Fraction
Kr-83m	6.6E-01
Kr-85m	5.1E-01
Kr-85	2.6E-01
Kr-87	7.3E-01
Kr-88	5.8E-01
Xe-131m	1.9E-01
Xe-133m	2.1E-01
Xe-133	1.9E-01
Xe-135m	8.8E-01
Xe-135	3.0E-01
Xe-137	9.6E-01
Xe-138	8.9E-01

* Assuming purge rate of 5.3 gallons per minute (normal).

Table 12.2-32 Spent Fuel Pit Radiation Sources Spent Fuel Pit Water

Spent Fuel Pit Water activity	
Nuclide	Activity ($\mu Ci/cm^3$)
Co-60	5.5E-03

Spent Fuel Pit Water source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	3.4E-04
0.3	4.6E-03
0.8	1.2E-02
1.0	2.0E+02
1.5	3.0E+02
2.0	4.5E-03
3.0	2.2E-05

Table 12.2-33 Spent Fuel Pit Demineralizer Sources (70 ft^3 of Resin)

Spent Fuel Pit Demineralizer activity	
Nuclide	Activity (μCi/cm^3)
Co-60	1.3E+01

Spent Fuel Pit Demineralizer source strength	
Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	8.0E-01
0.3	1.1E+01
0.8	3.0E+01
1.0	4.9E+05
1.5	7.3E+05
2.0	1.1E+01
3.0	5.2E-02

Table 12.2-34 Spent Fuel Pit Filter Source Strengths

Spent Fuel Pit Filter activity	
Nuclide	Activity ($\mu Ci/cm^3$)
Co-60	2.4E+02

Spent Fuel Pit Filter source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.5E+01
0.3	2.0E+02
0.8	5.4E+02
1.0	8.9E+06
1.5	1.3E+07
2.0	2.0E+02
3.0	9.6E-01

Calculation Model	
Source Dimensions	Source region material
Outer radius : 6.4 in.	Water : 100 %
Inner radius : 5.2 in.	
Height : 27.3 in.	

Table 12.2-35 SG Blowdown Demineralizer Activity (350 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	7.1E-04	Te-134	1.1E-05
Br-83	2.3E-04	I-130	1.8E+00
Br-84	1.0E-05	I-131	7.6E-01
Rb-86	8.3E-03	I-132	3.5E-02
Rb-88	3.4E-04	I-133	1.3E-01
Rb-89	5.9E-06	I-134	3.6E-04
Sr-89	5.7E-03	I-135	2.2E-02
Sr-90	3.6E-03	Cs-132	2.3E-02
Sr-91	2.4E-05	Cs-134	1.7E+01
Sr-92	2.5E-06	Cs-135m	5.3E-06
Y-90	3.6E-03	Cs-136	1.6E-01
Y-91m	1.5E-05	Cs-137	1.3E+01
Y-91	1.1E-03	Cs-138	1.2E-04
Y-92	6.7E-06	Ba-137m	1.2E+01
Y-93	4.9E-06	Ba-140	1.7E-03
Zr-95	1.4E-03	La-140	1.8E-03
Nb-95	2.1E-03	Ce-141	6.8E-04
Mo-99	7.0E-02	Ce-143	2.3E-05
Mo-101	1.1E-06	Ce-144	3.8E-03
Tc-99m	6.5E-02	Pr-144	3.8E-03
Ru-103	7.0E-04	Pm-147	7.0E-04
Ru-106	1.7E-03	Eu-154	7.8E-05
Ag-110m	1.3E-05		
Te-125m	1.5E-03	Na-24	1.2E-03
Te-127m	1.1E-02	Cr-51	6.2E-03
Te-129m	1.2E-02	Mn-54	3.8E-02
Te-129	6.9E-06	Mn-56	4.2E-04
Te-131m	1.1E-03	Fe-55	5.8E-02
Te-131	1.3E-06	Fe-59	1.2E-03
Te-132	3.2E-02	Co-58	2.5E-02
Te-133m	1.0E-05	Co-60	2.3E-02
		Zn-65	9.2E-03

Table 12.2-36 SG Blowdown Demineralizer Source Strength (350 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.0E+02
0.02	5.6E+00
0.03	1.1E+03
0.04	3.1E+02
0.05	8.5E+00
0.06	4.4E+01
0.08	8.8E+01
0.1	7.5E+00
0.15	4.6E+02
0.2	4.6E+02
0.3	1.7E+03
0.4	1.8E+04
0.5	4.1E+04
0.6	7.4E+05
0.8	5.2E+05
1.0	3.4E+04
1.5	3.2E+04
2.0	3.1E+02
3.0	1.5E+02
4.0	2.5E-01
5.0	9.7E-02

Table 12.2-37 Liquid Waste Management System Radiation Sources Waste Holdup Tank Activity

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	3.5E-03	Te-129m	2.5E-03
Br-83	2.4E-02	Te-129	2.0E-03
Br-84	1.1E-02	Te-131m	6.3E-03
Rb-86	1.1E-02	Te-131	2.2E-03
Rb-88	1.4E+00	Te-132	7.0E-02
Rb-89	2.5E-02	Te-133m	4.3E-03
Sr-89	8.3E-04	Te-134	7.6E-03
Sr-90	5.4E-05	I-130	2.7E-02
Sr-91	4.7E-04	I-131	6.7E-01
Sr-92	2.2E-04	I-132	2.9E-01
Y-90	1.8E-04	I-133	1.1E+00
Y-91m	2.7E-04	I-134	1.5E-01
Y-91	1.3E-04	I-135	6.4E-01
Y-92	2.1E-04	Cs-132	2.2E-03
Y-93	9.0E-05	Cs-134	2.0E+00
Zr-95	1.6E-04	Cs-135m	2.4E-03
Nb-95	1.8E-04	Cs-136	2.5E-01
Mo-99	1.8E-01	Cs-137	1.2E+00
Mo-101	5.0E-03	Cs-138	2.6E-01
Tc-99m	1.1E-01	Ba-137m	8.0E+00
Ru-103	1.3E-04	Ba-140	9.8E-04
Ru-106	4.7E-05	La-140	4.2E-04
Ag-110m	4.3E-07	Ce-141	1.5E-04
Te-125m	1.9E-04	Ce-143	1.2E-04
Te-127m	7.5E-04	Ce-144	1.2E-04
		Pr-144	2.9E-03
		Pm-147	1.3E-05
		Eu-154	1.2E-06
		Na-24	1.5E-02
		Cr-51	1.6E-03
		Mn-54	1.1E-03
		Mn-56	4.0E-02
		Fe-55	1.1E-03
		Fe-59	1.9E-04
		Co-58	2.6E-03
		Co-60	3.9E-04
		Zn-65	3.2E-04

Table 12.2-38 Liquid Waste Management System Radiation Sources Waste Holdup Tank Source Strength

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	5.2E+01
0.02	1.1E+01
0.03	7.2E+02
0.04	1.9E+02
0.05	1.9E+01
0.06	6.8E+01
0.08	1.0E+02
0.1	1.8E+01
0.15	8.4E+02
0.2	1.1E+03
0.3	2.6E+03
0.4	9.4E+03
0.5	2.3E+04
0.6	2.3E+05
0.8	9.4E+04
1.0	3.1E+04
1.5	4.1E+04
2.0	3.4E+04
3.0	9.0E+03
4.0	2.2E+02
5.0	3.9E+02

Table 12.2-39 Liquid Waste Management System Radiation Sources Waste Demineralizer (Anion Bed: 70 ft^3 of Resin)

Waste Demineralizer Activity (Anion Bed)	
Nuclide	Activity ($\mu Ci/cm^3$)
I-130	7.6E+01
I-131	1.2E+03
I-132	9.7E+00
I-133	3.3E+02
I-134	2.0E+00
I-135	6.2E+01

Waste Demineralizer Source Strength (Anion Bed)	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	3.6E+03
0.03	7.0E+04
0.08	9.3E+04
0.1	1.9E+01
0.15	1.5E+03
0.2	3.7E+04
0.3	9.1E+05
0.4	1.5E+07
0.5	7.1E+06
0.6	4.1E+06
0.8	3.8E+06
1.0	1.7E+06
1.5	2.5E+06
2.0	5.6E+05
3.0	9.5E+02

Table 12.2-40 Liquid Waste Management System Radiation Sources
Waste Demineralizer Activity (Cation Bed: 70 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	1.6E+00	Te-129m	5.8E+00
Br-83	7.6E-01	Te-129	3.1E-02
Br-84	7.7E-02	Te-131m	2.5E+00
Rb-86	2.3E+01	Te-131	1.2E-02
Rb-88	5.4E+00	Te-132	6.8E+01
Rb-89	8.5E-02	Te-133m	5.3E-02
Sr-89	2.0E+00	Te-134	7.1E-02
Sr-90	1.4E-01	Cs-132	5.8E+00
Sr-91	6.0E-02	Cs-134	5.2E+03
Sr-92	8.0E-03	Cs-135m	2.8E-02
Y-90	2.4E-01	Cs-136	4.8E+02
Y-91m	3.4E-02	Cs-137	3.1E+03
Y-91	3.2E-01	Cs-138	1.8E+00
Y-92	1.7E-02	Ba-137m	2.6E+03
Y-93	1.2E-02	Ba-140	1.9E+00
Zr-95	3.9E-01	La-140	1.6E+00
Nb-95	4.4E-01	Ce-141	3.5E-01
Mo-99	1.5E+02	Ce-143	5.3E-02
Mo-101	1.6E-02	Ce-144	3.0E-01
Tc-99m	1.3E+02	Pr-144	2.8E-01
Ru-103	3.1E-01	Pm-147	3.4E-02
Ru-106	1.2E-01	Eu-154	3.2E-03
Ag-110m	1.1E-03		
Te-125m	4.6E-01	Na-24	3.0E+00
Te-127m	1.9E+00	Cr-51	3.7E+00
		Mn-54	2.9E+00
		Mn-56	1.4E+00
		Fe-55	2.8E+00
		Fe-59	4.5E-01
		Co-58	6.5E+00
		Co-60	1.0E+00
		Zn-65	8.1E-01

Table 12.2-41 Liquid Waste Management System Radiation Sources
Waste Demineralizer Source Strength (Cation Bed: 70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	2.8E+04
0.02	1.2E+04
0.03	3.5E+05
0.04	9.6E+04
0.05	1.8E+04
0.06	1.3E+05
0.08	8.9E+04
0.1	1.8E+04
0.15	1.0E+06
0.2	1.0E+06
0.3	3.2E+06
0.4	3.3E+04
0.5	1.6E+06
0.6	1.9E+08
0.8	1.6E+08
1.0	2.3E+07
1.5	9.3E+06
2.0	1.7E+05
3.0	3.8E+05
4.0	1.6E+03
5.0	1.5E+03

Table 12.2-42 Liquid Waste Management System Radiation Sources Waste Demineralizer Activity (Mixed Bed: 70 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	1.8E-01	Te-129m	6.4E-01
Br-83	8.3E-02	Te-129	3.5E-03
Br-84	8.4E-03	Te-131m	2.8E-01
Rb-86	1.3E+00	Te-131	1.3E-03
Rb-88	3.0E-01	Te-132	7.4E+00
Rb-89	4.7E-03	Te-133m	5.8E-03
Sr-89	2.2E-01	Te-134	7.8E-03
Sr-90	1.5E-02	I-130	7.6E-01
Sr-91	6.6E-03	I-131	1.2E+01
Sr-92	8.8E-04	I-132	7.4E+01
Y-90	4.0E-02	I-133	3.4E+00
Y-91m	7.5E-03	I-134	9.8E-02
Y-91	3.6E-02	I-135	6.2E-01
Y-92	2.8E-03	Cs-132	3.2E-01
Y-93	1.3E-03	Cs-134	2.9E+02
Zr-95	4.3E-02	Cs-135m	1.6E-03
Nb-95	5.7E-02	Cs-136	2.6E+01
Mo-99	1.7E+01	Cs-137	1.7E+02
Mo-101	1.8E-03	Cs-138	1.0E-01
Tc-99m	2.9E+01	Ba-137m	4.5E+02
Ru-103	3.4E-02	Ba-140	2.1E-01
Ru-106	1.3E-02	La-140	3.8E-01
Ag-110m	1.2E-04	Ce-141	3.9E-02
Te-125m	5.0E-02	Ce-143	5.9E-03
Te-127m	2.1E-01	Ce-144	3.3E-02
		Pr-144	6.4E-02
		Pm-147	3.7E-03
		Eu-154	3.5E-04
		Na-24	3.3E-01
		Cr-51	4.0E-01
		Mn-54	3.2E-01
		Mn-56	1.5E-01
		Fe-55	3.1E-01
		Fe-59	4.9E-02
		Co-58	7.1E-01
		Co-60	1.1E-01
		Zn-65	8.9E-02

Table 12.2-43 Liquid Waste Management System Radiation Sources
Waste Demineralizer Source Strength (Mixed Bed: 70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	3.5E+03
0.02	2.1E+03
0.03	4.5E+04
0.04	1.2E+04
0.05	2.0E+03
0.06	7.3E+03
0.08	5.9E+03
0.1	1.5E+03
0.15	1.7E+05
0.2	8.6E+04
0.3	2.1E+05
0.4	1.8E+05
0.5	4.9E+05
0.6	1.9E+07
0.8	1.1E+07
1.0	2.1E+06
1.5	1.2E+06
2.0	2.2E+05
3.0	4.4E+04
4.0	1.5E+02
5.0	8.5E+01

Table 12.2-44 Gaseous Waste Management System Radiation Sources Charcoal Bed Activity*

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	1.3E+00	Rb-88	1.5E+01
Kr-85m	1.4E+01	I-130	6.9E+00
Kr-85	7.6E+01	I-131	1.2E-02
Kr-87	1.5E+00	I-132	3.6E-03
Kr-88	1.5E+01	I-133	3.1E-02
Xe-131m	2.7E+02	I-134	1.8E-03
Xe-133m	1.8E+02	I-135	4.1E-04
Xe-133	1.7E+04	Cs-138	4.2E-02
Xe-135m	5.7E-01		
Xe-135	1.5E+02		
Xe-138	4.2E-02		

*Activities listed above are the mean values among 4 charcoal beds.

Table 12.2-45 Gaseous Waste Management System Radiation Sources Charcoal Bed Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	5.2E+05
0.03	9.3E+06
0.04	4.5E+05
0.06	4.4E-02
0.08	1.9E+07
0.1	1.1E+02
0.15	1.5E+05
0.2	1.2E+06
0.3	3.2E+04
0.4	7.3E+04
0.5	1.5E+05
0.6	2.5E+05
0.8	3.0E+05
1.0	6.9E+04
1.5	1.6E+05
2.0	9.0E+05
3.0	8.3E+04
4.0	3.7E+02
5.0	4.1E+03

Table 12.2-46 Gaseous Waste Management System Radiation Sources Waste Gas Surge Tank Activity

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	1.6E+00
Kr-85m	8.6E+00
Kr-85	3.6E+02
Kr-87	2.4E+00
Kr-88	1.3E+01
Xe-131m	2.6E+01
Xe-133m	2.6E+01
Xe-133	1.9E+03
Xe-135m	3.3E+00
Xe-135	5.7E+01
Xe-138	3.0E-02
Rb-88	1.3E+01
Cs-138	3.0E-02

Table 12.2-47 Gaseous Waste Management System Radiation Sources
Waste Gas Surge Tank Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	5.9E+04
0.03	1.0E+06
0.04	5.0E+04
0.06	3.2E-02
0.08	2.2E+06
0.1	9.4E+01
0.15	4.9E+04
0.2	4.3E+05
0.3	1.5E+04
0.4	3.0E+04
0.5	8.0E+04
0.6	4.0E+04
0.8	1.2E+05
1.0	3.1E+04
1.5	1.4E+05
2.0	7.9E+05
3.0	9.0E+04
4.0	3.3E+02
5.0	3.6E+03

Table 12.2-48 Solid Waste Management System Radiation Sources Spent Resin Storage Tank activity

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Br-82	2.6E+00	Te-129m	4.0E+01
Br-83	1.6E+00	Te-129	7.3E-02
Br-84	1.9E-01	Te-131m	4.0E+00
Rb-86	1.7E+01	Te-131	3.0E-02
Rb-88	6.3E+00	Te-132	1.1E+02
Rb-89	1.2E-01	Te-133m	1.3E-01
Sr-89	2.0E+01	Te-134	1.7E-01
Sr-90	1.2E+01	I-130	6.2E+03
Sr-91	1.0E-01	I-131	2.6E+03
Sr-92	1.6E-02	I-132	1.3E+02
Y-90	1.2E+01	I-133	4.9E+02
Y-91m	6.3E-02	I-134	4.6E+00
Y-91	3.7E+00	I-135	1.0E+02
Y-92	3.3E-02	Cs-132	4.3E+01
Y-93	2.1E-02	Cs-134	3.2E+04
Zr-95	4.8E+00	Cs-135m	3.8E-02
Nb-95	7.3E+00	Cs-136	3.2E+02
Mo-99	2.5E+02	Cs-137	2.4E+04
Mo-101	4.1E-02	Cs-138	2.5E+00
Tc-99m	2.3E+02	Ba-137m	2.3E+04
Ru-103	2.4E+00	Ba-140	6.0E+00
Ru-106	6.0E+00	La-140	6.1E+00
Ag-110m	4.3E-02	Ce-141	2.4E+00
Te-125m	5.1E+00	Ce-143	8.4E-02
Te-127m	3.8E+01	Ce-144	1.3E+01
		Pr-144	1.3E+01
		Pm-147	2.4E+00
		Eu-154	2.7E-01
		Na-24	5.0E+00
		Cr-51	2.1E+01
		Mn-54	1.3E+02
		Mn-56	2.8E+00
		Fe-55	2.0E+02
		Fe-59	4.0E+00
		Co-58	8.8E+01
		Co-60	8.0E+01
		Zn-65	3.2E+01

Table 12.2-49 Solid Waste Management System Radiation Sources
Spent Resin Storage Tank Source Strength

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	2.3E+05
0.02	2.0E+04
0.03	2.2E+06
0.04	5.8E+05
0.05	3.0E+04
0.06	8.9E+04
0.08	2.7E+05
0.1	2.3E+04
0.15	1.4E+06
0.2	1.4E+06
0.3	4.3E+06
0.4	6.3E+07
0.5	1.3E+08
0.6	1.5E+09
0.8	1.1E+09
1.0	8.3E+07
1.5	6.9E+07
2.0	1.5E+06
3.0	6.4E+05
4.0	3.1E+03
5.0	1.8E+03

Table 12.2-50 Miscellaneous Sources – Refueling Water Storage Auxiliary Tank

Refueling Water Storage Auxiliary Tank activity	
Nuclide	Activity ($\mu Ci/cm^3$)
Co-60	2.0E-04

Refueling Water Storage Auxiliary Tank source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.2E-05
0.3	1.7E-04
0.8	4.5E-04
1.0	7.5E+00
1.5	1.1E+01
2.0	1.6E-04
3.0	8.1E-07

Table 12.2-51 Miscellaneous Sources – Primary Makeup Water Tank

Primary Makeup Water Tank activity	
Nuclide	Activity ($\mu Ci/cm^3$)
Co-60	2.2E-04

Primary Makeup Water Tank source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.4E-05
0.3	1.9E-04
0.8	5.1E-04
1.0	8.3E+00
1.5	1.2E+01
2.0	1.8E-04
3.0	9.0E-07

Table 12.2-52 Residual Heat Removal System Activity - 4 Hours after Shutdown

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	9.8E-02	Te-129m	3.4E-03
Kr-85m	7.7E-01	Te-129	4.0E-04
Kr-85	9.2E+01	Te-131m	8.2E-03
Kr-87	8.9E-02	Te-131	6.4E-06
Kr-88	9.5E-01	Te-132	9.5E-02
Xe-131m	4.1E+00	Te-133m	4.7E-04
Xe-133m	3.9E+00	Te-134	3.2E-04
Xe-133	3.0E+02	I-130	3.6E-02
Xe-135m	1.5E-01	I-131	9.1E-01
Xe-135	7.6E+00	I-132	2.3E-01
Xe-138	3.3E-06	I-133	1.4E+00
		I-134	1.6E-02
Br-82	4.6E-03	I-135	6.8E-01
Br-83	1.4E-02	Cs-132	6.3E-04
Br-84	1.3E-04	Cs-134	5.8E-01
Rb-86	5.6E-03	Cs-135m	2.9E-04
Rb-88	1.1E+00	Cs-136	1.5E-01
Rb-89	1.3E-06	Cs-137	3.3E-01
Sr-89	1.1E-03	Cs-138	6.5E-03
Sr-90	7.1E-05	Ba-137m	1.4E+01
Sr-91	5.5E-04	Ba-140	1.3E-03
Sr-92	1.5E-04	La-140	5.6E-04
Y-90	2.1E-04	Ce-141	2.1E-04
Y-91m	3.7E-04	Ce-143	1.6E-04
Y-91	1.7E-04	Ce-144	1.6E-04
Y-92	2.8E-04	Pr-144	5.8E-03
Y-93	1.1E-04	Pm-147	1.7E-05
Zr-95	2.1E-04	Eu-154	1.6E-06
Nb-95	2.2E-04		
Mo-99	2.5E-01	Na-24	1.9E-02
Mo-101	1.3E-07	Cr-51	2.2E-03
Tc-99m	1.8E-01	Mn-54	1.5E-03
Ru-103	1.8E-04	Mn-56	2.6E-02
Ru-106	6.2E-05	Fe-55	1.4E-03
Ag-110m	5.7E-07	Fe-59	2.5E-04
Te-125m	2.5E-04	Co-58	3.5E-03
Te-127m	1.0E-03	Co-60	5.1E-04
		Zn-65	4.2E-04

Table 12.2-53 Residual Heat Removal System Source Strength - 4 Hours after Shutdown

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	9.3E+03
0.02	1.7E+01
0.03	1.6E+05
0.04	8.2E+03
0.05	2.5E+01
0.06	4.2E+01
0.08	3.4E+05
0.1	2.6E+01
0.15	6.1E+03
0.2	5.7E+04
0.3	3.5E+03
0.4	1.4E+04
0.5	3.5E+04
0.6	3.1E+05
0.8	4.6E+04
1.0	2.3E+04
1.5	3.7E+04
2.0	6.7E+04
3.0	7.8E+03
4.0	3.1E+01
5.0	3.0E+02

Table 12.2-54 Spent Fuel Gamma Ray Source Strengths at Various Times after Shutdown (Sheet 1 of 2)

Gamma Ray Energy (MeV)	Source Strength at Various Times after Shutdown (MeV/watt-sec)			
	12 hours	24 hours	100 hours	7 days
0.01	3.3E+08	2.8E+08	1.4E+08	8.5E+07
0.025	9.7E+07	8.1E+07	5.0E+07	3.9E+07
0.0375	1.6E+08	1.4E+08	8.8E+07	6.7E+07
0.0575	1.9E+08	1.6E+08	9.9E+07	7.6E+07
0.085	5.3E+08	4.6E+08	2.4E+08	1.5E+08
0.125	1.9E+09	1.7E+09	7.8E+08	4.4E+08
0.225	2.7E+09	2.3E+09	9.6E+08	5.2E+08
0.375	1.1E+09	9.9E+08	6.2E+08	4.6E+08
0.575	5.7E+09	4.7E+09	2.9E+09	2.4E+09
0.85	7.0E+09	6.1E+09	4.4E+09	3.8E+09
1.25	2.3E+09	1.6E+09	8.1E+08	6.0E+08
1.75	3.2E+09	2.9E+09	2.4E+09	2.1E+09
2.25	3.9E+08	3.1E+08	2.4E+08	2.0E+08
2.75	1.7E+08	1.6E+08	1.4E+08	1.2E+08
3.5	4.7E+06	1.8E+06	1.5E+06	1.3E+06
5.0	2.5E+05	1.3E+04	2.0E+01	2.0E+01
7.0	3.3E+00	3.3E+00	3.3E+00	3.3E+00
9.5	5.2E-01	5.2E-01	5.1E-01	5.1E-01

Table 12.2-54 Spent Fuel Gamma Ray Source Strengths at Various Times after Shutdown (Sheet 2 of 2)

Gamma Ray Energy (MeV)	Source Strength at Various Times after Shutdown (MeV/watt-sec)			
	30 days	100 days	180 days	365 days
0.01	2.9E+07	1.9E+07	1.5E+07	9.7E+06
0.025	1.9E+07	1.1E+07	8.3E+06	5.5E+06
0.0375	2.9E+07	1.6E+07	1.2E+07	8.3E+06
0.0575	3.4E+07	2.2E+07	1.7E+07	1.2E+07
0.085	3.8E+07	2.2E+07	1.8E+07	1.2E+07
0.125	1.1E+08	4.9E+07	3.2E+07	2.0E+07
0.225	8.5E+07	5.2E+07	4.1E+07	2.8E+07
0.375	1.3E+08	4.7E+07	3.7E+07	2.6E+07
0.575	1.4E+09	8.1E+08	6.3E+08	5.0E+08
0.85	2.6E+09	1.5E+09	8.8E+08	3.7E+08
1.25	2.2E+08	1.1E+08	9.4E+07	7.7E+07
1.75	6.2E+08	2.6E+07	9.7E+06	6.4E+06
2.25	7.9E+07	2.2E+07	1.7E+07	1.1E+07
2.75	3.5E+07	1.2E+06	4.0E+05	2.7E+05
3.5	4.5E+05	8.1E+04	6.2E+04	4.4E+04
5.0	2.0E+01	1.9E+01	1.8E+01	1.7E+01
7.0	3.2E+00	3.1E+00	2.9E+00	2.7E+00
9.5	5.0E-01	4.8E-01	4.6E-01	4.2E-01

Table 12.2-55 Irradiated Incore Detector and Drive Cable Maximum Withdrawal Source Strengths

Gamma Ray Energy (MeV)	Incore Detector (MeV/(cm^3-s))	Drive Cable (MeV/(cm-s))
0.015	1.4E+07	1.0E+07
0.06	-	7.0E+02
0.1	2.0E+05	1.1E+04
0.15	2.5E+05	3.6E+05
0.2	7.5E+05	1.4E+06
0.3	1.4E+07	7.2E+07
0.4	7.6E+08	2.9E+07
0.5	4.7E+08	1.8E+07
0.6	3.5E+07	1.3E+06
0.8	4.4E+11	4.4E+11
1.0	7.0E+09	1.2E+09
1.5	1.5E+10	7.8E+08
2.0	4.5E+11	4.7E+11
3.0	3.5E+10	3.6E+10

Table 12.2-56 Irradiated Incore Detector Drive Cable Source Strengths

Gamma Ray Energy (MeV)	Source Strength at Time after Shutdown (MeV/(cm-s))				
	10h	1 day	1 week	30 days	1 year
0.015	9.3E+06	9.1E+06	7.9E+06	4.8E+06	2.5E+05
0.06	3.4E-02	3.2E-08	-	-	-
0.1	7.6E+03	7.6E+03	7.5E+03	7.1E+03	3.0E+03
0.15	3.6E+05	3.5E+05	3.2E+05	2.3E+05	2.0E+03
0.2	1.4E+06	1.4E+06	1.3E+06	9.1E+05	5.6E+03
0.3	7.2E+07	7.1E+07	6.1E+07	3.4E+07	9.3E+03
0.4	2.0E+06	6.4E+04	1.7E+04	1.2E+04	7.5E+01
0.5	1.6E+07	1.6E+07	1.5E+07	1.2E+07	4.6E+05
0.6	9.0E+04	2.2E+03	9.2E+01	8.6E+01	3.6E+01
0.8	3.0E+10	8.9E+08	1.9E+08	1.7E+08	5.2E+07
1.0	2.3E+08	1.6E+08	1.5E+08	1.1E+08	2.3E+07
1.5	2.3E+08	1.9E+08	1.8E+08	1.4E+08	3.4E+07
2.0	3.2E+10	7.3E+08	1.5E+03	5.5E+02	4.9E+02
3.0	2.4E+09	5.6E+07	1.4E+01	2.7E+00	2.4E+00

Table 12.2-57 Irradiated Type 316 Stainless Steel (0.10 Weight Percent Cobalt) Flux Thimble Source Strengths

Gamma Ray Energy (MeV)	Source Strength at Time after Shutdown (MeV/(cm^3-s))				
	10h	1 day	1 week	30 days	1 year
0.015	2.8E+09	2.7E+09	2.4E+09	1.5E+09	6.3E+07
0.15	1.9E+07	1.9E+07	1.8E+07	1.2E+07	7.6E+04
0.2	7.8E+07	7.7E+07	7.0E+07	5.0E+07	3.1E+05
0.3	2.0E+10	2.0E+10	1.7E+10	9.5E+09	6.7E+06
0.4	1.0E+06	1.0E+06	9.4E+05	6.6E+05	4.1E+03
0.5	1.4E+10	1.4E+10	1.3E+10	1.0E+10	4.0E+08
0.8	1.8E+11	1.0E+11	9.3E+10	7.8E+10	1.3E+10
1.0	2.3E+11	2.3E+11	2.3E+11	2.2E+11	1.9E+11
1.5	3.4E+11	3.4E+11	3.4E+11	3.3E+11	2.9E+11
2.0	8.3E+10	1.9E+09	4.8E+06	4.8E+06	4.3E+06
3.0	6.4E+09	1.5E+08	2.4E+04	2.4E+04	2.1E+04

Table 12.2-58 Integrated Gamma Ray and Beta Source Strengths at Various Times Following a DBA (RG 1.183 Release Fractions) (Sheet 1 of 2)

Gamma Ray Energy (MeV)	Source Strength at Time after Release (MeV)				
	0.5 h	1 h	2 h	10 h	1 day
0.015	5.1E+19	9.2E+19	1.6E+20	6.0E+20	1.2E+21
0.02	9.7E+17	1.8E+18	3.5E+18	1.5E+19	3.0E+19
0.03	6.6E+19	1.2E+20	2.2E+20	8.1E+20	1.6E+21
0.04	4.4E+20	8.7E+20	1.7E+21	8.2E+21	1.9E+22
0.05	5.7E+18	1.1E+19	2.2E+19	1.0E+20	2.3E+20
0.06	1.9E+17	3.8E+17	7.5E+17	3.7E+18	8.3E+18
0.08	2.4E+19	4.1E+19	6.3E+19	1.6E+20	3.0E+20
0.1	7.6E+20	1.5E+21	3.0E+21	1.5E+22	3.5E+22
0.15	1.4E+20	2.5E+20	4.0E+20	8.9E+20	1.5E+21
0.2	1.2E+21	2.3E+21	4.0E+21	1.2E+22	1.6E+22
0.3	3.5E+21	6.1E+21	1.1E+22	4.4E+22	8.8E+22
0.4	2.2E+21	4.1E+21	7.4E+21	3.1E+22	6.8E+22
0.5	5.2E+21	8.7E+21	1.3E+22	2.2E+22	2.7E+22
0.6	8.7E+21	1.6E+22	2.9E+22	1.1E+23	2.1E+23
0.8	1.4E+22	2.7E+22	4.8E+22	1.5E+23	2.4E+23
1.0	2.2E+22	3.9E+22	6.2E+22	1.3E+23	1.7E+23
1.5	3.3E+22	6.1E+22	9.9E+22	2.4E+23	3.4E+23
2.0	1.6E+22	2.9E+22	4.9E+22	1.4E+23	1.8E+23
3.0	3.5E+22	6.0E+22	9.6E+22	2.2E+23	2.6E+23
4.0	9.9E+20	1.7E+21	2.6E+21	4.6E+21	5.0E+21
5.0	6.5E+19	1.4E+20	2.6E+20	7.6E+20	9.1E+20
Beta	6.2E+22	1.2E+23	2.0E+23	5.8E+23	8.8E+23

Table 12.2-58 Integrated Gamma Ray and Beta Source Strengths at Various Times Following a DBA (RG 1.183 Release Fractions) (Sheet 2 of 2)

Gamma Ray Energy (MeV)	Source Strength at Time after Release (MeV)				
	4 days	10 days	30 days	6 months	1 year
0.015	3.3E+21	5.8E+21	8.3E+21	9.8E+21	1.0E+22
0.02	8.1E+19	1.2E+20	1.3E+20	1.3E+20	1.4E+20
0.03	4.7E+21	7.6E+21	1.1E+22	1.4E+22	1.6E+22
0.04	6.5E+22	1.2E+23	1.7E+23	1.9E+23	2.0E+23
0.05	7.0E+20	1.1E+21	1.3E+21	1.4E+21	1.5E+21
0.06	2.7E+19	4.7E+19	7.4E+19	1.1E+20	1.2E+20
0.08	9.5E+20	1.8E+21	2.9E+21	3.8E+21	3.8E+21
0.1	1.2E+23	2.3E+23	3.2E+23	3.5E+23	3.5E+23
0.15	3.4E+21	4.8E+21	6.1E+21	8.1E+21	8.9E+21
0.2	1.9E+22	2.3E+22	2.9E+22	3.6E+22	3.7E+22
0.3	1.5E+23	1.8E+23	2.1E+23	2.2E+23	2.2E+23
0.4	2.3E+23	4.6E+23	7.7E+23	9.4E+23	9.5E+23
0.5	4.2E+22	7.0E+22	1.3E+23	2.5E+23	3.1E+23
0.6	3.8E+23	4.4E+23	5.6E+23	1.2E+24	2.0E+24
0.8	6.0E+23	1.1E+24	2.4E+24	1.1E+25	2.1E+25
1.0	2.5E+23	3.6E+23	5.7E+23	1.2E+24	1.7E+24
1.5	4.6E+23	5.5E+23	7.3E+23	1.3E+24	1.8E+24
2.0	2.5E+23	3.9E+23	6.9E+23	1.0E+24	1.0E+24
3.0	2.7E+23	2.8E+23	3.0E+23	3.3E+23	3.3E+23
4.0	5.1E+21	5.1E+21	5.3E+21	5.5E+21	5.5E+21
5.0	9.1E+20	9.1E+20	9.1E+20	9.1E+20	9.1E+20
Beta	1.6E+24	2.3E+24	3.3E+24	5.7E+24	7.8 E+24

Table 12.2-59 Source Strength in the RHR Loop at Various Times Following an Equivalent Full-Core Meltdown Accident (Sheet 1 of 2)

Gamma Ray Energy (MeV)	Source Strength at Time after Release (MeV/g/s)					
	0 h	0.5 h	1 h	2 h	10 h	1 day
0.015	1.8E+06	1.6E+06	1.5E+06	1.5E+06	1.1E+06	9.8E+05
0.02	2.7E+05	2.3E+05	2.2E+05	2.1E+05	1.9E+05	1.6E+05
0.03	1.8E+07	1.3E+07	1.2E+07	1.0E+07	7.4E+06	6.2E+06
0.04	8.4E+06	6.7E+06	6.0E+06	5.6E+06	7.1E+06	8.2E+06
0.05	2.0E+06	1.9E+06	1.8E+06	1.7E+06	1.6E+06	1.4E+06
0.06	6.4E+04	6.4E+04	6.4E+04	6.3E+04	5.8E+04	5.1E+04
0.08	8.9E+06	5.0E+06	3.2E+06	1.6E+06	6.8E+05	6.7E+05
0.1	6.8E+06	5.5E+06	5.5E+06	5.7E+06	8.7E+06	1.2E+07
0.15	5.0E+07	3.7E+07	2.7E+07	1.5E+07	6.9E+06	5.8E+06
0.2	5.5E+07	3.5E+07	2.5E+07	1.6E+07	7.3E+06	6.2E+06
0.3	2.8E+08	2.4E+08	2.3E+08	2.4E+08	3.2E+08	2.3E+08
0.4	5.9E+08	5.1E+08	4.9E+08	4.7E+08	4.4E+08	4.1E+08
0.5	8.5E+08	5.4E+08	3.6E+08	2.0E+08	6.0E+07	3.9E+07
0.6	2.5E+09	2.4E+09	2.2E+09	2.0E+09	1.3E+09	7.7E+08
0.8	5.0E+09	4.3E+09	3.7E+09	2.9E+09	1.2E+09	9.5E+08
1.0	7.4E+09	5.2E+09	3.8E+09	2.2E+09	4.1E+08	2.6E+08
1.5	1.1E+10	7.3E+09	5.5E+09	3.8E+09	1.4E+09	5.2E+08
2.0	3.0E+09	2.2E+09	1.8E+09	1.3E+09	5.0E+08	1.9E+08
3.0	3.1E+09	1.5E+09	8.6E+08	4.0E+08	9.3E+07	3.4E+07
4.0	3.4E+08	1.6E+08	7.7E+07	2.1E+07	8.9E+04	5.0E+04
5.0	1.9E+07	7.0E+06	2.9E+06	6.0E+05	4.7E+02	8.3E-01

Table 12.2-59 Source Strength in the RHR Loop at Various Times Following an Equivalent Full-Core Meltdown Accident (Sheet 2 of 2)

Gamma Ray Energy (MeV)	Source Strength at Time after Release (MeV/g/s)				
	4 days	10 days	30 days	6 months	1 year
0.015	7.4E+05	4.1E+05	1.1E+05	2.3E+04	2.1E+04
0.02	7.0E+04	1.5E+04	7.0E+02	6.5E+01	4.0E+01
0.03	3.7E+06	1.7E+06	3.9E+05	5.3E+04	4.0E+04
0.04	7.6E+06	4.1E+06	8.8E+05	4.6E+05	4.4E+05
0.05	7.3E+05	2.1E+05	9.9E+03	2.1E+03	1.3E+03
0.06	3.2E+04	1.7E+04	4.0E+03	4.6E+02	1.5E+02
0.08	5.8E+05	4.0E+05	1.3E+05	1.1E+02	4.5E+01
0.1	1.4E+07	7.1E+06	7.4E+05	3.6E+03	2.3E+03
0.15	2.8E+06	8.6E+05	2.0E+05	2.9E+04	1.6E+04
0.2	4.7E+06	3.3E+06	1.1E+06	3.2E+04	2.8E+04
0.3	4.9E+07	2.0E+07	2.5E+06	1.9E+04	1.5E+04
0.4	3.2E+08	2.0E+08	3.7E+07	3.4E+04	2.7E+04
0.5	3.4E+07	2.9E+07	1.3E+07	2.2E+06	1.6E+06
0.6	1.2E+08	4.9E+07	3.6E+07	2.8E+07	2.4E+07
0.8	6.9E+08	5.2E+08	4.3E+08	3.8E+08	3.3E+08
1.0	1.5E+08	1.0E+08	4.8E+07	2.0E+07	1.6E+07
1.5	1.4E+08	8.7E+07	4.1E+07	1.8E+07	1.5E+07
2.0	1.7E+08	1.5E+08	5.3E+07	8.8E+05	5.0E+05
3.0	1.8E+07	1.2E+07	3.8E+06	5.0E+04	2.9E+04
4.0	1.0E+05	9.5E+04	3.3E+04	3.0E+02	2.0E+02
5.0	-	-	-	-	-

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Containment) (Sheet 1 of 3)

Parameter/ Assumption	Value
Reactor coolant leakage rate in normal operation	100 lb/d
Reactor coolant evaporation rate in refueling	1020 lb/h
Fraction of radioactive material to free volume	(in normal operation) 1.0 (for noble gas) 0.45 (others) (in refueling/shutdown) 1.0 (for noble gas & tritium) 0.1 (iodine) 0.001 (others)
Fuel defect	1%
Reactor coolant specific activity in normal operation (except tritium)	Table 11.1-2
Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)	Table 12.2-72
Reactor coolant tritium specific activity	3.5 μCi/g
Reactor cavity and SFP water tritium specific activity in refueling /shutdown	0.35 μCi/g
Low volume purge flow rate	(in normal operation) 2000 cfm
High volume purge flow rate	(in refueling) 30000 cfm
Purge flow duration	continuous*

* Assumed as a constant removal term for the calculation of the equilibrium airborne radioactive concentration, but the normally closed low volume purge valve is only operated intermittently, as needed, to ensure acceptable personnel containment entry conditions.

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Fuel Handling Area) (Sheet 2 of 3)

Parameter/ Assumption	Value
Reactor coolant evaporation rate in refueling	750 lb/h
Fraction of radioactive material to free volume	(in refueling/shutdown) 1.0 (for noble gas & tritium) 0.1 (iodine) 0.001 (others)
Fuel defect	1%
Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium)	Table 12.2-72
Reactor cavity and SFP water tritium specific activity in refueling /shutdown	0.35 μCi/g
Exhaust flow rate from the fuel handling area	24000 cfm
Flow duration	continuous

Table 12.2-60 Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building) (Sheet 3 of 3)

Parameter/ Assumption	Value
Reactor coolant leak rate in normal operation/refueling (Note)	100 lb/d (for Radiation Zone V or higher) 50 lb/d (for Radiation Zone IV) 2 lb/d (for Radiation Zone III)
Fraction of radioactive material to free volume	(in normal operation) 1.0 (for noble gas) 0.1 (iodine&tritium) 0.001 (others)
Fuel defect	1%
Reactor coolant specific activity in normal operation (except tritium)	Table 11.1-2
Reactor coolant tritium specific activity	3.5 μCi/g
Minimum flow rate	1500 cfm (for Radiation Zone V or higher) 14000 cfm (for Radiation Zone IV) 76000 cfm (for Radiation Zone III)
Flow duration	continuous

(Note) Reactor coolant leak rates were derived from the leakage flow rates of the valves under consideration. Each Radiation Zone has a different number of valves handling radioactive fluids. Radiation Zones V and higher have many component cubicles and valve galleries. These zones have many radioactive valves. Zone IV has relatively high radiation level corridors, but has fewer radioactive valves than Zone V. Zone III has low radiation level corridors and access areas, and has fewer radioactive valves than Zone IV. As a result, the leak rate in Zone V or higher is high, while in Zones IV and III, the leak rates is low.

Table 12.2-61 Airborne Radioactive Concentrations (Containment in normal operation) (Sheet 1 of 6)

Nuclide	Activity (μCi/cm^3)	10 CFR 20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10 CFR 20 DAC (μCi/cm^3)
Kr-83m	2.5E-07	1E-02	Ru-106	2.7E-11	5E-09
Kr-85m	1.0E-06	2E-05	Ag-110m	2.5E-13	4E-08
Kr-85	5.2E-05	1E-04	Te-125m	1.1E-10	2E-07
Kr-87	6.5E-07	5E-06	Te-127m	4.3E-10	1E-07
Kr-88	1.9E-06	2E-06	Te-127	2.3E-09	7E-06
Xe-131m	2.3E-06	4E-04	Sb-129	7.4E-12	4E-06
Xe-133m	2.4E-06	1E-04	Te-129m	1.5E-09	1E-07
Xe-133	1.8E-04	1E-04	Te-129	1.8E-09	3E-05
Xe-135m	4.3E-07	9E-06	Sb-131	3.0E-12	1E-05
Xe-135	5.8E-06	1E-05	Te-131m	3.9E-09	2E-07
Xe-138	3.8E-07	4E-06	Te-131	2.1E-09	2E-06
I-130	1.6E-08	3E-07	Te-132	4.3E-08	9E-08
I-131	4.0E-07	2E-08	Cs-132	2.1E-10	2E-06
I-132	2.2E-07	3E-06	Te-133m	4.1E-09	2E-06
I-133	6.9E-07	1E-07	Te-133	2.0E-09	9E-06
I-134	1.5E-07	2E-05	Cs-134	1.9E-07	4E-08
I-135	4.5E-07	7E-07	Te-134	7.4E-09	1E-05
Br-82	2.2E-09	2E-06	Cs-135m	2.3E-09	8E-05
Br-83	2.0E-08	3E-05	Cs-135	5.2E-13	5E-07
Br-84	1.1E-08	2E-05	Cs-136	5.1E-08	3E-07
Rb-86	1.9E-09	3E-07	Cs-137	1.1E-07	6E-08
Rb-87	-	6E-07	Cs-138	2.5E-07	2E-05
Rb-88	1.1E-06	3E-05	Ba-140	5.8E-10	6E-07
Rb-89	2.5E-08	6E-05	La-140	1.5E-10	5E-07
Sr-89	4.8E-10	6E-08	La-141	3.9E-11	4E-06
Sr-90	3.1E-11	2E-09	Ce-141	8.9E-11	2E-07
Y-90	7.0E-12	3E-07	Ce-143	7.6E-11	7E-07
Sr-91	3.2E-10	1E-06	Pr-143	8.2E-11	3E-07
Y-91m	1.7E-10	7E-05	Ce-144	6.8E-11	6E-09
Y-91	7.5E-11	5E-08	Pr-144	6.8E-11	5E-05
Sr-92	1.8E-10	3E-06	Pm-147	7.5E-12	5E-08
Y-92	1.4E-10	3E-06	Sm-147	-	2E-11
Y-93	6.1E-11	1E-06	Eu-154	7.0E-13	8E-09
Zr-93	1.2E-16	3E-09	Na-24	9.8E-09	2E-06
Zr-95	9.2E-11	5E-08	Cr-51	9.5E-10	8E-06
Nb-95m	6.6E-13	9E-07	Mn-54	6.5E-10	3E-07
Nb-95	9.3E-11	5E-07	Mn-56	3.3E-08	6E-06
Mo-99	1.1E-07	6E-07	Fe-55	6.3E-10	8E-07
Tc-99m	4.5E-08	6E-05	Fe-59	1.1E-10	1E-07
Tc-99	1.2E-15	3E-07	Co-58	1.5E-09	3E-07
Mo-101	4.9E-09	6E-05	Co-60	2.2E-10	1E-08
Tc-101	4.7E-09	1E-04	Zn-65	1.8E-10	1E-07
Ru-103	7.6E-11	3E-07	H-3	8.8E-07	2E-05
Rh-103m	7.5E-11	5E-04			

(Note) There is only limited access to the reactor containment during power operation.

Table 12.2-61 Airborne Radioactive Concentrations (Containment in refueling/shutdown) (Sheet 2 of 6)

Nuclide	Activity ($\mu Ci/cm^3$)	10 CFR 20 DAC ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)	10 CFR 20 DAC ($\mu Ci/cm^3$)
Kr-83m	1.8E-11	1E-02	Ru-106	-	5E-09
Kr-85m	3.6E-10	2E-05	Ag-110m	-	4E-08
Kr-85	4.9E-06	1E-04	Te-125m	-	2E-07
Kr-87	4.0E-15	5E-06	Te-127m	-	1E-07
Kr-88	4.8E-11	2E-06	Te-127	9.1E-13	7E-06
Xe-131m	3.4E-07	4E-04	Sb-129	-	4E-06
Xe-133m	2.4E-07	1E-04	Te-129m	1.5E-15	1E-07
Xe-133	2.4E-05	1E-04	Te-129	-	3E-05
Xe-135m	2.1E-09	9E-06	Sb-131	-	1E-05
Xe-135	1.4E-07	1E-05	Te-131m	2.3E-15	2E-07
Xe-138	-	4E-06	Te-131	-	2E-06
I-130	1.5E-12	3E-07	Te-132	3.5E-14	9E-08
I-131	3.6E-11	2E-08	Cs-132	3.0E-15	2E-06
I-132	4.9E-10	3E-06	Te-133m	-	2E-06
I-133	3.0E-11	1E-07	Te-133	-	9E-06
I-134	-	2E-05	Cs-134	2.7E-12	4E-08
I-135	3.5E-12	7E-07	Te-134	-	1E-05
Br-82	1.3E-15	2E-06	Cs-135m	-	8E-05
Br-83	-	3E-05	Cs-135	-	5E-07
Br-84	-	2E-05	Cs-136	6.9E-13	3E-07
Rb-86	2.6E-14	3E-07	Cs-137	1.6E-12	6E-08
Rb-87	-	6E-07	Cs-138	-	2E-05
Rb-88	5.3E-14	3E-05	Ba-140	-	6E-07
Rb-89	-	6E-05	La-140	4.0E-14	5E-07
Sr-89	-	6E-08	La-141	-	4E-06
Sr-90	-	2E-09	Ce-141	-	2E-07
Y-90	5.4E-14	3E-07	Ce-143	-	7E-07
Sr-91	-	1E-06	Pr-143	1.2E-15	3E-07
Y-91m	1.6E-15	7E-05	Ce-144	-	6E-09
Y-91	-	5E-08	Pr-144	1.4E-12	5E-05
Sr-92	-	3E-06	Pm-147	-	5E-08
Y-92	-	3E-06	Sm-147	-	2E-11
Y-93	-	1E-06	Eu-154	-	8E-09
Zr-93	-	3E-09	Na-24	3.2E-15	2E-06
Zr-95	-	5E-08	Cr-51	-	8E-06
Nb-95m	-	9E-07	Mn-54	-	3E-07
Nb-95	1.7E-15	5E-07	Mn-56	-	6E-06
Mo-99	8.7E-14	6E-07	Fe-55	-	8E-07
Tc-99m	6.6E-12	6E-05	Fe-59	-	1E-07
Tc-99	-	3E-07	Co-58	1.5E-15	3E-07
Mo-101	-	6E-05	Co-60	-	1E-08
Tc-101	-	1E-04	Zn-65	-	1E-07
Ru-103	-	3E-07	H-3	3.2E-06	2E-05
Rh-103m	2.1E-13	5E-04			

Table 12.2-61 Airborne Radioactive Concentrations (Fuel Handling Area) (Sheet 3 of 6)

Nuclide	Activity (μCi/cm^3)	10 CFR 20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10 CFR 20 DAC (μCi/cm^3)
Kr-83m	1.6E-11	1E-02	Ru-106	-	5E-09
Kr-85m	3.3E-10	2E-05	Ag-110m	-	4E-08
Kr-85	4.5E-06	1E-04	Te-125m	-	2E-07
Kr-87	3.7E-15	5E-06	Te-127m	-	1E-07
Kr-88	4.4E-11	2E-06	Te-127	8.4E-13	7E-06
Xe-131m	3.2E-07	4E-04	Sb-129	-	4E-06
Xe-133m	2.2E-07	1E-04	Te-129m	1.3E-15	1E-07
Xe-133	2.2E-05	1E-04	Te-129	-	3E-05
Xe-135m	1.9E-09	9E-06	Sb-131	-	1E-05
Xe-135	1.3E-07	1E-05	Te-131m	2.1E-15	2E-07
Xe-138	-	4E-06	Te-131	-	2E-06
I-130	1.4E-12	3E-07	Te-132	3.2E-14	9E-08
I-131	3.3E-11	2E-08	Cs-132	2.8E-15	2E-06
I-132	4.5E-10	3E-06	Te-133m	-	2E-06
I-133	2.8E-11	1E-07	Te-133	-	9E-06
I-134	-	2E-05	Cs-134	2.5E-12	4E-08
I-135	3.3E-12	7E-07	Te-134	-	1E-05
Br-82	1.2E-15	2E-06	Cs-135m	-	8E-05
Br-83	-	3E-05	Cs-135	-	5E-07
Br-84	-	2E-05	Cs-136	6.3E-13	3E-07
Rb-86	2.4E-14	3E-07	Cs-137	1.4E-12	6E-08
Rb-87	-	6E-07	Cs-138	-	2E-05
Rb-88	4.9E-14	3E-05	Ba-140	-	6E-07
Rb-89	-	6E-05	La-140	3.6E-14	5E-07
Sr-89	-	6E-08	La-141	-	4E-06
Sr-90	-	2E-09	Ce-141	-	2E-07
Y-90	4.9E-14	3E-07	Ce-143	-	7E-07
Sr-91	-	1E-06	Pr-143	1.1E-15	3E-07
Y-91m	1.5E-15	7E-05	Ce-144	-	6E-09
Y-91	-	5E-08	Pr-144	1.3E-12	5E-05
Sr-92	-	3E-06	Pm-147	-	5E-08
Y-92	-	3E-06	Sm-147	-	2E-11
Y-93	-	1E-06	Eu-154	-	8E-09
Zr-93	-	3E-09	Na-24	3.0E-15	2E-06
Zr-95	-	5E-08	Cr-51	-	8E-06
Nb-95m	-	9E-07	Mn-54	-	3E-07
Nb-95	1.6E-15	5E-07	Mn-56	-	6E-06
Mo-99	8.0E-14	6E-07	Fe-55	-	8E-07
Tc-99m	6.0E-12	6E-05	Fe-59	-	1E-07
Tc-99	-	3E-07	Co-58	1.4E-15	3E-07
Mo-101	-	6E-05	Co-60	-	1E-08
Tc-101	-	1E-04	Zn-65	-	1E-07
Ru-103	-	3E-07	H-3	2.9E-06	2E-05
Rh-103m	2.0E-13	5E-04			

Table 12.2-61 Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone V or higher) (Sheet 4 of 6)

Nuclide	Activity (μCi/cm^3)	10 CFR 20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10 CFR 20 DAC (μCi/cm^3)
Kr-83m	3.4E-07	1E-02	Ru-106	7.9E-14	5E-09
Kr-85m	1.3E-06	2E-05	Ag-110m	7.3E-16	4E-08
Kr-85	6.9E-05	1E-04	Te-125m	3.2E-13	2E-07
Kr-87	8.7E-07	5E-06	Te-127m	1.3E-12	1E-07
Kr-88	2.5E-06	2E-06	Te-127	6.8E-12	7E-06
Xe-131m	3.1E-06	4E-04	Sb-129	2.2E-14	4E-06
Xe-133m	3.1E-06	1E-04	Te-129m	4.4E-12	1E-07
Xe-133	2.3E-04	1E-04	Te-129	5.5E-12	3E-05
Xe-135m	5.7E-07	9E-06	Sb-131	8.9E-15	1E-05
Xe-135	7.7E-06	1E-05	Te-131m	1.2E-11	2E-07
Xe-138	5.0E-07	4E-06	Te-131	6.3E-12	2E-06
I-130	4.7E-09	3E-07	Te-132	1.3E-10	9E-08
I-131	1.2E-07	2E-08	Cs-132	6.2E-13	2E-06
I-132	6.4E-08	3E-06	Te-133m	1.2E-11	2E-06
I-133	2.1E-07	1E-07	Te-133	6.0E-12	9E-06
I-134	4.4E-08	2E-05	Cs-134	5.7E-10	4E-08
I-135	1.3E-07	7E-07	Te-134	2.2E-11	1E-05
Br-82	6.4E-12	2E-06	Cs-135m	6.7E-12	8E-05
Br-83	5.8E-11	3E-05	Cs-135	1.5E-15	5E-07
Br-84	3.1E-11	2E-05	Cs-136	1.5E-10	3E-07
Rb-86	5.6E-12	3E-07	Cs-137	3.2E-10	6E-08
Rb-87	-	6E-07	Cs-138	7.4E-10	2E-05
Rb-88	3.2E-09	3E-05	Ba-140	1.7E-12	6E-07
Rb-89	7.3E-11	6E-05	La-140	4.5E-13	5E-07
Sr-89	1.4E-12	6E-08	La-141	1.2E-13	4E-06
Sr-90	9.2E-14	2E-09	Ce-141	2.6E-13	2E-07
Y-90	2.1E-14	3E-07	Ce-143	2.2E-13	7E-07
Sr-91	9.5E-13	1E-06	Pr-143	2.4E-13	3E-07
Y-91m	4.9E-13	7E-05	Ce-144	2.0E-13	6E-09
Y-91	2.2E-13	5E-08	Pr-144	2.0E-13	5E-05
Sr-92	5.3E-13	3E-06	Pm-147	2.2E-14	5E-08
Y-92	4.1E-13	3E-06	Sm-147	-	2E-11
Y-93	1.8E-13	1E-06	Eu-154	2.1E-15	8E-09
Zr-93	-	3E-09	Na-24	2.9E-11	2E-06
Zr-95	2.7E-13	5E-08	Cr-51	2.8E-12	8E-06
Nb-95m	2.0E-15	9E-07	Mn-54	1.9E-12	3E-07
Nb-95	2.7E-13	5E-07	Mn-56	9.6E-11	6E-06
Mo-99	3.3E-10	6E-07	Fe-55	1.9E-12	8E-07
Tc-99m	1.3E-10	6E-05	Fe-59	3.3E-13	1E-07
Tc-99	-	3E-07	Co-58	4.5E-12	3E-07
Mo-101	1.5E-11	6E-05	Co-60	6.6E-13	1E-08
Tc-101	1.4E-11	1E-04	Zn-65	5.4E-13	1E-07
Ru-103	2.3E-13	3E-07	H-3	2.6E-07	2E-05
Rh-103m	2.2E-13	5E-04			

Table 12.2-61 Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone IV) (Sheet 5 of 6)

Nuclide	Activity ($\mu Ci/cm^3$)	10 CFR 20 DAC ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)	10 CFR 20 DAC ($\mu Ci/cm^3$)
Kr-83m	1.8E-08	1E-02	Ru-106	4.3E-15	5E-09
Kr-85m	7.2E-08	2E-05	Ag-110m	-	4E-08
Kr-85	3.7E-06	1E-04	Te-125m	1.7E-14	2E-07
Kr-87	4.6E-08	5E-06	Te-127m	6.9E-14	1E-07
Kr-88	1.3E-07	2E-06	Te-127	3.6E-13	7E-06
Xe-131m	1.7E-07	4E-04	Sb-129	1.2E-15	4E-06
Xe-133m	1.7E-07	1E-04	Te-129m	2.3E-13	1E-07
Xe-133	1.3E-05	1E-04	Te-129	2.9E-13	3E-05
Xe-135m	3.1E-08	9E-06	Sb-131	4.7E-16	1E-05
Xe-135	4.1E-07	1E-05	Te-131m	6.2E-13	2E-07
Xe-138	2.7E-08	4E-06	Te-131	3.4E-13	2E-06
I-130	2.5E-10	3E-07	Te-132	6.8E-12	9E-08
I-131	6.4E-09	2E-08	Cs-132	3.3E-14	2E-06
I-132	3.4E-09	3E-06	Te-133m	6.5E-13	2E-06
I-133	1.1E-08	1E-07	Te-133	3.2E-13	9E-06
I-134	2.3E-09	2E-05	Cs-134	3.0E-11	4E-08
I-135	7.2E-09	7E-07	Te-134	1.2E-12	1E-05
Br-82	3.4E-13	2E-06	Cs-135m	3.6E-13	8E-05
Br-83	3.1E-12	3E-05	Cs-135	-	5E-07
Br-84	1.7E-12	2E-05	Cs-136	8.1E-12	3E-07
Rb-86	3.0E-13	3E-07	Cs-137	1.7E-11	6E-08
Rb-87	-	6E-07	Cs-138	4.0E-11	2E-05
Rb-88	1.7E-10	3E-05	Ba-140	9.2E-14	6E-07
Rb-89	3.9E-12	6E-05	La-140	2.4E-14	5E-07
Sr-89	7.5E-14	6E-08	La-141	6.2E-15	4E-06
Sr-90	4.9E-15	2E-09	Ce-141	1.4E-14	2E-07
Y-90	1.1E-15	3E-07	Ce-143	1.2E-14	7E-07
Sr-91	5.1E-14	1E-06	Pr-143	1.3E-14	3E-07
Y-91m	2.6E-14	7E-05	Ce-144	1.1E-14	6E-09
Y-91	1.2E-14	5E-08	Pr-144	1.1E-14	5E-05
Sr-92	2.8E-14	3E-06	Pm-147	1.2E-15	5E-08
Y-92	2.2E-14	3E-06	Sm-147	-	2E-11
Y-93	9.6E-15	1E-06	Eu-154	1.1E-16	8E-09
Zr-93	-	3E-09	Na-24	1.5E-12	2E-06
Zr-95	1.5E-14	5E-08	Cr-51	1.5E-13	8E-06
Nb-95m	1.1E-16	9E-07	Mn-54	1.0E-13	3E-07
Nb-95	1.5E-14	5E-07	Mn-56	5.2E-12	6E-06
Mo-99	1.8E-11	6E-07	Fe-55	9.9E-14	8E-07
Tc-99m	7.2E-12	6E-05	Fe-59	1.7E-14	1E-07
Tc-99	-	3E-07	Co-58	2.4E-13	3E-07
Mo-101	7.8E-13	6E-05	Co-60	3.5E-14	1E-08
Tc-101	7.5E-13	1E-04	Zn-65	2.9E-14	1E-07
Ru-103	1.2E-14	3E-07	H-3	1.4E-08	2E-05
Rh-103m	1.2E-14	5E-04			

Table 12.2-61 Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone III) (Sheet 6 of 6)

Nuclide	Activity ($\mu Ci/cm^3$)	10 CFR 20 DAC ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)	10 CFR 20 DAC ($\mu Ci/cm^3$)
Kr-83m	1.3E-10	1E-02	Ru-106	-	5E-09
Kr-85m	5.3E-10	2E-05	Ag-110m	-	4E-08
Kr-85	2.7E-08	1E-04	Te-125m	1.3E-16	2E-07
Kr-87	3.4E-10	5E-06	Te-127m	5.1E-16	1E-07
Kr-88	9.9E-10	2E-06	Te-127	2.7E-15	7E-06
Xe-131m	1.2E-09	4E-04	Sb-129	-	4E-06
Xe-133m	1.2E-09	1E-04	Te-129m	1.7E-15	1E-07
Xe-133	9.3E-08	1E-04	Te-129	2.2E-15	3E-05
Xe-135m	2.3E-10	9E-06	Sb-131	-	1E-05
Xe-135	3.0E-09	1E-05	Te-131m	4.6E-15	2E-07
Xe-138	2.0E-10	4E-06	Te-131	2.5E-15	2E-06
I-130	1.8E-12	3E-07	Te-132	5.0E-14	9E-08
I-131	4.7E-11	2E-08	Cs-132	2.4E-16	2E-06
I-132	2.5E-11	3E-06	Te-133m	4.8E-15	2E-06
I-133	8.1E-11	1E-07	Te-133	2.4E-15	9E-06
I-134	1.7E-11	2E-05	Cs-134	2.2E-13	4E-08
I-135	5.3E-11	7E-07	Te-134	8.6E-15	1E-05
Br-82	2.5E-15	2E-06	Cs-135m	2.6E-15	8E-05
Br-83	2.3E-14	3E-05	Cs-135	-	5E-07
Br-84	1.2E-14	2E-05	Cs-136	5.9E-14	3E-07
Rb-86	2.2E-15	3E-07	Cs-137	1.3E-13	6E-08
Rb-87	-	6E-07	Cs-138	2.9E-13	2E-05
Rb-88	1.3E-12	3E-05	Ba-140	6.8E-16	6E-07
Rb-89	2.9E-14	6E-05	La-140	1.8E-16	5E-07
Sr-89	5.6E-16	6E-08	La-141	-	4E-06
Sr-90	-	2E-09	Ce-141	1.0E-16	2E-07
Y-90	-	3E-07	Ce-143	-	7E-07
Sr-91	3.7E-16	1E-06	Pr-143	-	3E-07
Y-91m	1.9E-16	7E-05	Ce-144	-	6E-09
Y-91	-	5E-08	Pr-144	-	5E-05
Sr-92	2.1E-16	3E-06	Pm-147	-	5E-08
Y-92	1.6E-16	3E-06	Sm-147	-	2E-11
Y-93	-	1E-06	Eu-154	-	8E-09
Zr-93	-	3E-09	Na-24	1.1E-14	2E-06
Zr-95	1.1E-16	5E-08	Cr-51	1.1E-15	8E-06
Nb-95m	-	9E-07	Mn-54	7.6E-16	3E-07
Nb-95	1.1E-16	5E-07	Mn-56	3.8E-14	6E-06
Mo-99	1.3E-13	6E-07	Fe-55	7.3E-16	8E-07
Tc-99m	5.3E-14	6E-05	Fe-59	1.3E-16	1E-07
Tc-99	-	3E-07	Co-58	1.8E-15	3E-07
Mo-101	5.8E-15	6E-05	Co-60	2.6E-16	1E-08
Tc-101	5.5E-15	1E-04	Zn-65	2.1E-16	1E-07
Ru-103	-	3E-07	H-3	1.0E-10	2E-05
Rh-103m	-	5E-04			

Table 12.2-62 Chemical and Volume Control System Radiation Sources Deborating Demineralizer Activity (70 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	1.5E-02
Br-83	2.6E-02
Br-84	3.1E-03
I-130	1.4E-01
I-131	3.4E+00
I-132	2.1E+00
I-133	4.3E+00
I-134	7.4E-02
I-135	1.5E+00

Table 12.2-63 Chemical and Volume Control System Radiation Sources
Deborating Demineralizer Source Strength (70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	1.3E+01
0.03	2.6E+02
0.08	2.6E+02
0.1	9.0E-01
0.15	6.7E+01
0.2	3.7E+02
0.3	4.0E+03
0.4	4.4E+04
0.5	8.4E+04
0.6	7.0E+04
0.8	8.1E+04
1.0	4.9E+04
1.5	7.1E+04
2.0	1.9E+04
3.0	1.6E+02
4.0	3.3E+01

Table 12.2-64 Chemical and Volume Control System Radiation Sources B.A. Evaporator Feed Demineralizer Activity (70 ft^3 of Resin)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Br-82	1.2E-02	Te-129m	2.0E-01
Br-83	3.0E-03	Te-129	1.5E-04
Br-84	8.5E-05	Te-131m	3.7E-02
Rb-86	2.9E+00	Te-131	2.1E-05
Rb-88	1.1E+00	Te-132	1.1E+00
Rb-89	1.3E-03	Te-133m	2.0E-04
Sr-89	9.3E-02	Te-134	2.1E-04
Sr-90	5.9E-03	I-130	1.5E+00
Sr-91	7.9E-04	I-131	1.3E+01
Sr-92	6.7E-05	I-132	1.3E+00
Y-90	1.3E-01	I-133	2.2E+00
Y-91m	5.3E-04	I-134	3.8E-03
Y-91	1.3E-02	I-135	3.4E-01
Y-92	2.8E-04	Cs-132	5.5E-01
Y-93	1.6E-04	Cs-134	5.0E+02
Zr-95	1.5E-02	Cs-135m	1.4E-03
Nb-95	4.0E-02	Cs-136	6.3E+01
Mo-99	2.5E+00	Cs-137	2.9E+02
Mo-101	1.7E-05	Cs-138	9.1E-02
Tc-99m	3.5E+00	Ba-137m	2.5E+02
Ru-103	1.1E-02	Ba-140	5.2E-02
Ru-106	5.0E-03	La-140	1.0E-01
Ag-110m	4.5E-05	Ce-141	1.2E-02
Te-125m	1.7E-02	Ce-143	8.0E-04
Te-127m	7.5E-02	Ce-144	1.2E-02
		Pr-144	1.7E-02
		Pm-147	1.4E-03
		Eu-154	1.3E-04
		Na-24	4.2E-02
		Cr-51	1.2E-01
		Mn-54	1.2E-01
		Mn-56	1.1E-02
		Fe-55	1.2E-01
		Fe-59	1.6E-02
		Co-58	2.5E-01
		Co-60	4.2E-02
		Zn-65	3.3E-02

Table 12.2-65 Chemical and Volume Control System Radiation Sources B.A. Evaporator Feed Demineralizer Source Strength (70 ft^3 of Resin)

Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	2.6E+03
0.02	2.7E+02
0.03	3.3E+04
0.04	9.8E+03
0.05	3.0E+02
0.06	1.8E+04
0.08	1.3E+04
0.1	1.1E+03
0.15	6.3E+04
0.2	7.7E+04
0.3	4.3E+05
0.4	1.6E+05
0.5	2.2E+05
0.6	1.9E+07
0.8	1.6E+07
1.0	2.9E+06
1.5	8.9E+05
2.0	2.5E+04
3.0	9.2E+03
4.0	4.3E+01
5.0	3.0E+02

Table 12.2-66 Chemical and Volume Control System Radiation Sources
B.A. Evaporator Activity

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Br-82	2.0E-04	Te-129m	4.0E-04
Br-83	9.8E-05	Te-129	5.1E-06
Br-84	2.8E-06	Te-131m	6.7E-04
Rb-86	6.2E-02	Te-131	7.0E-07
Rb-88	3.1E-01	Te-132	9.7E-03
Rb-89	3.7E-04	Te-133m	6.5E-06
Sr-89	1.7E-04	Te-134	6.7E-06
Sr-90	8.4E-06	I-130	1.8E-03
Sr-91	2.4E-05	I-131	4.4E-02
Sr-92	2.2E-06	I-132	4.9E-02
Y-90	1.3E-03	I-133	4.3E-02
Y-91m	3.1E-05	I-134	1.3E-04
Y-91	2.2E-05	I-135	1.0E-02
Y-92	1.1E-05	Cs-132	7.1E-03
Y-93	4.8E-06	Cs-134	6.5E+00
Zr-95	2.5E-05	Cs-135m	4.1E-04
Nb-95	9.0E-05	Cs-136	1.6E+00
Mo-99	2.4E-02	Cs-137	3.7E+00
Mo-101	5.5E-07	Cs-138	2.7E-02
Tc-99m	1.3E-01	Ba-137m	2.0E+01
Ru-103	2.0E-05	Ba-140	1.5E-04
Ru-106	7.3E-06	La-140	1.7E-03
Ag-110m	6.7E-08	Ce-141	2.4E-05
Te-125m	2.9E-05	Ce-143	1.3E-05
Te-127m	1.2E-04	Ce-144	1.8E-05
		Pr-144	9.3E-04
		Pm-147	2.0E-06
		Eu-154	1.9E-07
		Na-24	1.1E-03
		Cr-51	2.5E-04
		Mn-54	1.8E-04
		Mn-56	3.7E-04
		Fe-55	1.7E-04
		Fe-59	3.0E-05
		Co-58	4.1E-04
		Co-60	6.1E-05
		Zn-65	5.0E-05

Table 12.2-67 Chemical and Volume Control System Radiation Sources
B.A. Evaporator Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	1.3E+02
0.02	7.8E+00
0.03	1.7E+03
0.04	5.2E+02
0.05	2.6E+00
0.06	4.6E+02
0.08	3.1E+02
0.1	2.7E+01
0.15	1.8E+03
0.2	1.8E+03
0.3	1.1E+04
0.4	5.9E+02
0.5	3.3E+03
0.6	5.7E+05
0.8	2.3E+05
1.0	6.9E+04
1.5	1.3E+04
2.0	5.6E+03
3.0	1.4E+03
4.0	1.1E+01
5.0	8.8E+01

Table 12.2-68 Chemical and Volume Control System Radiation Sources
B.A. Evaporator Vent Condenser Activity

Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	7.3E-01
Kr-85m	4.5E+00
Kr-85	4.3E+02
Kr-87	1.1E+00
Kr-88	6.3E+00
Xe-131m	1.9E+01
Xe-133m	1.9E+01
Xe-133	1.4E+03
Xe-135m	1.6E+00
Xe-135	3.9E+01
Xe-138	1.1E-01

Table 12.2-69 Chemical and Volume Control System Radiation Sources B.A. Evaporator Vent Condenser Source Strength

Gamma Ray Energy (MeV)	Source Strength (MeV/cm^3/sec)
0.015	4.3E+04
0.03	7.6E+05
0.04	3.7E+04
0.06	1.1E-01
0.08	1.6E+06
0.1	4.6E+01
0.15	2.7E+04
0.2	2.9E+05
0.3	8.3E+03
0.4	1.5E+04
0.5	5.9E+04
0.6	2.7E+04
0.8	3.2E+04
1.0	1.5E+04
1.5	6.4E+04
2.0	2.9E+05
3.0	2.3E+04

Table 12.2-70 Parameters and Assumptions for Calculating Spent Fuel Source Strength

Parameter / Assumption	Value
Burn up	62 GWd/MTU
Core thermal output	4451 MWt
Number of fuel assemblies	257
Specific power	32.1 MW/MTU
Fuel enrichment	4.55 wt%
Effective fuel length and width	420 cm x 21.4 cm x 21.4 cm

Table 12.2-71 Parameters and Assumptions for Calculating Irradiated Incore Detector, Drive Cable and Flux Thimble Source Strength

Parameter / Assumption	Value				
	Detector		Drive Cable		Flux Thimble
	Fission Chamber	Fe-Ni cover			
Chemical composition *1 (%)	-*4	Fe: 49 Cr: - Ni: 50 Mn: 0.7 Co: 0.03 P: - S: - Si: 0.19 Mo: -	Fe: 65 Cr: 19 Ni: 13 Mn: 2.0 Co: 0.2 P: 0.04 S: 0.03 Si: 1.0 Mo: -	Fe: 98 Cr: 0.15 Ni: 0.7 Mn: 0.6 Co: 0.02 P: 0.04 S: 0.04 Si: 0.35 Mo: 0.6	Fe: 66 Cr: 18 Ni: 14 Mn: 2.0 Co: 0.1 P: - S: - Si: - Mo: -
Density (g/cm^3)	-*4	7.85	7.9	7.85	7.98
Mixing ratio *2	-*4	1.0	0.1	0.9	1.0
Neutron flux *3 ($n/cm^2/s$)	ϕ_1: 8.5E+13 ϕ_2: 1.4E+14 ϕ_3: 9.5E+13 ϕ_4: 6.0E+13		ϕ_1: 8.5E+13 ϕ_2: 1.4E+14 ϕ_3: 9.5E+13 ϕ_4: 6.0E+13	ϕ_1: 8.5E+13 ϕ_2: 1.4E+14 ϕ_3: 9.5E+13 ϕ_4: 6.0E+13	ϕ_1: 8.2E+13 ϕ_2: 1.4E+14 ϕ_3: 9.3E+13 ϕ_4: 5.8E+13
Irradiation period	20 (h)		20 (h)	20 (h)	60 (y)

*1 For chemical composition, a "-" after an element indicates "essentially zero".

*2 Drive cable consists of two materials. Source strength for drive cable was calculated by summing source strength for each material with this mixing ratio.

*3 Subscripts for neutron flux indicate range of neutron energy. (1: E > 1MeV, 2: 5.53 keV < E < 1 MeV, 3: 0.414 eV < E < 5.53 keV, 4: E < 0.414 eV)

*4 The mass of U-235 in Fission Chamber is 1.6 (mg)

Table 12.2-72 Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium) (Sheet 1 of 2)

Nuclide	Specific Activity (μCi/g)	Nuclide	Specific Activity (μCi/g)
Kr-83m	1.9E-06	Ru-106	3.0E-09
Kr-85m	4.0E-05	Ag-110m	2.7E-11
Kr-85	5.4E-01	Te-125m	1.2E-08
Kr-87	4.4E-10	Te-127m	4.8E-08
Kr-88	5.2E-06	Te-127	1.0E-04
Xe-131m	3.8E-02	Sb-129	1.7E-11
Xe-133m	2.6E-02	Te-129m	1.6E-07
Xe-133	2.7E+00	Te-129	5.9E-10
Xe-135m	2.3E-04	Sb-131	-
Xe-135	1.6E-02	Te-131m	2.5E-07
Xe-138	-	Te-131	-
I-130	1.7E-06	Te-132	3.8E-06
I-131	3.9E-05	Cs-132	3.3E-07
I-132	5.4E-04	Te-133m	6.7E-15
I-133	3.3E-05	Te-133	2.7E-14
I-134	1.1E-13	Cs-134	3.0E-04
I-135	3.9E-06	Te-134	3.5E-17
Br-82	1.4E-07	Cs-135m	2.4E-14
Br-83	2.0E-09	Cs-135	8.1E-10
Br-84	-	Cs-136	7.6E-05
Rb-86	2.9E-06	Cs-137	1.7E-04
Rb-87	3.7E-14	Cs-138	2.0E-17
Rb-88	5.8E-06	Ba-140	6.1E-08
Rb-89	-	La-140	4.4E-06
Sr-89	5.2E-08	La-141	6.3E-11
Sr-90	3.4E-09	Ce-141	9.7E-09
Y-90	5.9E-06	Ce-143	5.0E-09
Sr-91	6.1E-09	Pr-143	1.3E-07
Y-91m	1.8E-07	Ce-144	7.4E-09
Y-91	9.2E-09	Pr-144	1.6E-04
Sr-92	4.3E-11	Pm-147	8.3E-10
Y-92	1.7E-09	Sm-147	9.2E-20
Y-93	1.3E-09	Eu-154	7.8E-11
Zr-93	1.4E-14	Na-24	3.6E-07
Zr-95	1.0E-08	Cr-51	1.0E-07
Nb-95m	1.2E-08	Mn-54	7.2E-08
Nb-95	1.9E-07	Mn-56	5.7E-09
Mo-99	9.6E-06	Fe-55	6.9E-08
Tc-99m	7.2E-04	Fe-59	1.2E-08
Tc-99	3.9E-12	Co-58	1.7E-07
Mo-101	-	Co-60	2.5E-08

Table 12.2-72 Reactor cavity and SFP water specific activity in refueling /shutdown (except tritium) (Sheet 2 of 2)

Nuclide	Specific Activity (µCi/g)	Nuclide	Specific Activity (µCi/g)
Tc-101	-	Zn-65	2.0E-08
Ru-103	8.3E-09		
Rh-103m	2.3E-05		

Note: The initial concentrations used in the calculation that determined the values in this table conservatively exclude the decay and purification at 180 gpm during the 4-hour time period prior to RHRS initiation. Decay and purification are credited for 24 hours after connection of RCS to RHRS.

Table 12.2-73 Parameters for the US-APWR demineralizers

Component	Parameters				Note
	DF	Flow rate	Term of Service	inlet flow stream activity concentration	
Mixed bed demineralizer	Kr, Xe=1, Br, I=100, Cs, Rb=2 , Others=50	180 gpm	731 days	Table 12.2-13	The values in the left columns are listed in Table 11.1-1.
Cation-bed demineralizer	Kr, Xe=1, Br, I=1, Cs, Rb=10 , Others=10	18 gpm	731 days	Table 12.2-74	
Deborating demineralizer	Anion=100, Cs, Rb=1, Others=1	180 gpm	22 hours	Table 12.2-74	
B.A. evaporator feed demineralizer	Anion=10, Cs, Rb=2, Others=10	30 gpm	780 hours	Table 12.2-75	
Waste demineralizer (Anion-bed)	I=100, Cs, Rb=1, Others=1	90 gpm	280 hours	Table 12.2-37	
Waste demineralizer (Cation-bed)	I=1, Cs, Rb=10, Others=10	90 gpm	280 hours	Table 12.2-76	
Waste demineralizer (Mixed bed) In case of treating HT system	Kr, Xe=1, I=5, Cs, Rb=1 , Others=10	30 gpm	780 hours	Table 12.2-77	Parameters used when treating distilled water in the boron recycle system
Waste demineralizer (Mixed bed) In case of treating WHT system	Kr, Xe=1, I=100, Cs, Rb=2 , Others=100	90 gpm	280 hours	Table 12.2-78	Parameters used when treating waste liquid in the waste liquid storage tank
Spent fuel pit demineralizer	Kr, Xe=1, Br, I=100 Cs, Rb=2 , Others=100	265 gpm	731 days	Table 12.2-34	
SG Blowdown demineralizer	Br, I=100, Cs, Rb=100, Others=1000	1.554E+05 lb/hr	731 days	Table 11.1-5	

Table 12.2-74 Inlet Flow Stream Activity of Cation-bed demineralizer and Deborating demineralizer

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	5.5E-01	Te-129m	1.2E-04
Kr-85m	1.8E+00	Te-129	1.5E-04
Kr-85	9.2E+01	Te-131m	3.1E-04
Kr-87	1.2E+00	Te-131	1.7E-04
Kr-88	3.3E+00	Te-132	3.4E-03
Xe-131m	4.1E+00	Te-133m	3.2E-04
Xe-133m	4.2E+00	Te-134	5.8E-04
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	7.9E+00	I-131	1.6E-02
Xe-135	1.2E+01	I-132	6.5E-02
Xe-138	6.7E-01	I-133	2.7E-02
		I-134	6.1E-03
Br-82	8.6E-05	I-135	1.8E-02
Br-83	7.8E-04	Cs-132	4.1E-04
Br-84	4.2E-04	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	4.5E-03
Rb-88	2.1E+00	Cs-136	1.0E-01
Rb-89	4.9E-02	Cs-137	2.2E-01
Sr-89	3.8E-05	Cs-138	4.9E-01
Sr-90	2.4E-06	Ba-137m	1.1E+03
Sr-91	2.5E-05	Ba-140	4.6E-05
Sr-92	1.4E-05	La-140	3.5E-04
Y-90	4.4E-04	Ce-141	7.1E-06
Y-91m	1.8E-04	Ce-143	6.0E-06
Y-91	6.1E-06	Ce-144	5.3E-06
Y-92	2.2E-05	Pr-144	1.0E-01
Y-93	4.8E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.9E-03	Na-24	7.7E-04
Mo-101	3.9E-04	Cr-51	7.5E-05
Tc-99m	8.7E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	2.6E-03
Ru-106	2.1E-06	Fe-55	5.0E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.7E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

Table 12.2-75 Inlet Flow Stream Activity of B.A. evaporator feed demineralizer

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	1.6E-01	Te-129m	1.2E-04
Kr-85m	9.7E-01	Te-129	2.8E-05
Kr-85	9.2E+01	Te-131m	2.8E-04
Kr-87	2.4E-01	Te-131	1.1E-05
Kr-88	1.4E+00	Te-132	3.3E-03
Xe-131m	4.1E+00	Te-133m	4.8E-05
Xe-133m	4.0E+00	Te-134	6.6E-05
Xe-133	3.1E+02	I-130	6.2E-04
Xe-135m	3.3E-01	I-131	1.6E-02
Xe-135	8.4E+00	I-132	2.5E-02
Xe-138	2.6E-02	I-133	2.4E-02
		I-134	9.3E-04
Br-82	7.9E-05	I-135	1.2E-02
Br-83	2.8E-04	Cs-132	4.1E-04
Br-84	3.6E-05	Cs-134	3.8E-01
Rb-86	3.7E-03	Cs-135m	6.4E-04
Rb-88	1.4E+00	Cs-136	9.9E-02
Rb-89	2.0E-03	Cs-137	2.2E-01
Sr-89	4.8E-05	Cs-138	6.9E-02
Sr-90	2.4E-06	Ba-137m	8.1E+00
Sr-91	1.9E-05	Ba-140	4.6E-05
Sr-92	5.6E-06	La-140	3.3E-04
Y-90	4.2E-04	Ce-141	7.1E-06
Y-91m	3.4E-05	Ce-143	5.5E-06
Y-91	6.2E-06	Ce-144	5.3E-06
Y-92	1.4E-05	Pr-144	4.8E-03
Y-93	3.6E-06	Pm-147	6.0E-07
Zr-95	7.3E-06	Eu-154	5.6E-08
Nb-95	2.0E-05		
Mo-99	8.5E-03	Na-24	6.3E-04
Mo-101	1.6E-05	Cr-51	7.5E-05
Tc-99m	5.7E-02	Mn-54	5.1E-05
Ru-103	6.0E-06	Mn-56	9.8E-04
Ru-106	2.1E-06	Fe-55	4.9E-05
Ag-110m	1.9E-08	Fe-59	8.7E-06
Te-125m	8.6E-06	Co-58	1.2E-04
Te-127m	3.4E-05	Co-60	1.8E-05
		Zn-65	1.4E-05

Table 12.2-76 Inlet Flow Stream Activity of Waste Demineralizer (Cation Bed)

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Xe-131m	3.1E-03	Te-129m	2.5E-03
Xe-133m	1.2E-02	Te-129	2.0E-03
Xe-133	1.8E-01	Te-131m	6.3E-03
Xe-135m	2.5E+00	Te-131	2.2E-03
Xe-135	4.6E-01	Te-132	7.0E-02
		Te-133m	4.3E-03
Br-82	3.5E-03	Te-134	7.6E-03
Br-83	2.4E-02	I-130	2.7E-04
Br-84	1.1E-02	I-131	6.7E-03
Rb-86	1.1E-02	I-132	2.9E-03
Rb-88	1.4E+00	I-133	1.1E-02
Rb-89	2.5E-02	I-134	1.5E-03
Sr-89	8.3E-04	I-135	6.4E-03
Sr-90	5.4E-05	Cs-132	2.2E-03
Sr-91	4.7E-04	Cs-134	2.0E+00
Sr-92	2.2E-04	Cs-135m	2.4E-03
Y-90	1.8E-04	Cs-136	2.5E-01
Y-91m	2.7E-04	Cs-137	1.2E+00
Y-91	1.3E-04	Cs-138	2.6E-01
Y-92	2.1E-04	Ba-137m	8.0E+00
Y-93	9.0E-05	Ba-140	9.8E-04
Zr-95	1.6E-04	La-140	4.2E-04
Nb-95	1.8E-04	Ce-141	1.5E-04
Mo-99	1.8E-01	Ce-143	1.2E-04
Mo-101	5.0E-03	Ce-144	1.2E-04
Tc-99m	1.1E-01	Pr-144	2.9E-03
Ru-103	1.3E-04	Pm-147	1.3E-05
Ru-106	4.7E-05	Eu-154	1.2E-06
Ag-110m	4.3E-07		
Te-125m	1.9E-04	Na-24	1.5E-02
Te-127m	7.5E-04	Cr-51	1.6E-03
		Mn-54	1.1E-03
		Mn-56	4.0E-02
		Fe-55	1.1E-03
		Fe-59	1.9E-04
		Co-58	2.6E-03
		Co-60	3.9E-04
		Zn-65	3.2E-04

Table 12.2-77 Inlet Flow Stream Activity of Waste Demineralizer (Mixed bed)*

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
Kr-83m	1.6E-04	Te-129m	1.2E-08
Kr-85m	9.7E-04	Te-129	2.9E-09
Kr-85	9.2E-02	Te-131m	2.8E-08
Kr-87	2.4E-04	Te-131	1.2E-09
Kr-88	1.4E-03	Te-132	3.3E-07
Xe-131m	4.1E-03	Te-133m	4.8E-09
Xe-133m	4.0E-03	Te-134	6.6E-09
Xe-133	3.1E-01	I-130	6.2E-07
Xe-135m	3.7E-04	I-131	1.6E-05
Xe-135	8.4E-03	I-132	1.2E-04
Xe-138	2.6E-05	I-133	2.4E-05
		I-134	9.8E-07
Br-82	7.9E-09	I-135	1.2E-05
Br-83	2.8E-08	Cs-132	2.1E-07
Br-84	3.6E-09	Cs-134	1.9E-04
Rb-86	1.9E-06	Cs-135m	3.2E-07
Rb-88	7.2E-04	Cs-136	5.0E-05
Rb-89	1.0E-06	Cs-137	1.1E-04
Sr-89	4.8E-09	Cs-138	3.4E-05
Sr-90	2.4E-10	Ba-137m	1.3E-01
Sr-91	1.9E-09	Ba-140	4.6E-09
Sr-92	5.6E-10	La-140	5.9E-08
Y-90	4.4E-08	Ce-141	7.1E-10
Y-91m	1.4E-08	Ce-143	5.5E-10
Y-91	6.3E-10	Ce-144	5.3E-10
Y-92	1.8E-09	Pr-144	1.4E-06
Y-93	3.6E-10	Pm-147	6.0E-11
Zr-95	7.3E-10	Eu-154	5.6E-12
Nb-95	2.4E-09		
Mo-99	8.5E-07	Na-24	6.3E-08
Mo-101	1.6E-09	Cr-51	7.5E-09
Tc-99m	1.3E-05	Mn-54	5.1E-09
Ru-103	6.0E-10	Mn-56	9.8E-08
Ru-106	2.1E-10	Fe-55	4.9E-09
Ag-110m	1.9E-12	Fe-59	8.7E-10
Te-125m	8.6E-10	Co-58	1.2E-08
Te-127m	3.4E-09	Co-60	1.8E-09
		Zn-65	1.4E-09

*: These activities are used when this demineralizer processes the distilled water from the boron recycle system.

Table 12.2-78 Inlet Flow Stream Activity of Waste Demineralizer (Mixed bed)*

Nuclide	Activity (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)
Kr-83m	2.8E-02	Te-129m	2.5E-04
Xe-131m	3.1E-03	Te-129	2.0E-04
Xe-133m	1.2E-02	Te-131m	6.3E-04
Xe-133	1.8E-01	Te-131	2.2E-04
Xe-135m	2.5E+00	Te-132	7.0E-03
Xe-135	4.6E-01	Te-133m	4.3E-04
		Te-134	7.6E-04
Br-82	3.5E-04	I-130	2.7E-04
Br-83	2.4E-03	I-131	6.7E-03
Br-84	1.1E-03	I-132	2.0E+00
Rb-86	1.1E-03	I-133	1.1E-02
Rb-88	1.4E-01	I-134	7.0E-03
Rb-89	2.5E-03	I-135	6.4E-03
Sr-89	8.3E-05	Cs-132	2.2E-04
Sr-90	5.4E-06	Cs-134	2.0E-01
Sr-91	4.7E-05	Cs-135m	2.4E-04
Sr-92	2.2E-05	Cs-136	2.5E-02
Y-90	3.3E-05	Cs-137	1.2E-01
Y-91m	3.1E-04	Cs-138	2.6E-02
Y-91	1.3E-05	Ba-137m	4.6E+02
Y-92	3.7E-05	Ba-140	9.8E-05
Y-93	9.0E-06	La-140	3.6E-04
Zr-95	1.6E-05	Ce-141	1.5E-05
Nb-95	2.1E-05	Ce-143	1.2E-05
Mo-99	1.8E-02	Ce-144	1.2E-05
Mo-101	5.0E-04	Pr-144	7.3E-03
Tc-99m	1.6E-01	Pm-147	1.3E-06
Ru-103	1.3E-05	Eu-154	1.2E-07
Ru-106	4.7E-06		
Ag-110m	4.3E-08	Na-24	1.5E-03
Te-125m	1.9E-05	Cr-51	1.6E-04
Te-127m	7.5E-05	Mn-54	1.1E-04
		Mn-56	4.0E-03
		Fe-55	1.1E-04
		Fe-59	1.9E-05
		Co-58	2.6E-04
		Co-60	3.9E-05
		Zn-65	3.2E-05

*: These activities are used when this demineralizer processes the liquid effluent from the waste holdup tank.

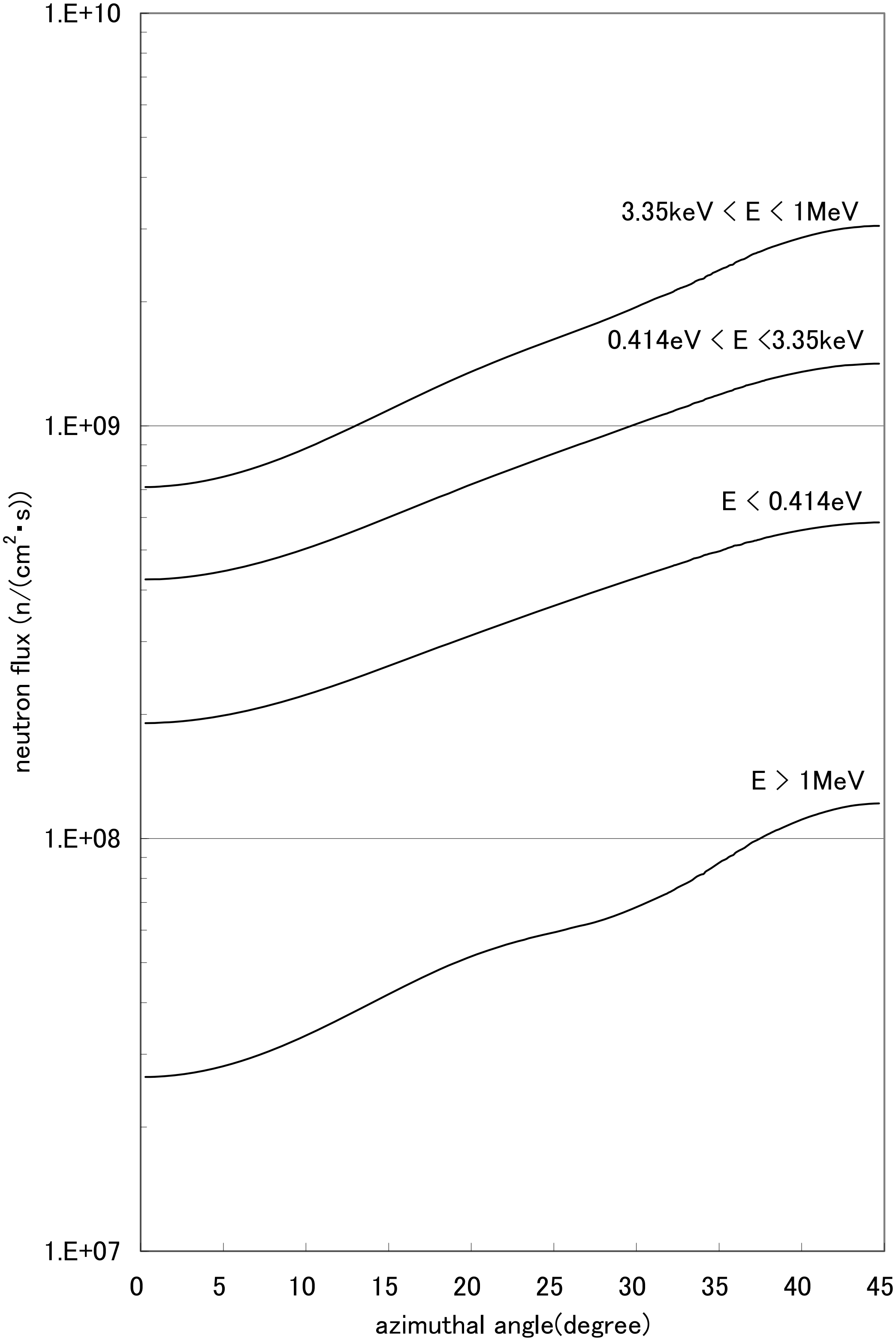

Figure 12.2-1 Azimuthal Distribution of Neutron Flux Incident on the Primary Shield at the Reactor Core Midplane

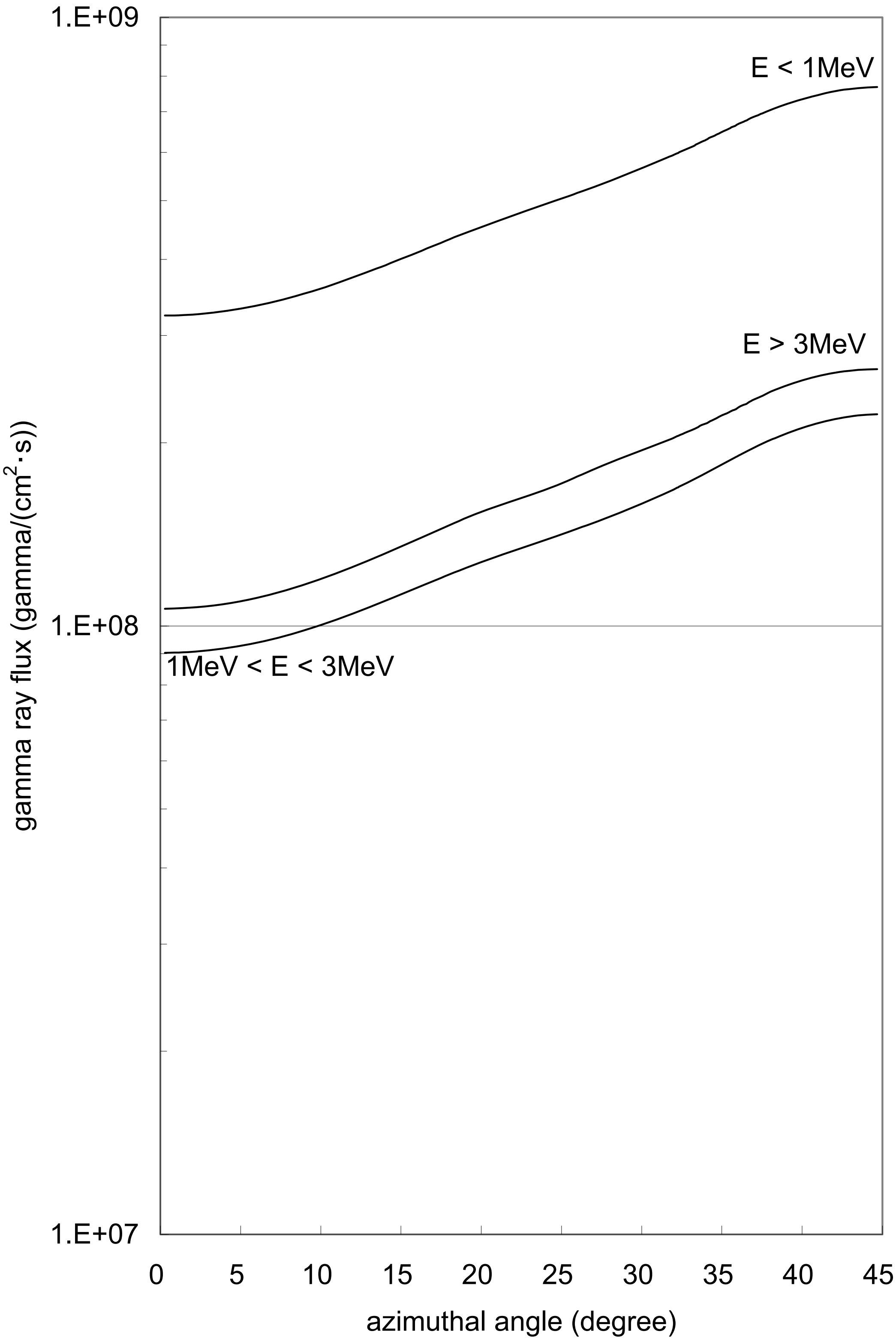

Figure 12.2-2 Azimuthal Distribution of Gamma Ray Flux Incident on the Primary Shield at the Reactor Core Midplane

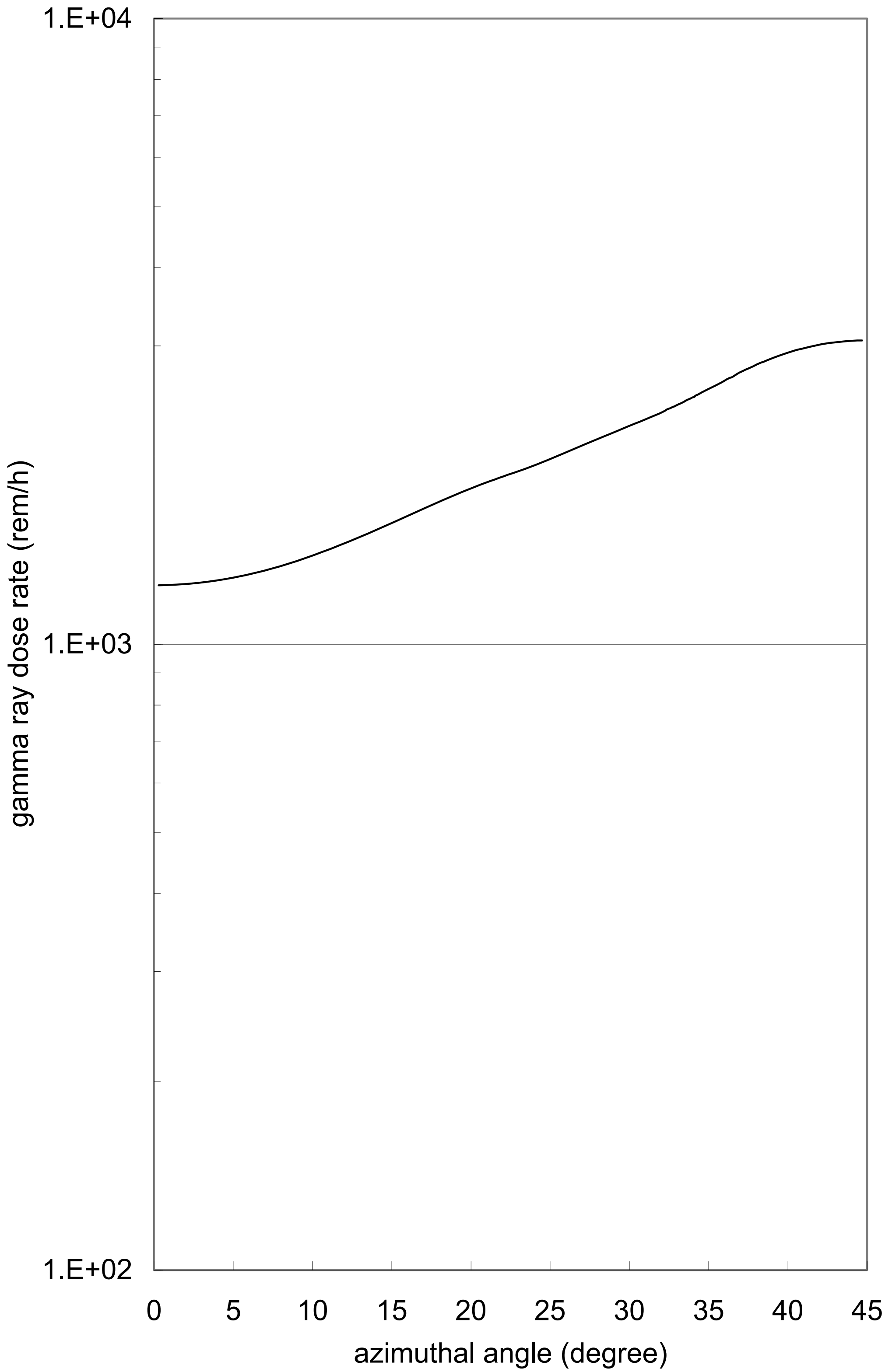

Figure 12.2-3 Azimuthal Distribution of Gamma Ray Dose Rate Incident on the Primary Shield at the Reactor Core Midplane

12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

Specific design features for maintaining personnel exposure ALARA are discussed in this section. The design feature recommendations, in accordance with the guidance in RG 8.8, Paragraph C.2 (Reference 12.3-1), are utilized to minimize exposures to personnel.

12.3.1.1 Plant Design Features for As Low As Reasonably Achievable

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of Subsection 12.1.2 and are outlined in this subsection for several general classes of equipment (Subsection 12.3.1.1.1) and several typical plant layout situations (Subsection 12.3.1.1.2).

12.3.1.1.1 Common Equipment and Component Designs for As Low As Reasonably Achievable

This paragraph describes the design features utilized for several general classes of equipment or components. These classes of equipment are common to many of the plant systems; thus, the features employed for each system to keep exposures ALARA are similar and are discussed by equipment class in the following paragraphs.

12.3.1.1.1.1 Nuclear Steam Supply System Equipment

A. Reactor Vessel

The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the reinforced areas to ensure a smooth transition, and pipe branch locations are selected to ensure no interference from one branch to the next. All weld-to-pipe interfaces have a smooth, high quality finish.

B. RCPs

The RCP design includes the use of an assembled cartridge seal for the No. 2 and No. 3 pump seal that reduces the time required for replacement. The RCP design also includes a spool piece to facilitate the assembly or disassembly of the seal system without the replacement of the motor from the pump.

C. Reactor Vessel Insulation

Insulation, in the area of the reactor vessel nozzle welds, is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate the removal of the insulation for the inspection of the welds.

D. SGs

The SGs incorporate several design features to facilitate maintenance and inspection in reduced radiation fields. The SGs have the following design aspects:

1. Manways of the channel head are sized to facilitate access for tube bundle inspections and maintenance.

2. The channel head has a cylindrical region just below the tube sheet primary side to enhance the access of tooling to all tubes, including those on the periphery of the tube bundle.

3. Rapid entry/exit nozzle dam systems are provided in both primary nozzles to minimize occupational radiation exposure and to enhance personnel safety.

The specification of low cobalt tubing material for the US-APWR steam generator design is an important feature of the design; not only in terms of reduced exposure relative to the steam generator, but to the total plant radiation source term. The cobalt content is controlled to be not more than 0.016 mass percent and an average of 0.014 mass percent for the US-APWR steam generator tubing.

E. Materials

Equipment specifications for components exposed to high temperature reactor coolant contain limitations on the cobalt content of the base metal as given in Table 12.3-7. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations.

Nickel-based alloys in the reactor coolant system (Co-58 is produced from activation of Ni-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the reactor coolant system is the inconel steam generator tube.

12.3.1.1.1.2 Balance of Plant Equipment

The following subsections describe the general design criteria for several types of plant components. The materials selection criteria described in Subsection 12.3.1.1.1.1 E are also applicable to the various types of equipment described below. The component specifications for the procurement of components for specific applications will be based on the latest relevant national and international industry guidance in order to ensure consistency with regulatory requirements related to ALARA best practices, contamination minimization, and component reliability needs.

A. Filters

Filters that accumulate radioactivity are supplied with the means either to back-flush the filter remotely or to perform cartridge replacement with semi-remote tools.

For cartridge filters, adequate space is provided to allow removal, cask loading, and transportation of the cartridge to the solid radwaste area.

Back-flushable filters are designed so that the filter internals may be remotely removed and placed in a shielded cask for offsite shipping and disposal, in the unlikely event that a filter loses its back-flush capability.

Liquid systems containing radioactive cartridge filters are provided with a remote filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the drumming station for packaging and shipment for burial. The process is accomplished so that exposure to personnel and the possibility of an inadvertent radioactive release to the environment are minimized. Each filter is contained in a shielded compartment and provided with vent and drain valving, and individual compartments have drainage capabilities. The filter handling system has also been designed with a minimum of components susceptible to malfunction.

B. Demineralizers

Demineralizers for highly radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin storage tanks so that fresh resin can be loaded into the demineralizer remotely. The demineralizers and piping are designed with the ability to be flushed with demineralized water. Strainers are installed in the vent lines to prevent the entry of spent resin into the exhaust duct.

C. Evaporators

Adequate space and flanged connections for easy removal are provided for the maintenance of evaporator components. Additionally, the evaporator can be operated in an automatic operation mode that can reduce the exposure of the operator to radiation from the equipment.

D. Pumps

Wherever practicable, pumps are sealed with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed to allow easy removal, if necessary. All pumps in the radioactive waste systems are provided with flanged connections for ease of removal. Pump casings are provided with drain connections for draining pumps for maintenance.

Pumps used for radioactive fluids should utilize seal cartridges, as applicable, to minimize field repair time, reduce exposure times, minimize errors in reassembly, and reduce the consequent leakage potential.

E. Tanks

Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to minimize the potential for the spread of contamination within plant structures. Tanks containing radioactive fluids have overboard lines at least equal in size to the largest inlet line. The tank vent line is either open to the cubicle or connected directly to the ventilation system. The spent resin tank vents are equipped with a break-pot, which separates the air from the moisture and any entrained resin, which are subsequently sent to the A/B sump tank, and vents the air to the exhaust ductwork. These measures minimize the possible contamination of the area and the ductwork. Tanks containing radioactive particulate material shall have smooth welds and mixing, flushing and cleaning capabilities to prevent retention of the radioactive particulate material. Tanks containing radioactive particulate material shall include one or more of the features mentioned below:

- Purification of radioactive fluids (filtration or ion exchange) prior to entering the tank
- Sloped or cone-shaped tank bottom
- Grinding of internal welds
- Tank flushing capability
- Agitation by recirculation flow capability
- Lancing or chemical cleaning capability

Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The cubicles are equipped with a drainage system to direct any leakage and overflows to sumps with pumps to redirect flow to other tanks. The drainage system is equipped with a liquid detection instrument which can provide early warning for leakage and/or overflow condition to initiate operator actions. The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation, and provided with coating with non-porous material to prevent cross contamination.

F. Heat Exchangers

Most of the heat exchangers are shell and tube type heat exchangers. These are provided with corrosion-resistant tubes of stainless steel or other suitable materials to minimize leakage. Impact baffles are provided and the tube side and shell side velocities are limited to minimize erosion effects. Wherever possible, the radioactive fluid passes through the tube side of the heat exchanger.

Some heat exchangers, specifically the SFP heat exchangers and the component cooling water heat exchangers, are plate-type heat exchangers constructed of austenitic stainless steel. For the SFP heat exchangers, the SFP water circulates through one side of the heat exchanger while the component cooling water circulates through the other side. For the component cooling water heat exchangers, the component cooling water circulates through one side of the heat exchanger while the essential service water circulates through the other side.

G. Instruments

Instruments are located in low radiation zones and away from radiation sources, whenever practical.

Instruments, which are located in high radiation zones, are designed for easy removal to a lower radiation zone for calibration.

Transmitters and readout devices are located in low radiation zones, such as corridors and the control room.

Some instruments in high radiation zones, such as thermocouples, are provided in duplicate to reduce required access and service time. In the containment, instruments are located outside the secondary shield (the area of lowest radiation at power and during shutdown), whenever practical.

Check sources for response verification for airborne radiation monitors and safety-related area radiation monitors are provided.

Chemical seals are provided on the instrument sensing lines of process piping, which may contain highly radioactive solids, to reduce the servicing time required to keep the lines free of solids. Instruments and sensing line connections are located slightly above the pipe mid-plane wherever practical to minimize radioactive crud or gas buildup.

H. Valves

To minimize personnel exposures from valve operations, motor-operated, air-operated, or other remotely actuated valves are used where justified by the activity levels and frequency of use. Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries.

In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided such that personnel need not enter a high radiation area for valve operation.

When equipment in high radiation areas is operated infrequently, only manual valves associated with the safe operation of the equipment are provided with remote-manual operators or reach rods. All other valve operations are performed with equipment in the shutdown mode.

For valves located in radiation areas, provisions are made to drain the adjacent radioactive components when maintenance is required. To the extent practicable, valves are not located at piping low points.

Valves in the containment that are expected to exhibit stem leakage are provided with leak-off connections, piped to the reactor coolant drain tank (reactor coolant drain tank or a reduced packing configuration with the valve stem leak-off line capped).

Valves for clean, non-radioactive systems are separated from radioactive sources and are located in readily accessible areas.

For most large valves in lines carrying radioactive fluids, a double set of packing with a lantern ring is provided. A stuffing box with a leak-off connection that is piped to a drain header is also provided. Hermetically sealed (packless) valves are used on those systems where essentially no leakage can be tolerated such as hazardous gas supply lines or gaseous radioactive waste systems. Diaphragm valves are not used for modulating the flow, but are only used for the purpose of isolation. Valves 2 inches and under in diameter, except for modulating valves, located in piping normally carrying highly radioactive fluids are hermetically sealed to preclude radioactive releases to the environment. Valves which have fully engineered packing system with appropriate assembly of parts are also used on where the application of hermetically sealed valves is not practical. Valves greater than 2 inches in diameter and all modulating valves, regardless of size, have fully engineered packing packages, with live loading and/or leak off connections if necessary to reduce the potential for radioactive liquid or gas leakage. Live-loaded packing is used for valves that include cyclic service conditions and critical valves that cannot be retorqued during plant operation.

Check valves will be used only where necessary. Check valves shall be properly located and oriented within the piping system. The type and size of valve selected shall be compatible with the system requirements to minimize flow-induced disk flutter related wear damage.

Manually operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are equipped with reach rods extending through or over the valve gallery wall.

Personnel do not enter the valve gallery during spent resin or cartridge transfer operations. The valve gallery shield walls are designed to minimize personnel exposure during the maintenance of components within or adjacent to the gallery and to protect personnel who remotely operate the valves.

Relief valves are located in an associated equipment compartment or valve gallery. Check valves are located in the equipment compartment or associated valve gallery unless they are the locking type requiring manipulation during normal operation. In this case, check valves are treated as normal manual valves.

I. Piping

The piping in pipe chases is designed for the lifetime of the unit. Wherever radioactive piping is routed through areas where routine maintenance is required, pipe chases are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection or maintenance requirements. Butt welds are used to the fullest extent possible in radwaste piping utilized for the transport of spent resins or slurries. Piping containing radioactive material is routed to minimize radiation exposure to the unit personnel. Provisions are made in radioactive systems for flushing the piping with sufficient water to reduce crud buildup. Welds are made smooth, as much as possible, to prevent crud traps from forming.

Connections between piping, fittings, flanges and valves shall be welded or flanged only. O-ring type pipe caps, O-ring face seal fittings and threaded joints are not used for radioactive piping.

J. Floor Drains

Floor drains and properly sloped floors are provided for each room or cubicle containing serviceable components containing radioactive liquids. When practicable, shielded pipe chases are used for radioactive pipes. If a radioactive drain line must pass through a plant area requiring personnel access, shielding is provided, as necessary, to ensure that radiation levels are consistent with the required access.

K. Heating, Ventilation, and Air-Conditioning

The HVAC system design facilitates the replacement of the filter elements.

L. Sample Stations

Proper shielding and ventilation are provided at the local sample stations to maintain radiation zoning in proximate areas and minimize personnel exposure during sampling. The use of concrete containing fly ash is minimized in the counting room and laboratory areas.

M. Clean Services

Whenever possible, clean services and equipment such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipeways.

12.3.1.1.2 Common Facility and Layout Designs for As Low As Reasonably Achievable

This subsection describes the design features utilized for the standard type plant process and layout situations. The features used in conjunction with the general equipment described in Subsection 12.3.1.1.1 are discussed in the following paragraphs.

A. Valve Galleries

Valve galleries are provided with shielded entrances for personnel protection. Floor drains are provided to recover radioactive leakage. To facilitate decontamination in the valve galleries, concrete surfaces are covered with a smooth surface coating that allows easy decontamination.

B. Piping

Pipes carrying radioactive materials are routed through controlled access areas properly zoned for that level of activity. Each piping run is individually analyzed to determine the potential radioactivity level and surface dose rate. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways are provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Equipment compartments are used as pipeways only for those pipes associated with equipment in the compartment.

Radioactive and non-radioactive piping are separated to minimize personnel exposure, when possible and practical. Should maintenance be required, provisions are made to isolate and drain radioactive piping and associated equipment.

Potentially radioactive piping is located in appropriately zoned and restricted areas. Process piping is monitored to ensure that access is controlled to limit exposure (Section 12.5).

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. In radioactive systems, the use of non-removable backing rings in the piping joints is prohibited. Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal mid-plane of the main pipe.

Piping which carries resin slurries or evaporator bottoms is run vertically as much as possible. Horizontal runs carrying spent resin are sloped toward the spent resin tanks. Large radius bends are utilized instead of elbows. Where sloped lines or large radius bends are impractical, adequate flush and drain capability is provided to prevent flow blockage and minimize crud traps.

C. Penetrations

To minimize radiation streaming through penetrations, as many penetrations as practicable are located with an offset between the source and the accessible areas. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. If these two methods are not used, alternate means are employed, such as labyrinths or grouting the area around the penetration.

D. Contamination Control

Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.

Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All-welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at the joints. The valves in some radioactive systems are provided with leak-off connections piped directly to the collection system.

The decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of decontaminable paints and suitable smooth-surface coatings to the concrete floors and walls.

Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant. In addition, radioactive and potentially radioactive drainage is separated from non-radioactive drainage.

In controlled access areas where contamination is expected, radiation-monitoring equipment is provided (Subsection 12.3.4). Those systems that become highly radioactive, such as the spent resin lines in the radwaste system, are provided with flush and drain connections.

The role of the ventilation systems in minimizing the spread of airborne contamination is discussed in Subsection 12.3.3.

E. Equipment Layout

In those systems where process equipment is a major radiation source (such as fuel pit cleanup, coolant, boric acid recycle, chemical waste, and miscellaneous waste), pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. In general, control panels are located in low radiation zones.

Major components such as tanks, demineralizers, and filters in radioactive systems are isolated in individual shielded compartments as far as practical. Thicknesses of the concrete walls that enclose these components are tabulated in Table 12.3-1.

Labyrinth entranceway shields or shielding doors are provided for each compartment from which radiation could stream or scatter to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as demineralizer or heat exchangers and tanks in the primary coolant system), shielded compartments with elevated accessible by ladder/stairs or completely enclosed shielded compartments with hatch openings or removable concrete block walls are used. For instance, the removable blocks have been installed in a wall of the equipment room such as residual heat removal pump room, charging pump room, etc. Provision is made on some major plant components for the removal of these components to lower radiation zones for maintenance.

Equipment in non-radioactive systems that require lubrication is located in lower radiation areas.

Wherever practicable, tube-type extensions are used to lubricate equipment in radiation areas to reduce exposure during maintenance.

Exposure from routine in-plant inspection is controlled by locating, whenever possible, inspection points in properly shielded, low-background radiation areas.

Radioactive and non-radioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of non-radioactive systems. For radioactive systems, emphasis is placed on adequate space and the ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is provided. For example, a remotely operated device is provided for ISI of the reactor vessel. When this is not practicable, written procedures are used which reduce the total occupancy time in the radiation field. In addition, access to high radiation areas is under the direct supervision of the unit health physics personnel.

F. Field Run Piping

Field run piping will be minimized wherever possible.

Fabrication isometrics of radioactive process piping are reviewed by the plant architect/engineer to ensure that adequate shielding is provided.

G. Decontamination Facilities

A hot shower room is provided in the personnel contamination monitoring area in the Access Building at 3’-7” level for onsite decontamination of personnel. A first aid room is also provided in the same general area, next to the health physics room. An equipment decontamination station is placed in the hot machine shop at the basement level (-26'-4") of the Access Building. Service water is provided to various areas throughout the plant and is available for decontamination of instruments and small equipment items. The decontamination water from these facilities is drained to the floor drain sump and is forwarded to one of the waste holdup tanks for processing.

H. Lighting

Adequate illumination levels shall be provided in radiation areas for performing actions required during normal, shutdown, maintenance, and emergency conditions as described in Subsection 9.5.3.

Extended service lamps shall be utilized in high radiation areas to reduce the exposure of station personnel who service the lamps. Wherever possible, design features that permit servicing of the lamps from lower radiation areas shall be implemented.

In order to reduce the potential of station personnel exposure, an adequate emergency lighting system will be provided in high radiation areas to permit prompt egress if the station lighting system fails.

12.3.1.2 Radiation Zoning and Access Control

12.3.1.2.1 Normal Conditions

12.3.1.2.1.1 Radiation Zoning

Access to areas inside the plant structures and plant yard areas is regulated and controlled by radiation zoning and access control (Section 12.5). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding.

During plant operation, personnel normally access to radiation-controlled areas through designated access control points determined by the health physics staff.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the requirements of 10 CFR 20 (Reference 12.3-2). Each room, corridor, and pipe-way of every plant building is evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning.

The radiation zone categories employed and their descriptions are tabulated in Table 12.3-2. The radiation zones for each plant area under normal operation/shutdown conditions are shown in Figure 12.3-1. In Figure 12.3-1, the radiation zones for site is indicated on a typical plant arrangement plan. The COL Applicant is to provide the one that is shown on the site-specific plan.

Radiation zones shown in the figures are based upon conservative design data. Actual in-plant zones and control of personnel access will be based upon surveys conducted by the health physics staff, as described in Section 12.5.

12.3.1.2.1.2 Access Control

Ingress or egress of plant operating personnel to radiation controlled areas is controlled by the plant health physics staff to ensure that radiation levels and exposures are within the limits required in 10 CFR 20 (Reference 12.3-2).

Any area having a radiation level that could result in an individual receiving a dose equivalent in excess of 5 mrem/h at 12” from the radiation source or from any surface that the radiation penetrates will be posted “Caution, Radiation Area.”

Any area having a radiation level that could result in an individual receiving a dose equivalent in excess of 100 mrem/h at 12” from the radiation source or from any surface that the radiation penetrates will be barricaded and posted “Caution, High Radiation Area” or “Danger, High Radiation Area.”

High radiation areas and enclosures from which radiation emanates are provided with locked or alarmed barriers.

For individual high radiation areas accessible to personnel with radiation levels that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device. During periods when access to a high radiation area is required, positive control is exercised over each individual entry.

Any area having a radiation level that could result in an individual receiving an absorbed dose in excess of 500 rad in 1 hour at 1 meter from the radiation source or from any surface that the radiation penetrates will be posted “Grave Danger, Very High Radiation Area.” Measures taken to control access to very high radiation areas will meet the guidance of RG 8.38 (Reference 12.3-3). To the extent practicable, the measured radiation level and the location of the source are posted at the entry to any radiation area or high radiation area.

Posting of radiation signs, control of personnel access, and use of alarms and locks are in accordance with the requirements of 10 CFR 20.1601 (Reference 12.3-4) and 10 CFR 20.1902 (Reference 12.3-5).

Access control of the US-APWR is illustrated in Figure 12.3-1.

Entry into high radiation areas is controlled by the Technical Specifications.

12.3.1.2.1.3 Access Control for Personnel and Materials

1. Access is limited to the radiologically controlled areas (RCAs) as follows:

a. Access into and out of the RCA is limited to pre-assigned persons and only if access is necessary.

b. Access to the RCA is controlled at the access control point in the Access Building. However, in the case of fuel and large equipment, access control is conducted at the equipment entrances of the Reactor Building (R/B).

2. The principles of access control are incorporated as follows:

a. There is only one entrance and exit for an RCA.

b. Persons who enter an RCA are required to carry dosimeters such as digital dosimeters with alarms.

c. Persons who enter the RCA may be required to wear specified clothes, commensurate with the hazard as required by the Radiation Protection Program. If individuals enter a contaminated area, they are required to wear appropriate protective clothing, as needed, and precautions are taken to prevent internal exposure.

d. Persons who leave an RCA go through a required check for surface contamination with radiation monitors.

e. Entry and exit from an RCA is verified and recorded by Radiation Protection documents.

3. Regulations to be followed in the RCA:

a. At the entrance to the RCA, personnel are required to carry dosimeters, and may be required to wear specified clothes. At the same time, attention is paid to avoid carrying in unnecessary items.

b. In the RCA, eating, drinking, and smoking are prohibited.

c. If an abnormal or potentially abnormal condition is detected, persons are required to report immediately to the designated areas and are required to follow any instruction issued from these areas.

d. At the exit from the RCA, personnel are checked for radioactive materials on body surfaces with a radiation monitor, and they are required to report to radiation control personnel and follow instructions if any contamination is found.

e. When carrying material out of an RCA, it is verified by measurement that the external radiation and the surface contamination of the material does not exceed the value associated with the controlled area.

f. Removal of radioactive materials from the RCA for transport must meet packaging and transportation regulations regarding radiation dose rates and allowable contamination values.

12.3.1.2.1.4 Work Control

In principle, work in an RCA is carried out in order to keep the dose to the workers ALARA. Details of work control and allowed radiation exposure are described in the COLA Radiation Protection Program in Section 12.5.

12.3.1.2.2 Accident Conditions

In accordance with the guidance in Section II.B.2 of NUREG-0737 (Reference 12.3-6), a radiation and shielding design review has been performed to identify vital areas and equipment. Areas that may require occupancy to permit an operator to aid in the long-term recovery from an accident are considered vital. Vital areas include the Main Control Room (MCR), Technical Support Center (TSC), post-accident sampling system, radiochemistry laboratory (sample analysis), and hot counting room.

A general plant arrangement drawing with the location of vital areas is shown in Figure 12.3-2. Radiation levels are determined from the post-accident sources given in Section 12.2.1.3. The resulting post-accident radiation zones are shown on Figures 12.3-3 through 12.3-6.

Projected dose rates and cumulative mission doses for tasks performed within the vital areas at various times after an accident are given in Table 12.3-3. In this mission dose evaluation, it is assumed that workers use respiratory protection devices, thus only direct dose is considered. Alternatively, the dose calculation for the MCR personnel under accident conditions does not assume the use of respiratory protection devices, and exposure due to airborne activity is considered. The total mission doses for the post accident sampling activity and MCR personnel are given in Table 12.3-3 (Sheet 4 of 4).

Table 3D-2 shows the equipment which can be repaired, replaced, or recalibrated after 2 weeks from the accident, if required. Some of this equipment is installed in a location that remains under harsh radiation conditions; therefore, mission doses associated with the maintenance of this equipment are estimated. Figure 12.3-11 shows the equipment location and the associated personnel access routes. Table 12.3-9 summarizes the projected dose rates and mission doses for the equipment maintenance activities. Note that these doses are conservatively based on the estimates for 1 week after the accident even though the actual maintenance activity will not occur until after 2 weeks.

The US-APWR is designed to ensure the capability to achieve cold shutdown without subjecting personnel to excessive radiation exposure. This capability is further described in Chapter 7, Section 7.4. Radiation protection design features and access controls are described in Sections 12.3 and 12.5. In the event that entry is desired into areas where excessive radiation exposures may occur, due consideration is given to the dose rates defined on Figures 12.3-3 through 12.3-6 and Table 12.3-3, and appropriate time limits for presence in the area are imposed.

12.3.1.3 Minimization of Contamination and Radioactive Waste Generation

This section describes the US-APWR design features and operational programs to address the requirements of 10 CFR 20.1406 (Reference 12.3-29) and RG 4.21 (Reference 12.3-30). These design features, working in conjunction with operational programs, minimize environmental contamination and the generation of radioactive waste, maintain occupational doses ALARA, and facilitate the eventual decommissioning of the facility.

12.3.1.3.1 Design Consideration

The requirements of Regulatory Position C.1 through C.4 of Regulatory Guide 4.21 (Reference 12.3-30) can be met by addressing the following design objectives.

- Objective 1 – Minimize leaks and spills and provide containment in areas where such events might occur.

- Objective 2 – Provide adequate leak detection capability to provide prompt detection of leakage for any structure, system or component that has the potential for leakage.

- Objective 3 – Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of environment.

- Objective 4 – Reduce the need to decontaminate equipment and structures by decreasing the probability of any released, reducing any amounts released, and decreasing the spread of the contaminant from the source.

- Objective 5 – Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.

- Objective 6 – Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).

Subsection 12.1.2.1 describes the US-APWR design features used to maintain personnel exposures ALARA and reduce the plant source term. Some of these design features also address several RG 4.21 (Reference 12.3-30) design objectives and ultimately help minimize contamination, reduce the generation of radioactive waste, and facilitate decommissioning. Additionally, many of the US-APWR structures, systems, and components (SSCs) incorporate specific design features designed to meet the requirements of 10 CFR 20.1406 (Reference 12.3-29) and RG 4.21 (Reference 12.3-30). These design features are captured in Table 12.3-8 “Design Features for Minimizing Contamination and Generation of Radioactive Waste”. The table also provides cross-references to the applicable DCD subsections that describe how these SSCs address these design objectives to provide compliance with the requirements of 10 CFR 20.1406 (Reference 12.3-29).

12.3.1.3.2 Operational/Programmatic Considerations

Operational programs and procedures that address the requirements of 10 CFR 20.1406 (Reference 12.3-29) are necessary adjuncts to the design features discussed in the Subsection 12.3.1.3.1. The operational and post-construction requirements of Regulatory Position C.1 through C.4 of RG 4.21 (Reference 12.3-30) can be met by addressing the following operational/programmatic objectives.

- Periodically review operational practices to ensure that, operating procedures are revised to reflect the installation of new or modified equipment, personnel qualification and training are kept current, and facility personnel are following the operational procedures.
- Facilitate decommissioning by maintenance of records relating to facility design and construction, facility design changes, site conditions before and after construction, onsite waste disposal and contamination and results of radiological surveys.
- Develop a conceptual site model (based on site characterization and facility design and construction) which will aid in the understanding of the interface with environmental systems and the features that will control the movement of contamination in the environment.
- Evaluate the final site configuration after construction to assist in preventing the migration of radionuclides offsite via unmonitored pathways.
- Establish and perform an onsite contamination monitoring program for the potential pathways from the release sources to the receptor points.

COL action items 12.1(6), 12.1(7), 12.1(8) and 12.3(10) are incorporated to capture the above operational and programmatic objectives.

12.3.2 Shielding

The bases for the nuclear radiation shielding and the shielding configurations are discussed in this section.

12.3.2.1 Design Objectives

The objective of the plant radiation shielding is to reduce personnel and population exposures, in conjunction with a program of controlled personnel access to, and occupancy of, radiation areas to levels that are within the requirements of 10 CFR 50 (Reference 12.3-7) and are ALARA within the dose standards of and requirements of 10 CFR 20 (Reference 12.3-2).

Shielding and equipment layout and design are considered in ensuring that exposures are kept ALARA during anticipated personnel activities in areas of the plant containing radioactive materials, utilizing the design recommendations in accordance with the guidance in RG 8.8, Paragraph C.2 (Reference 12.3-1), where practicable.

Three plant conditions are considered in the nuclear radiation shielding design:

- Normal, full-power operation
- Shutdown conditions
- Emergency operations (for required access to safety-related equipment)

The shielding design objectives for the plant during normal operation (including anticipated operational occurrences), for shutdown operations, and for emergency operations are as follows:

- To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and individuals at and beyond the site boundary are ALARA and in accordance with the requirements of 10 CFR 20 (Reference 12.3-2).
- To ensure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.
- To reduce potential equipment neutron activation and to mitigate the possibility of radiation damage to materials.
- To provide sufficient shielding for the control room so that for DBAs, the direct dose plus the inhalation dose (calculated in Chapter 15, Subsection 15.6.5.5) will not exceed the limits in accordance with the requirements of 10 CFR 50, Appendix A (Reference 12.3-8) “General Design Criterion (GDC) 19.”

12.3.2.2 General Shielding Design

Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area shown in Figure 12.3-1. Changes to radiation zones for shutdown conditions are also shown on Figure 12.3-1 with the changes indicated in parentheses.

The general locations of the plant areas and equipment discussed in this subsection are shown in the general arrangement drawings of Chapter 1, Section 1.2. Design criteria for penetrations are consistent with the guidance of RG 8.8, C.2 (Reference 12.3-1) and are discussed in Subsection 12.3.1.1.2.

The material used for most of the plant shielding is ordinary concrete. The specification of concrete followed RG 1.69(Reference 12.3-9). The concrete specification used for a shielding design has a bulk density of approximately 140 pound per cubic foot and a chemistry composition of Type04 that are recommended in ANSI/ANS 6.4-2006.

Whenever poured concrete has been replaced by concrete blocks, the design ensures protection on an equivalent shielding basis as determined by the density of the concrete block selected.

12.3.2.2.1 Outer Shielding Design

During reactor operation, the outer shielding protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The outer shielding is integral with the containment vessel and consists of a pre-stressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat reinforced concrete foundation slab. The inside surface of the structure is lined with carbon steel. The outer shielding, the reactor vessel, and the secondary shielding reduce the radiation levels outside the outer shielding to less than 0.25 mrem/h from sources inside the containment. The wall is of a minimum thickness of 4’-4” and the dome is of a minimum thickness of 3’-8”.

For DBAs, the outer shielding and the MCR shielding reduce the plant radiation intensities from fission products inside the containment to acceptable emergency levels, in accordance with the requirements of GDC 19 for the MCR (see Subsection 12.3.2.2.7).

Where personnel locks, equipment hatches, or penetrations pass through the containment wall, additional shielding is provided to attenuate radiation to the required level defined by the outside radiation zone during normal operation and shutdown and to acceptable emergency levels in accordance with the requirements of GDC 19 during DBAs.

12.3.2.2.2 Containment Vessel Interior Shielding Design

During reactor operation, many areas inside the containment are over Zone VI.

The main sources of radiation are the reactor vessel and the primary loop components, consisting of the SGs, pressurizer, RCPs, and associated piping. The reactor vessel is shielded by the primary shield and by the secondary shield, which also surrounds other primary loop components. Air cooling is provided to prevent overheating, dehydration, and degradation of the shielding and structural properties of the primary shield.

The primary shield is a large plate of steel reinforced concrete that surrounds the reactor vessel and extends upward from the containment floor. The minimum concrete thickness of the primary shield is 9’-2”. The primary shield meets the following objectives:

- In conjunction with the secondary shield, reduction of the radiation level from sources within the reactor vessel and the RCS, thus allowing a limited access to the containment during normal, full-power operation.

- Minimization of neutron streaming to the containment vessel free volume by incorporating a labyrinth style gap between the reactor vessel and the primary shield wall in the structural design.

- Limitation of the radiation level from sources within the reactor vessel after shutdown, by utilizing remote inspection through penetrations, thus limiting access to the RCS equipment.

- Minimization of neutron activation of component and structural materials.

The secondary shield consists of a steel reinforced concrete plate that surrounds the RCS equipment, including piping, pumps, pressurizer, and SGs. This shield protects personnel from the direct gamma ray radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma ray radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation. The minimum thickness of the secondary shield walls are 4'.

Components of the letdown portion of the CVCS in the containment are located in shielded compartments that are normally over Zone VI, restricted access areas. Shielding is provided for each piece of equipment in the letdown system, consistent with its postulated maximum activity (Subsection 12.2.1) and with the access and zoning requirements of the adjacent areas. This equipment includes the regenerative heat exchanger, the excess letdown heat exchanger, the letdown heat exchanger, and the letdown lines.

After shutdown, the containment is accessible for limited periods-of-time and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range widely, depending on the location inside the containment (excluding the refueling cavity). These dose rates result from residual fission surveyed from residual fission products, and neutron activation products (components and corrosion products) in the RCS to establish allowable working periods.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding is provided for areas surrounding the refueling cavity and the containment racks. The shielding ensures that radiation levels remain below zone levels specified for the adjacent areas. The water provides the shielding over the spent fuel assemblies during fuel handling. A more detailed description of containment racks used to temporarily store up to six fuel assemblies is provided in Section 9.1.

12.3.2.2.3 Reactor Building Shielding Design

During normal operations, the major components in the reactor building that contain radioactivity are the RHR, and charging systems. Under accident conditions, the RHR, containment spray and injection systems will contain high levels of radioactivity because these systems will be used following an accident. Charging systems will not be used following an accident, but will be expected to continue to contain same radioactive material as contained during normal operation. Shielding is provided for each piece of equipment consistent with its postulated maximum activity (Section 12.2 of this chapter) and with the access and zoning requirements of the adjacent areas (see Figure 12.3-1).

Depending on the equipment in the compartments, the radiation zones under normal conditions will vary from Zone IV through Zone X. Corridors are generally shielded to allow Zone III access. Operator areas for valve compartments are generally Zone IV for access. Under accident conditions, the radiation levels will be considerably higher (see Subsection 12.3.1.2.2).

Removable sections of block shield walls and concrete plugs are used as necessary for equipment maintenance. For instance, the removable blocks have been installed in a wall of the equipment room such as residual heat removal pump room, charging pump room, etc. Permanent or temporary shielding is used between equipment in compartments with more than one piece of equipment to permit access for maintenance. Where necessary, labyrinth entrances with provisions for adequate ingress and egress for equipment maintenance and inspection are provided and are designed to be consistent with the access and zoning requirements of the adjacent areas.

Following a reactor shutdown, the RHRS pumps and heat exchangers are in operation to remove heat from the RCS. The radiation levels near this equipment will temporarily reach Zone VII levels due to corrosion and fission products in the reactor water. Shielding is provided to attenuate radiation from the RHR equipment during shutdown-cooling operations to levels consistent with the radiation zoning requirements of the adjacent areas. The shielding around the RHR equipment is a minimum of 2.5’ of concrete.

12.3.2.2.4 Fuel Handling Area Shielding Design

The concrete shield walls surrounding the spent fuel cask loading and storage area and the shield walls surrounding the fuel transfer and storage areas are of sufficient thickness to limit the radiation levels outside of the shield walls in all accessible areas to Zone II. The building external walls are sufficient to shield the external plant areas to Zone I.

All spent fuel removal and transfer operations are performed under borated water to provide radiation protection and to maintain sub-criticality conditions. The total water depth above the active fuel to the minimum water level in the spent fuel pit is approximately 29’. The dose rate with 29’ of water is significantly less than the Zone III criteria of 2.5 mrem/h. The minimum water depth above the active fuel during fuel handling is 11’-1” in the refueling cavity and 11’-1” in the fuel transfer canal and spent fuel pit. This depth of water limits the dose at the water surface to less than 2.5 mrem/h for an assembly in a vertical position.

The walls of the fuel transfer canal and SFP walls are a minimum thickness of 7’-1”-thick concrete and supplement the shielding provided by the water and limit the maximum radiation dose in working areas to less than 2.5 mrem/h.

The SFPCS (Chapter 9, Section 9.1) shielding is based on the maximum activity discussed in Subsection 12.2.1 and the access and zoning requirements of the adjacent areas. Equipment in the SFPCS that is shielded includes the SFPCS heat exchangers, pumps, and piping.

12.3.2.2.5 Auxiliary Building Shielding Design

During normal operations, the major components in the A/B with potentially high radioactivity are those in the CVCS, SGBDS, boron recycle, GWMS, LWMS, and SWMS. Shielding is provided for each piece of equipment consistent with its postulated maximum activity (Section 12.2 of this chapter) and with the access and zoning requirements of the adjacent areas (see Figure 12.3-1).

Depending on the equipment in the compartments, the radiation zones vary from Zone IV through Zone X. Corridors are generally shielded to allow Zone III access, and operator areas for valve compartments are generally Zone IV for access.

Removable sections of block shield walls and concrete plugs are utilized as necessary for equipment maintenance and spent filter cartridge replacement. Permanent or temporary shielding is used between equipment in compartments with more than one piece of equipment to permit access for maintenance. Where necessary, labyrinth entrances with provisions for adequate ingress and egress for equipment maintenance and inspection are provided and are designed to be consistent with the access and zoning requirements of the adjacent areas.

12.3.2.2.6 Turbine Building Shielding Design

Radiation shielding is not required for any process equipment located in the Turbine Building.

12.3.2.2.7 Control Room Shielding Design

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the control room. The design basis LOCA is described in Chapter 15, Subsection 15.6.5.5.

Consideration is given to the shielding provided by the containment structure.

Shielding, combined with other engineered safety features, is provided to permit access to and occupancy of the control room following a postulated LOCA, so that radiation doses are limited to 5 rem total effective dose equivalent from all contributing modes of exposure for the duration of the accident, in accordance with GDC 19 (Reference 12.3-8).

The contribution from direct radiation from airborne fission products inside the containment to personnel doses inside the control room following a postulated LOCA is shown in Chapter 15, Table 15.6.5-16. The shielding of the control room ensures compliance in accordance with GDC 19 (Reference 12.3-8).

The parameters used in the demonstration of control room habitability, in addition to those contained in RG 1.183 (Reference 12.3-10) are tabulated in Chapter 15, Table 15.6.5-5. Control room ventilation system parameters are provided in Chapter 6, Section 6.4.2.2. Figure 12.3-7 provides an isometric view of the control room shielding.

12.3.2.2.8 Spent Fuel Transfer Canal and Tube Shielding Design

Radiation streaming from the spent fuel transfer tube in the seismic gap between the containment wall and the internal containment structure and in the seismic gap between the containment wall and the fuel handling building is shielded by the labyrinth structure gap and the shock absorber shielding to maintain the radiation zone limits for normal operation in accessible areas near the seismic gap. Therefore, there is no unshielded portion of the spent fuel transfer tube during the refueling operation. The labyrinth for the fuel transfer tube is shown in Figure 12.3-8.

Administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube is to discussed by the COL Applicant.

12.3.2.2.9 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for all plant buildings containing radiation sources so that the radiation levels at the outside surfaces of the buildings are maintained below Zone I levels. The plant yard areas that are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed-off from areas accessible to the public. The RWSAT and PMWTS are outside storage tanks that have a contact dose rate greater than 0.25 mrem/h. These tanks are located inside of the tank house enclosure next to the A/B. The primary purpose of the tank house is to protect the tanks from the elements (rainfall, snow, direct sun, etc.) and thus prevent erroneous actuation of the leakage detection system. The dose rate outside of the tank house is less than 0.25 mrem/h. Figure 12.3-12 shows the layout of the tank house enclosure including the RWSAT and the PMWTs.

12.3.2.3 Shielding Calculation Methods

The shielding thicknesses provided to ensure compliance with the plant radiation zoning and to minimize the plant personnel exposure are based on the maximum equipment activities described in Chapter 11, Section 11.1, and Section 12.2.

The thickness of each shield wall surrounding the radioactive equipment is determined by approximating, as closely as possible, the actual geometry, and the physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard references (Reference 12.3-11, 12.3-12).

The geometric model assumed for shielding evaluation of tanks, heat exchangers, demineralizers, evaporators, and the containment is a finite cylindrical volume with maximum source volume capacity. Filters are assumed to be a finite cylindrical annulus in a cylindrical shield. Shielding evaluation of radioactivity in piping uses a cylindrical volume. In cases where radioactive materials are deposited on surfaces, such as pipes, sources are modeled as an annular cylindrical surface source. For spent fuels, the geometric model is a rectangular parallel pipe.

Shielding that is only for gamma ray attenuation is designed using codes that use the point kernel method (buildup factor, exponential attenuation, and geometry factor). The industry accepted program, MICROSHIELD (Reference 12.3-11), is the code used for these calculations.

Shielding for neutron fields and mixed gamma ray/neutron fields is designed using the industry-accepted computer codes ANISN, DORT (Reference 12.3-13), and MCNP (Reference 12.3-14), as necessary. ANISN and DORT are multi-group, discrete ordinates transport codes that solve the Boltzmann transport equation for neutrons and gamma rays. Using a finite-difference technique, ANISN and DORT allow for general anisotropic scattering (i.e., an Lth order Legendre expansion of the scattering cross-sections). The Monte Carlo N-Particle Transport Code (MCNP) is used for more complicated geometries such as penetrations.

Shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area.

Shielding requirements in each plant area are evaluated at the point of maximum radiation dose through any wall. In addition, for shielding design purposes, the concrete density of 140 pounds per cubic feet was assumed. Therefore, the actual anticipated radiation level in each plant area is less than this maximum dose and, consequently, less than the radiation zone upper limit.

Where shielded entryways to compartments containing high-radiation sources are necessary, labyrinths are designed using the gamma ray scattering code GGG-GP (Reference 12.3-15). The labyrinths are constructed so that the scattered dose rate, plus the transmitted dose rate through the shield wall from all contributing sources, is below the upper limit of the radiation zone specified for each plant area.

12.3.3 Ventilation

The plant HVAC systems are designed to provide a suitable environment for personnel and equipment during normal operation, during anticipated events of moderate frequency, and during certain infrequent events. Parts of the plant HVAC systems perform safety-related functions.

12.3.3.1 Design Objectives

The plant HVAC systems for normal plant operation, anticipated events of moderate frequency, and certain infrequent events are designed to meet the requirements of 10 CFR 20 (Reference 12.3-2) and 10 CFR 50 (Reference 12.3-7).

12.3.3.2 Design Criteria

The design criteria for the plant HVAC systems include the following.

- During normal operation, anticipated events of moderate frequency, and certain infrequent events, the average and maximum airborne radioactivity levels to which the plant personnel are exposed in radiation controlled areas of the plant are ALARA and within the requirements specified in 10 CFR 20 (Reference 12.3-2).

The average and maximum airborne radioactivity levels outside the radiation controlled areas of the plant during normal operation, events of moderate frequency, and certain infrequent events are ALARA and within the requirements of 10 CFR 20, Appendix B, Table II (Reference 12.3-16).

- During normal operations, anticipated events of moderate frequency, and certain infrequent events, the dose from concentrations of airborne radioactive material in Low Population Area unrestricted areas beyond the site boundary is ALARA and within the requirements specified in 10 CFR 20.1301 (Reference 12.3-17) and 10 CFR 50, Appendix I (Reference 12.3-18).

- The requirements of 10 CFR 20, Appendix B (Reference 12.3-16) is satisfied in the control room following the DBAs described in Chapter 15, Subsection 15.6.5.5.

- The dose to control room personnel shall not exceed the limits specified in GDC 19 of Appendix A to 10 CFR 50 (Reference 12.3-8) following the DBAs described in Chapter 15, Subsection 15.6.5.5.

12.3.3.3 Design Features

To accomplish the design objectives and to conform to the design criteria, the following design guidelines are employed wherever practicable.

- Guidelines to minimize airborne radioactivity:

  Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.

  Equipment vents and drains are piped directly to a collection device connected to the collection system. This is to prevent any contaminated fluid from flowing across the floor to a floor drain.

  Welded piping systems are employed on systems containing highly radioactive fluids to the maximum extent practicable.

  Suitable coatings are applied to the concrete floors and walls of potentially contaminated areas to facilitate decontamination.

  Diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated.

  The design of the equipment incorporates features that minimize the spread of radioactivity during maintenance operations.

  Ventilation openings, in areas where flooding might occur, are located so that water entry in not possible.

HEPA filters are specified not to fail for at least 20 inches wg differential pressure across them. Ventilation systems containing HEPA filters have fans with static pressure capacities well below 20 inches wg.

- Guidelines to control airborne radioactivity:

  The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.

  In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than is supplied to that area. This provides control of the release of potentially radioactive airborne materials from the area.

  Consideration is given to the possible disruption of normal airflow patterns by maintenance operations, and provisions are made in the design to prevent adverse airflow direction.

  The air cleaning system's design, maintenance, and testing criteria are discussed in detail (in accordance with the guidance in to RGs 1.52 [Reference 12.3-19] and 1.140 [Reference 12.3-20]) in Chapter 6, Section 6.4 and Subsection 6.5.1, and Chapter 9, Section 9.4. An illustrative example of an air cleaning system design is given in Subsection 12.3.3.5 of this chapter.

  Air being discharged from potentially contaminated areas in the containment is passed through high-efficiency particulate air (HEPA) filters to remove particulates. Means are provided to isolate the affected areas in the containment, fuel handling area, and A/B upon an indication of contamination. This minimizes the discharge of contaminants to the environment and in-plant exposures.

  Means are provided to isolate the control room to minimize in-leakage of contaminated air to protect personnel.

  Suitable containment isolation valves are installed in accordance with GDC 54 and 56 of 10 CFR 50, Appendix A (Reference 12.3-8), including valve controls, to ensure that containment integrity is maintained. See the additional discussion in Chapter 6, Subsection 6.2.4.

  Redundant seismic Category 1 systems and/or components are provided for portions of the ventilation system that serve areas required for the safe shutdown of the reactor plant. Included are the plant control room and selected engineered safety feature equipment rooms.

- Guidelines to minimize personnel exposure from HVAC equipment:

  The guidelines of RG 8.8 (Reference 12.3-1) are used, as practicable, in the design of the plant ventilation systems.

  Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. Filter-adsorber unit conformance, with the recommendations of RGs 1.52 and 1.140 (Reference

12.3-19, 12.3-20) for access and service requirements is summarized in Chapter 6, Section 6.4 and Chapter 9, Section 9.4.

Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.

Ventilating air is re-circulated only in areas outside the RCA. Exhaust from potentially contaminated areas in the RCA is filtered and then discharged.

- Access and service of ventilation systems in potentially radioactive areas is provided by component location to minimize operator exposure during maintenance, inspection, and testing.

- The outside air supply units and building exhaust system components are enclosed in ventilation equipment rooms. These equipment rooms are accessible to the operators. Workspace is provided around each unit for anticipated maintenance, testing, and inspection.

12.3.3.4 Design Description

The ventilation systems serving the following structures are considered to be potentially radioactive and are discussed in detail in Chapter 6, Subsections 6.5.1, and in Chapter 9, Section 9.4.

- Containment (see Chapter 6, Subsections 6.5.1.)

  A small amount of radioactive material becomes mixed with the ventilation air (as noble gases and iodine in the reactor coolant) and is partially released into the containment when reactor coolant leaks occur in the containment.

  In addition, Argon-40 in the air inside the reactor containment is converted to Argon-41 due to neutron activation between the reactor vessel and the primary shield.

  Air is discharged by the containment purge system, as necessary, when workers enter the reactor containment during reactor shutdown and in other cases.

  The air is released from the stack, and the concentration of radioactive materials is monitored by the stack gas monitors. Before release, the concentration of radioactive materials is verified by the containment gas and particulate monitors and particles are removed by the containment exhaust filter unit.

  During reactor operation, the pressure inside the containment is kept constant through operation of the containment low-volume purge. This air is treated through HEPA and charcoal filters and is vented from the stack after monitoring the radioactivity with the exhaust gas monitor.

- R/B (see Chapter 9, Subsection 9.4.3)

A small amount of radioactivity exists in the ventilation air, due to noble gases and iodine in the reactor coolant, and is partially released into the air when a reactor coolant leak occurs in the A/B.

The R/B ventilation air is released from the stack, and the concentration of radioactivity is monitored with the vent stack radiation gas monitor.

When the plant is at cold shutdown for periodic inspections etc., the quantity of noble gases transferred to the ventilation air in the R/B and A/B in the course of fuel handling and repair of equipment can be negligible. However, this air is assumed to contain I-131

- Fuel handling area (see Chapter 9, Subsection 9.4.2)

Although the control room is considered a non-radioactive area, radiation protection is provided to ensure habitability (see Chapter 6, Section 6.4 and Chapter 9, Subsection 9.4.1).

Other structures (e.g., pump intake structures, the administrative building) contain no potential source of radioactivity and are not addressed in this chapter.

12.3.3.5 Air Filtration Units

The guidance and recommendations of RGs 1.52 and 1.140 (Reference 12.3-19, 12.3-20) concerning maintenance and in-place testing provisions for atmospheric cleanup systems, air filtration, and adsorption units have been used as a references in the design of the various ventilation systems. The extent to which RGs 1.52 and 1.140 (Reference 12.3-19, 12.3-20) have been followed is discussed in Chapter 6, Section 6.4 and Chapter 9, Section 9.4.

Figure 12.3-9 shows the typical layout of an air handling unit.

Provisions specifically included to minimize personnel exposures and to facilitate maintenance or in-place testing operations are as follows:

- In normal operation, filters and absorbers in ventilation of containment are in operation. The level of radioactivity and dose rate developed during normal operation will not to be high and will not exceed the dose limit for the access area. Therefore, no shielding is provided and occupational exposure for the filter elements replacement will not be high.

- During accident and post accident conditions, filters and absorbers in the air purification systems for the MCR and the TSC, and filters in annulus air purification system, are in operation. The level of radioactivity and dose rate developed during accident and post accident conditions of these filters and absorbers will increase, therefore these filters and absorbers are in shielded cubicles. It will not be necessary for workers to handle filter units immediately after a DBA. Exposures can be minimized by allowing the short-lived isotopes to decay before changing the filter and absorber.

- Active components of the atmospheric cleanup systems are designed to permit ready removal.
- Access to active components is direct from working platforms to simplify element handling. Ample space is provided on the platforms for safe personnel movement during the replacement of components, including the use of necessary material handling facilities and in-place testing operations.
- If a HEPA filter has more than three stacked banks, where each bank has 2' by 2' in size, a platform is to be facilitated for access to the upper bank easily.
- The filters are designed with replaceable units that are clamped in place against compression seals. The filter housing is designed, tested, and proven airtight with bulkhead-type doors that are closed against compression seals.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The radiation monitoring system consists of the following:

- Area Radiation Monitoring System (ARMS)
- Airborne Radioactivity Monitoring System
- Process and Effluent Radiation Monitoring System
- Sampling system
- Post-Accident Monitoring (PAM) radiation monitors

The process and effluent radiation monitoring system and sampling systems are described in Chapter 11, Section 11.5.

The PAM are described in Chapter 7, Section 7.5. The portable dose rate and activity monitoring instruments are Type E PAM.

The ARMS and Airborne Radioactivity Monitoring System supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and assure compliance with the personnel radiation protection requirements of 10 CFR 20 (Reference 12.3-2), 10 CFR 50 (Reference 12.3-7), 10 CFR 70 (Reference 12.3-21), and the guidelines of RGs 1.21 (Reference 12.3-22), 1.97 (Reference 12.3-23), 8.2 (Reference 12.3-24), and 8.8 (Reference 12.3-1), ANSI N13.1-1999 (Reference 12.3-25), and IEEE 497-2002 (Reference 12.3-28).

The design of the spent fuel storage racks and containment racks precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 50.68 (Reference 12.3-26), are not needed.

The ARMS are in conformance with ANSI/ANS HPSSC-6.8.1 (Reference 12.3-27).

The use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737, are to be described by the COL Applicant.

12.3.4.1 Area Radiation Monitoring System

12.3.4.1.1 Design Objectives

The design objectives of the ARMS during normal operating plant conditions and anticipated operational occurrences are as follows:

- To record radiation levels in specific areas of the plant
- To warn of uncontrolled or inadvertent movement of radioactive material in the plant
- To provide local and remote indication of ambient gamma radiation and local and remote alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area
- To furnish information for making radiation surveys

By meeting the above objectives, the ARMS aids health physics personnel in keeping radiation exposures ALARA.

The design objectives of the ARMS during postulated accidents are as follows:

- To provide the capability to alarm and initiate a containment ventilation isolation signal in the event of a LOCA or abnormally high radiation inside the containment (monitors RMS-RE-091, RMS-RE-092, RMS-RE-093, and RMS-RE-094). In Modes 1 through 4, four trains of radiation monitors are required to ensure radiation-monitoring instrumentation necessary to initiate the containment ventilation isolation.
- To provide long-term post-accident monitoring (Chapter 7, Section 7.5)

12.3.4.1.2 Criteria for Location of Area Radiation Monitors

The locations of the area radiation monitors are shown in Figure 12.3-1.

Considerations for area radiation monitor locations include:

- Areas which are normally accessible, and where changes in plant conditions can cause significant increases in personnel exposure rate above that expected for the area

- Areas which are normally accessible or occasionally accessible where a significant increase in exposure rate resulting from operational transients or maintenance activities may occur

- The containment area where the level of radioactivity needs to be monitored to detect the presence of fission products during a DBA

- Area monitor detectors are located such that inadvertent shielding by structural materials is minimized

- In the selection of area monitors, consideration is given to the range of temperature, pressure and humidity of the areas where the detectors or electronics are located

The ARMS provides a continuous, direct indication or recording of radiation levels in the control room and raises alarms locally and in the control room when radiation levels exceed the set values.

The fixed area monitors are installed in the following locations to warn occupants of the area of a deteriorated radiological condition:

a. MCR

b. Inside of the containment

c. Radio Chemical Lab

d. SFP area

e. Nuclear sampling room

f. Inside of the containment (near the air lock)

g. Inside of the containment (near the ICIS)

h. Waste Management System (WMS) area

i. TSC

For areas with positive access control features, such as normally locked doors, or areas where a radiological hazard only exists during specific work activities, a fixed ARM is not required. Instead, a portable ARM is installed to warn occupants of a deteriorated radiological condition. Portable ARMs are utilized in the following locations:

j. Refueling platform

k. Residual heat removal pump and heat exchanger areas

l. Hot machine shop

m. HVAC filter area

n. Cask handling area

o. Equipment decontamination area

p. Safe shutdown panel area

12.3.4.1.3 General System Description

The ARMSs are located at selected locations throughout the plant to detect, indicate, and store information through their associated data processing module on the radiation levels and, if necessary, annunciate abnormal radiation conditions.

Each monitor is composed of the requisite number of channels, with a channel consisting of a radiation detector and check source, except for monitors RMS-RE-091, RMS-RE-092, RMS-RE-093, RMS-RE-094.

The detectors for all area monitors are sensitive to gamma rays. If exposed to radiation in excess of full-scale indication, the area monitors indicate that the full-scale reading has been exceeded and remains at the full-scale value. If the radiation field causing the overload condition is removed, the system returns to its normal operating condition unless the detector has failed. An administrative procedure (positioning the check source) is initiated to ensure that the radiation-monitoring equipment has not been damaged. All channels are indicated and annunciated in the control room (with the exception of the containment area monitors) and indicated and alarmed to the personnel near the detector location. The WMS Area Monitor alarm also synchronously alerts the personnel in the A/B control room. Monitors RMS-RE-091, RMS-RE-092, RMS-RE-093, RMS-RE-094, which are safety-related, Class-1E, are also indicated at the safety-related display console.

12.3.4.1.4 Data Processing Module and Display Console

A description of these components is given in Chapter 11, Subsection 11.5.2.1.

12.3.4.1.5 Local Annunciation

All area monitors, except those in the containment, have local annunciation consisting of an audible alarm and a warning light at the local readout.

12.3.4.1.6 Power Supplies

Each channel-installed rack is provided with a common power supply. However, the signal process unit of each channel is designed such that a failure in that channel does not affect any other channel.

Monitors that are identified as safety-related are redundant and are supplied with power from the Class 1E 120 V buses. Monitors that are identified as non-safety-related are supplied from the Non-Class 1E 120 V buses that is backed up by the Non-Class 1E Alternate AC Gas Turbine Generator.

12.3.4.1.7 Redundancy, Diversity, and Independence

Monitors designated as safety-related are part of the safety-related portion of the Protection and Safety Monitoring System and are designed for redundancy, diversity, and independence in accordance with the Institute of Electrical and Electronics Engineers (IEEE) Standards (Reference 12.3-28).

12.3.4.1.8 Area Radiation Monitor Description

Table 12.3-6 gives the conditions of service for the area radiation monitors. A brief description of each area radiation monitor's function is given below.

- MCR Area Radiation Monitor

  To continuously indicate the radiation levels in the control room. An alarm signal warns the control room personnel of a deteriorated radiological condition inside the MCR.

- Containment High Range Area Radiation Monitors

  To continuously indicate the radiation levels inside the containment. During refueling operations, a high radiation alarm indicates a fuel drop accident. During power operations, an alarm indicates a possible LOCA and isolates the containment ventilation system. These four radiation monitors are installed widely separated inside the containment.

- SFP Area Radiation Monitor

  To continuously indicate the radiation levels inside the fuel handling area. An alarm signal warns the occupants of the fuel handling area of a deteriorated radiological condition.

- Radio Chemical Lab. Area Radiation Monitor

  To continuously indicate the radiation levels in the radio chemical lab. An alarm signal warns the occupants of the radio chemical laboratory of a deteriorated radiological condition.

- Nuclear Sampling Room Area Radiation Monitor

  To continuously indicate the radiation levels in the nuclear sampling room. An alarm signal warns the occupants of the nuclear sampling room of a deteriorated radiological condition.

- Containment Air Lock Area Radiation Monitor

  To continuously indicate the radiation levels in the containment access hatch and establish the radiological habitability prior to entry. An alarm signal warns the occupants of the containment access hatch of a deteriorated radiological condition.

- ICIS Area Radiation Monitor

  To continuously indicate the radiation levels in the ICIS area. An alarm warns the occupants of the ICIS area of a deteriorated radiological condition.

- TSC Area Radiation Monitor

  To continuously indicate the radiation levels in the TSC. An alarm signal warns the TSC personnel of a deteriorated radiological condition inside the TSC.

- WMS Area Radiation Monitor

  To continuously indicate the radiation levels in the WMS. An alarm signal warns the occupants of the WMS of a deteriorated radiological condition.

12.3.4.1.9 Range and Alarm Setpoints

The range and control function of the ARMS is given in Table 12.3-4.

Alarm setpoints are controlled by plant procedures and the offsite dose calculation manual, where appropriate. And the methodology to determine the calibration interval and setpoints for the Area Radiation Monitors and Process and effluent Radiation Monitors are described in the Subsection 7.2.2.7. The calibration procedures are described in the Subsection 13.5.2.2.

Radiation zones for the normal operation of US-APWR are described in Table 12.3-2.

The following monitors are located in radiation Zones I or II:

- MCR Area Radiation Monitor
- Radio Chemical Lab. Area Radiation Monitor
- TSC Area Radiation Monitor

The MCR Area Radiation Monitor has a greater sensitivity than the other area radiation monitors since it is located in a Zone I radiation area and the reactor operators are present. The installed containment high radiation monitor has sufficient instrumentation range to measure radiation levels during an accident.

Each area radiation monitor has two alarm setpoints – intermediate and high. If a monitor has a control function (i.e., Containment High Range Area Radiation Monitor), the control function is triggered coincidentally with the high alarm setpoint. An intermediate alarm gives both a visual and audible indication in the control room (or alternate radwaste control room in the case of the ARMS) and near the detector where the radiation level has reached the intermediate setpoint. A high alarm gives both a visual and audible indication in the control room and near the detector where the high alarm setpoint has been reached.

12.3.4.2 Airborne Radioactivity Monitoring Systems

The Airborne Radioactivity Monitoring System is provided for monitoring in-plant airborne radioactivity levels.

12.3.4.2.1 Design Objectives

The design objectives of the Airborne Radioactivity Monitoring System during normal operating plant conditions and anticipated operational occurrences are as follows:

- To measure the airborne radioactivity in the HVAC exhaust ducts of the air exhausted from cubicles
- To warn of an abnormal release of radioactive material from cubicles

12.3.4.2.2 Criteria for Location of Airborne Radioactivity Monitors

Considerations for airborne monitor sampling points are HVAC exhaust ducts that are installed in the Radioactive Controlled Area.

The Airborne Radioactivity Monitors are sampled at locations where airborne radioactivity may normally exist. If the gas is detected, the existence of Iodine or other radioactive materials are to be determined. The Airborne Radioactivity Monitors are installed in the following areas:

- Fuel Handling Area
- Annulus and Safeguard Area
- R/B
- A/B
- Sample and Lab Area

All of these areas are RCA.

The sampling points of the airborne radioactivity monitors are shown in the Figure 12.3-10. The detailed flow diagram of HVAC system includes the airborne radioactivity monitors are shown in Figure 9.4.3-1.

12.3.4.2.3 General System Description

The system description of the airborne radioactivity monitors is the same as process gas monitors (see Chapter 11, Subsection 11.5.2.1).

12.3.4.2.4 Data Processing Module and Display Console

A description of these components is given in Chapter 11, Section 11.5.

12.3.4.2.5 Local Annunciation

All airborne radiation monitors have local annunciation consisting of an audible alarm and a warning light at the local readout.

12.3.4.2.6 Power Supplies

Each channel-installed rack is provided with a common power supply. However, the signal process unit of each channel is designed such that a failure in that channel does not affect any other channel.

Monitors that are identified as non-safety-related are supplied from the Non-Class 1E 120 V buses that are backed up by the Non-Class 1E Alternate AC Gas Turbine Generator.

12.3.4.2.7 Redundancy, Diversity, and Independence

All airborne radioactivity monitors have no redundancy, diversity, or independence.

12.3.4.2.8 Airborne Radioactivity Monitors Component Description

12.3.4.2.8.1 Fuel Handling Area HVAC Radiation Gas Monitor (RMS-RE-049)

The Fuel Handling Area HVAC Radiation Gas monitor is a β scintillation-type monitor. The detection range and other details are summarized in Table 12.3-5.

This monitor measures the radiation level in the exhaust air contained in the HVAC duct. An alarm in the MCR is initiated when the radiation level in the ventilation system exceeds the setpoint.

The monitor is not safety-related.

12.3.4.2.8.2 Annulus and Safeguard Area HVAC Radiation Gas Monitor (RMS-RE-046)

The Annulus and Safeguard Area HVAC Radiation Gas monitor is a β scintillation-type monitor. The detection range and other details are summarized in Table 12.3-5.

This monitor measures the radiation level in the exhaust air contained in the HVAC duct. An alarm in the MCR is initiated when the radiation level in the ventilation system exceeds the setpoint.

The monitor is not safety-related.

12.3.4.2.8.3 Reactor Building HVAC Radiation Gas Monitor (RMS-RE-048A)

The R/B HVAC Radiation Gas monitor is a β scintillation-type monitor. The detection range and other details are summarized in Table 12.3-5.

This monitor measures the radiation level in the exhaust air contained in the HVAC duct. An alarm in the MCR is initiated when the radiation level in the ventilation system exceeds the setpoint.

The monitor is not safety-related.

12.3.4.2.8.4 Auxiliary Building HVAC Radiation Gas Monitor (RMS-RE-048B)

The A/B HVAC Radiation Gas monitor is a β scintillation-type monitor. The detection range and other details are summarized in Table 12.3-5.

This monitor measures the radiation level in the exhaust air contained in the HVAC duct. An alarm in the MCR is initiated when the radiation level in the ventilation system exceeds the setpoint.

The monitor is not safety-related.

12.3.4.2.8.5 Sample and Lab Area HVAC Radiation Gas Monitor (RMS-RE-048C)

The Sample and Lab Area HVAC Radiation Gas monitor is a β scintillation-type monitor. The detection range and other details of this monitor are summarized in Table 12.3-5.

This monitor measures the radiation level in the exhaust air contained in the HVAC duct. An alarm in the MCR is initiated when the radiation level in the ventilation system exceeds the setpoint.

The monitor is not safety-related.

12.3.4.2.9 Range and Alarm Setpoints

The range and control function of the airborne monitor are given in Table 12.3-5.

Alarm setpoints are controlled by plant procedures.

The alarm setpoint is determined by providing a margin relative to the normal radiation levels, as the main purpose of the alarm is to detect any abnormal situation.

The airborne radioactivity monitoring system is capable of detecting 10 DAC-hours of particulate and iodine radioactivity from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel, taking into account dilution in the ventilation system.

12.3.5 Dose Assessment

Refer to Section 12.4 for the discussion of the dose assessment.

12.3.6 Combined License Information

COL 12.3(1) *The COL Applicant shall describe portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident, in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.*

COL 12.3(2) *Deleted.*

COL 12.3(3) *Deleted.*

COL 12.3(4) *The COL Applicant is to provide the site radiation zones that is shown on the site-specific plant arrangement plan.*

COL 12.3(5) *The COL Applicant is to discuss the administrative control of the fuel transfer tube inspection and the access control of the area near the seismic gap below the fuel transfer tube.*

COL 12.3(6) *If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about the radiation protection aspects of the system and to indicate how the system is consistent with the guidance in SRP Section 12.3-12.4, RG 1.206 C.I.12.3.2 and RG 1.69.*

COL 12.3(7) *If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to provide information about prevention and detection of contamination of the environment and minimization of decommissioning costs and to explain how the system meets the requirements of 10 CFR 20.1406 and RG 4.21.*

COL 12.3(8) *If the COL Applicant adopts the Mobile Liquid Waste Processing System, the COL Applicant is to confirm the radiation zone(s) where the system is installed in and to revise Figure 12.3-1, if necessary.*

COL 12.3(9) *In order to ensure that the B.A. evaporator room does not become a VHRA during the end of cycle, the COL Applicant is to stipulate a need for routine surveillance in the Radiation Protection Program. In the event that the routine surveillance shows an increase in dose level, the COL Applicant must provide an appropriate strategy to sufficiently reduce the dose rate below the criteria for a VHRA.*

COL 12.3(10) *The COL Applicant will address the site-specific design features, operational, post-construction objectives, and conceptual site model guidance of Regulatory Guide 4.21.*

12.3.7 References

12.3-1 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable. RG 8.8, Paragraph C.2, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 1978.

12.3-2 "Standards for Protection against Radiation," Energy. Title 10 Code of Federal Regulations Part 20, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-3 Control of Access to High and Very High Radiation Areas of Nuclear Plants. RG 8.38, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, May 2006.

12.3-4 'Control of Access to High Radiation Areas,' "Standards for Protection against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20.1601, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-5 'Posting Requirements,' "Standards for Protection against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20.1902, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-6 "Plant Shielding," Section II.B.2, Clarification of TMI Action Plan Requirements. NUREG-0737, U.S. Nuclear Regulatory Commission, Washington, DC, January 1980.

12.3-7 "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-8 'General Design Criteria for Nuclear Power Plants,' "Domestic Licensing of Production and Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50, Appendix A, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-9 Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants. RG 1.69, Rev.1, U.S. Nuclear Regulatory Commission, Washington, DC, May 2009.

12.3-10 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. RG 1.183, Rev. 0, U.S. Nuclear Regulatory Commission, Washington, DC, July 2000.

12.3.11 MicroShield User's Manual Version 7. Grove Software, Inc. 2006

12.3-12 RSICC Computer Code Collection CCC-371, ORIGEN 2.2: Isotope Generation and Depletion Code - Matrix Exponential Method.

12.3-13 RSICC Computer Code Collection CCC-650, DOORS3.2: One, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System

12.3-14 RSICC Computer Code Collection CCC-710, MCNP5: Monte Carlo N-Particle Transport Code System

12.3-15 RSICC Computer Code Collection CCC-564, GGG-GP: Kernel Integration Code System Multigroup Gamma-Ray Scattering Using the GP Buildup Factor.

12.3-16 "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," Energy. Title 10, Code of Federal Regulations, Part 20, Appendix B, U.S. Nuclear Regulatory Commission, Washington DC.

12.3-17 'Dose Limits for Individual Members of the Public,' "Standards for Protection Against Radiation," Energy. Title 10, Code of Federal Regulations, Part 20.1301, U.S. Nuclear Regulatory Commission, Washington DC.

12.3-18 'Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,' "Domestic Licensing Of Production And Utilization Facilities," Energy. Title 10, Code of Federal Regulations, Part 50, Appendix I, U.S. Nuclear Regulatory Commission, Washington DC.

12.3-19 Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants. Regulatory Guide 1.52, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.

12.3-20 Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants. Regulatory Guide 1.140, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.

12.3-21 "Domestic Licensing of Special Nuclear Material," Energy. Title 10, Code of Federal Regulations, Part 70, U.S. Nuclear Regulatory Commission, Washington DC.

12.3-22 Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants. RG 1.21, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1974.

12.3-23 Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants. Regulatory Guide 1.97, Rev. 4, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-24 Guide for Administrative Practices in Radiation Monitoring. Regulatory Guide 8.2, Rev. 0, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973.

12.3-25 Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of nuclear facilities. American National Standard Institute N13.1-1999.

12.3-26 "Criticality accident requirements." Energy Title 10 Code of Federal Regulations Part 50.68, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-27 Location and Design Criteria for Area Radiation Monitoring Systems for Light-Water Nuclear Reactors. American National Standard Institute/American Nuclear Society (ANS) HPSSC-6.8.1-1981.

12.3-28 IEEE Standards, IEEE 603-1991, IEEE 497-2002

12.3-29 "Minimization of Contamination." Energy. Title 10 Code of Federal Regulations, Part 20.1406, U.S. Nuclear Regulatory Commission, Washington, DC.

12.3-30 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. RG 4.21, Rev.0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

Table 12.3-1 Thicknesses of Concrete walls that enclose the major components (Sheet 1 of 4)

Elevation	Room Name	North	East	South	West	Floor	Ceiling
Inside C/V							
	Primary Shielding	9'-2"					
	Secondary Shielding	4'-0"					
	Outer Shielding	4'-4"					
50'-2''	Letdown heat exchanger Room	4'-0"	2'-0"	2'-0"	2'-0"	1'-1.5"	1'-9"
50'-2''	Regenerative heat exchanger Room	1'-2"	2'-0"	4'-0"	2'-0"	1'-1.5"	2'-0"
50'-2"	Excess letdown heat exchanger Room	2'-0"	2'-0"	4'-0"	2'-0"	1'-1.5"	1'-9"
Reactor Building							
-26'-4"	A-Charging pump Room	3'-8"	5'-11"	4'-0"	2'-0"	Ground	1'-10"
-26'-4"	B-Charging pump Room	3'-8"	2'-0"	4'-0"	4'-0"	Ground	1'-10"
-26'-4"	A-Containment spray/residual heat removal pump Room	3'-4"	3'-4"	2'-6"	-	Ground	2'-6"
-26'-4"	B-Containment spray/residual heat removal pump Room	2'-6"	3'-4"	3'-8"	-	Ground	2'-6"
-26'-4"	C-Containment spray/residual heat removal pump Room	2'-6"	-	3'-8"	3'-4"	Ground	2'-6"
-26'-4"	D-Containment spray/residual heat removal pump Room	3'-4"	-	2'-6"	3'-4"	Ground	2'-6"
-8'-7"	A-Piping Area	3'-4"	3'-4"	-	-	2'-6"	3'-10" 1)
-8'-7"	B-Piping Area	-	3'-4"	3'-8"	-	2'-6"	3'-10" 1)
-8'-7"	C-Piping Area	-	-	3'-8"	3'-4"	2'-6"	3'-10" 1)
-8'-7"	D-Piping Area	3'-4"	-	-	3'-4"	2'-6"	3'-10" 1)
-8'-7"	Piping Area 2)	3'-8"	4'-0"	4'-0"	4'-0"	3'-2"	4'-0"
-8'-7"	Piping Area 3)	3'-8"	4'-0"	3'-2"	3'-2"	1'-10"	3'-2"

1) Face to Safeguard component area AHU room

2) East side

3) West side

Table 12.3-1 Thicknesses of Concrete walls that enclose the major components (Sheet 2 of 4)

Elevation	Room Name	North	East	South	West	Floor	Ceiling
Reactor Building							
3'-7"	A-Containment spray/residual heat removal heat exchanger Room	3'-4"	3'-6"	4'-0"	3'-10"	3'-4"	3'-4"
3'-7"	B-Containment spray/residual heat removal heat exchanger Room	4'-0"	3'-6"	3'-4"	3'-10"	3'-4"	3'-4"
3'-7"	C-Containment spray/residual heat removal heat exchanger Room	4'-0"	3'-10"	3'-4"	3'-6"	3'-4"	3'-4"
3'-7"	D-Containment spray/residual heat removal heat exchanger Room	3'-4"	3'-10"	4'-0"	4'-0"	3'-4"	3'-4"
3'-7"	Piping Area 1)	3'-10"	3'-10"	3'-4"	3'-10"	3'-4"	3'-10"
3'-7"	Piping Area 2)	3'-4"	3'-10"	3'-2"	3'-10"	3'-4"	3'-10"
3'-7"	Piping Area 3)	3'-4"	3'-10"	3'-2"	3'-10"	3'-4"	3'-10"
3'-7"	Piping Area 4)	3'-10"	3'-10"	3'-4"	3'-10"	3'-4"	3'-4" 5)
25'-3"	Volume control tank Room	5'-9"	7'-1"	5'-3" 6)	5'-3"	2'-4"	5'-3"
25'-3"	Piping Area 7)	3'-0"	7'-1"	3'-10"	3'-10"	2'-4"	3'-10"
25'-3"	Main Control Room	3'-4"	3'-4"	5'-0"	3'-4"	3'-4"	3'-4"
25'-3"	A-Piping Penetration Area	3'-10"	3'-10"	4'-2"	3'-8"	3'-4" 8)	3'-10"
25'-3"	B-Piping Penetration Area	4'-2"	3'-10"	3'-10"	2'-8"	3'-4" 8)	3'-10"
25'-3"	C-Piping Penetration Area	4'-2"	2'-8"	3'-10"	3'-10"	3'-4" 8)	3'-10"
25'-3"	D-Piping Penetration Area	3'-4"	-	4'-2"	3'-10"	3'-4" 8)	3'-10"
30'-1"	Spent Fuel Pit	7'-9"	5'-11"	7'-1"	7'-1"	10'-6"	-

1) Adjacent to A-Containment spray/residual heat removal heat exchanger Room
2) Adjacent to B-Containment spray/residual heat removal heat exchanger Room
3) Adjacent to C-Containment spray/residual heat removal heat exchanger Room
4) Adjacent to D-Containment spray/residual heat removal heat exchanger Room
5) Face to Piping Area
6) Face to area of zone III
7) Adjacent to Volume control tank Room
8) Face to Containment spray/residual heat removal heat exchanger Room

Table 12.3-1 Thicknesses of Concrete walls that enclose the major components (Sheet 3 of 4)

Elevation	Room Name	North	East	South	West	Floor	Ceiling
Auxiliary Building							
-26'-4"	A-Holdup tank Room	3'-4"	3'-8"	4'-2"	4'-2"	Ground	4'-2"
-26'-4"	B-Holdup tank Room	2'-0"	3'-8"	3'-4"	4'-2"	Ground	4'-2"
-26'-4"	C-Holdup tank Room	3'-8" 1)	3'-8"	2'-0"	4'-2"	Ground	4'-2"
-26'-4"	A-Waste holdup tank Room	2'-0"	3'-4"	3'-4"	3'-4"	Ground	2'-8"
-26'-4"	B-Waste holdup tank Room	3'-4"	3'-4"	2'-0"	3'-4"	Ground	2'-8"
-26'-4"	C-Waste holdup tank Room	2'-0"	2'-6" 1)	3'-4"	3'-4"	Ground	3'-0"
-26'-4"	D-Waste holdup tank Room	3'-4"	2'-0"	2'-0"	3'-4"	Ground	3'-0"
-26'-4"	Charcoal bed Room (A)	2'-6"	2'-6"	4'-4" 1)	2'-6"	Ground	4'-1" 2)
-26'-4"	Charcoal bed Room (B)	2'-6"	2'-6"	4'-4" 1)	2'-6"	Ground	4'-1" 2)
-26'-4"	Waste gas surge tank Room (A)	3'-4"	3'-4"	4'-4" 1)	2'-8"	Ground	2'-8"
-26'-4"	Waste gas surge tank Room (B)	3'-4"	2'-8"	4'-4" 1)	3'-4"	Ground	2'-8"
-26'-4"	A-Spent resin storage tank Room	3'-2"	3'-2"	3'-4"	4'-6"	Ground	3'-2"
-26'-4"	B-Spent resin storage tank Room	4'-6"	3'-2"	3'-2"	4'-6"	Ground	3'-2"
-26'-4"	Valve Area 3)	2'-2"	2'-2"	2'-6"	2'-6"	Ground	3'-0" 8)
-26'-4"	Valve Area 4)	3'-4"	3'-4"	2'-8"	3'-4"	Ground	2'-8"
-26'-4"	Valve Area 5)	3'-6"	3'-6"	3'-4"	3'-2"	Ground	3'-2"
-8'-7"	Piping Area 5)	3'-6"	3'-6"	4'-6" 6)	3'-4"	3'-6"	3'-6"
-8'-7"	Piping Area 7)	3'-4"	3'-4"	3'-4"	3'-8"	2'-8"	2'-8"

1) Face to area of Zone III
2) Face to Waste Mobile System Room
3) Adjacent to Charcoal bed Room (A)
4) Adjacent to Waste gas surge tank Room
5) Adjacent to Spent resin storage tank Room
6) Face to Spent resin storage tank room
7) Adjacent to Holdup tank Room
8) Face to Truck Bay Area

Table 12.3-1 Thicknesses of Concrete walls that enclose the major components (Sheet 4 of 4)

Elevation	Room Name	North	East	South	West	Floor	Ceiling
Auxiliary Building							
3’-7”	Mixed bed demineralizer Room	3’-4”	3’-4”	3’-4”	4’-8”	3’-0”	4’-8”
3’-7”	Cation-bed demineralizer Room	3’-4”	2’-10”	3’-4”	4’-0”	3’-2”	3’-9”
3’-7”	Spent fuel pit demineralizer Room	2’-10”	2’-0”	2’-0”	3’-4”	3’-2”	3’-4”
3’-7”	Valve Area 1)	2’-10”	4’-2”	4’-2” 2)	3’-4” 3)	3’-2” 4)	4’-8”
3’-7”	A-Reactor coolant filter Room	2’-0”	2’-8”	2’-0”	2’-0”	2’-8”	2’-2”
3’-7”	B-Reactor coolant filter Room	3’-4”	2’-8”	2’-0”	2’-0”	2’-8”	2’-2”
3’-7”	A-Spent fuel pit filter Room	1’-6”	2’-2”	2’-0”	2’-0”	2’-8”	2’-2”
3’-7”	B-Spent fuel pit filter Room	1’-6”	2’-2”	1’-6”	2’-0”	2’-8”	2’-2”
3’-7”	A,B-Waste demineralizer Room	3’-4”	2’-5”	2’-5”	3’-9”	3’-3”	3’-9”
3’-7”	C,D-Waste demineralizer Room	3’-4”	3’-9” 2)	3’-4”	3’-9”	3’-0”	3’-9”
3’-7”	Valve Area 5)	2’-10”	2’-8”	3’-4”	2’-5” 6)	2’-8” 2)	3’-4”
3’-7”	Waste Mobile Systems	3’-4”	--- 7)	3’-4”	3’-4”	4’-1” 10)	4’-0”
3’-7”	B.A. evaporator feed demineralizer Room	2’-0”	2’-0”	3’-4”	3’-4”	3’-2”	3’-4”
15’-9”	Piping Area 8)	3’-4”	3’-4”	2’-8”	3’-4” 2)	2’-8”	3’-4”
15’-9”	Hold up Tank Valve Area	3’-4”	2’-8” 2)	3’-4”	3’-8”	2’-8”	3’-4”
25’-3”	Steam generator blowdown demineralizer Room	3’-4”	3’-4”	3’-4”	2’-6” 2)	3’-4”	2’-2”
25’-3”	Valve Area 9)	3’-4”	1’-6”	3’-4”	3’-4”	3’-4”	1’-10”

1) Adjacent to Mixed bed demineralizer Room
2) Face to area of Zone III
3) Face to Mixed bed demineralizer Room
4) Face to Valve Area
5) Adjacent to Waste demineralizer Room
6) Face to A,B-Waste demineralizer Room
7) Removable Shield is to be used, if necessary
8) East side of demineralizer Rooms
9) Adjacent to Steam generator blowdown demineralizer Room
10) Face to Charcoal Beds Room

Table 12.3-2 Radiation Zones

Zone	Maximum Dose Rate	Description
I	0.25 mrem/h	Controlled area, unlimited occupancy
II	1 mrem/h	Restricted area, limited occupancy
III	2.5 mrem/h	Restricted area, limited occupancy
IV	15 mrem/h	Restricted area, limited occupancy
V	100 mrem/h	Restricted area, limited occupancy
VI	1 rem/h	High radiation sources. Restricted area, limited occupancy for very short periods. Access controlled as stated in the Technical Specifications.
VII	10 rem/h	Same as Zone VI above
VIII	100 rem/h	Same as Zone VI above
IX	500 rad/h	Same as Zone VI above
X	> 500 rad/h	Very high radiation sources. Restricted area, very limited occupancy for the shortest periods. Access controlled as stated in the Technical Specifications.

NOTE:

Controlled access, unlimited occupancy areas: where entry and exit by plant employees and visitors are not under the direct supervision of the plant health physics staff. These areas can be occupied by plant personnel or visitors on an unlimited time basis with a minimum probability of health hazard from radiation exposure.

Controlled access, limited occupancy areas: Where higher radiation levels and/or radioactive contamination, which have a greater probability of radiation health hazard to individuals, can be expected. Only individuals directly involved in the operation of the plant will, in general, be allowed to enter these areas. Entry and exit are authorized and supervised by the plant health physics staff.

Occupancy: The time spent by an individual in a particular area. Occupancy is to be determined on an area by area and individual by individual basis.

Restricted access, limited occupancy areas: Where extremely high radiation levels and/or radioactive contamination is expected. Only individuals directly involved in the operation of the plant will, in general, be allowed to enter these areas. These areas are normally inaccessible with locked doors and positive control of access. Entry and exit are under the supervision of the plant health physics staff.

Table 12.3-3 Projected Dose Rates for the Vital Areas at Various times after an Accident (sheet 1 of 4)

POST ACCIDENT Vital Areas	Various Times after an Accident			
	1 hour	1 day	1 week	1 month
MCR	≦ 1.0 mrem/h	≦ 1.0 mrem/h	≦ 1.0 mrem/h	≦ 1.0 mrem/h
TSC	≦ 1.0 mrem/h	≦ 1.0 mrem/h	≦ 1.0 mrem/h	≦ 1.0 mrem/h
Radio chemical Laboratory	≦ 2.5 mrem/h	≦ 2.5 mrem/h	≦ 2.5 mrem/h	≦ 2.5 mrem/h
Hot counting room	≦ 2.5 mrem/h	≦ 2.5 mrem/h	≦ 2.5 mrem/h	≦ 2.5 mrem/h
Post accident sampling system (Liquid sampling)	≦ 1 rem/h	≦ 15 mrem/h	≦ 15 mrem/h	≦ 15 mrem/h
Post accident sampling system (Gas sampling)	≦ 1 rem/h	≦ 15 mrem/h	≦ 15 mrem/h	≦ 15 mrem/h

Table 12.3-3 Mission Dose for the Vital Areas access route after an Accident (1 hour after) (sheet 2 of 4)

Vital Area	Task description	Time when access required [h]	Max dose rate [rem/h]	Mission dose [rem]	Access route zone map No.
Main control room (MCR)	Access to MCR from AC/B for operation.	2.8E-02	1.0E-03	2.8E-05	Figure 12.3-3 Sheet 3,4,5
		7.6E-03	1.0	7.6E-03	
		Total		7.6E-03	
	Return to MCR from Radiochemical laboratory	5.0E-02	2.5E-03	1.3E-04	Figure 12.3-3 Sheet 3,4,5
		7.6E-03	1.0	7.6E-03	
		Total		7.7E-03	
Technical support center (TSC)	Access to TSC from AC/B for operation.	1.4E-02	1.0E-03	1.4E-05	Figure 12.3-3 Sheet 3,4,5
Postaccident sampling system (Liquid or gas sampling)	Access to PASS from MCR for sampling. (Sampling time is included)	3.6E-02	2.5E-03	9.1E-05	Figure 12.3-3 Sheet 3 to 8
		9.3E-02	1.0	9.3E-02	
		Total		9.3E-02	
Radiochemical laboratory & Hot counting room	Access to Radiochemical laboratory from PASS for sample analysis. Access to HOT counting room from PASS for sample counting. (Analysis and counting time are included)	8.5E-01	2.5E-03	2.1E-03	Figure 12.3-3 Sheet 3 to 8
		5.2E-02	1.0	5.2E-02	
		Total		5.4E-02	

(Note) Walk speed is usually about 13000 ft/h (4 km/h) and stairs are about 6500 ft/h (2 km/h).
Sampling time is 2 minutes and analysis time is 50 minutes.

Table 12.3-3 Mission Dose for the Vital Areas access route after an Accident (1 day to 1 month after) (sheet 3 of 4)

Vital Area	Task description	Time when access required [h]	Max dose rate [rem/h]	Mission dose [rem]	Access route zone map No.
Main control room (MCR)	Access to MCR from AC/B for operation.	2.8E-02	1.0E-03	2.8E-05	Figure 12.3-4 Sheet 3,4,5
		7.6E-03	1.5E-02	1.1E-04	
		Total		1.4E-04	
	Return to MCR from Radiochemical laboratory	5.0E-02	2.5E-03	1.3E-04	Figure 12.3-4 Sheet 3,4,5
		7.6E-03	1.5E-02	1.1E-04	
		Total		2.4E-04	
Technical support center (TSC)	Access to TSC from AC/B for operation.	1.4E-02	1.0E-03	1.4E-05	Figure 12.3-4 Sheet 3,4,5
Postaccident sampling system (Liquid or gas sampling)	Access to PASS from MCR for sampling. (Sampling time is included)	3.6E-02	2.5E-03	9.1E-05	Figure 12.3-4 Sheet 3 to 8
		9.3E-02	1.5E-02	1.4E-03	
		Total		1.5E-03	
Radiochemical laboratory & Hot counting room	Access to Radiochemical laboratory from PASS for sample analysis. Access to HOT counting room from PASS for sample counting. (Analysis and counting time are included)	8.5E-01	2.5E-03	2.1E-03	Figure 12.3-4 Sheet 3 to 8
		5.2E-02	1.5E-02	7.8E-04	
		Total		2.9E-03	

(Note) Walk speed is usually about 13000 ft/h (4 km/h) and stairs are about 6500 ft/h (2 km/h).
Sampling time is 2 minutes and analysis time is 50 minutes.

Table 12.3-3 Activity Mission Dose (sheet 4 of 4)

Activity Description	Mission Dose [rem]	Note
Post accident sampling and analysis activity (1 hour after)	1.5E-01	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 2 of 4)
Post accident sampling and analysis activity (1 day to 1 month after)	4.6E-03	Sum of the mission doses for the related tasks described in Table 12.3-3 (sheet 3 of 4)
Operation activity in the MCR (30 days after the LOCA)	4.6	Described in Table 15.6.5-16

Table 12.3-4 Area Radiation Monitors

Detector	Type	Service	Nominal Range	Safety-Related	Quantity	Control Function
RMS-RE-001	Gamma ray	MCR Area Radiation Monitor	1E-5 to 1E+0 R/h	No	1	None
RMS-RE-002	Gamma ray	Containment Air Lock Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-003	Gamma ray	Radio Chemical Lab. Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-005	Gamma ray	SFP Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-006	Gamma ray	Nuclear Sampling Room Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-007	Gamma ray	ICIS Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-008	Gamma ray	WMS Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-009	Gamma ray	TSC Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-091A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation
RMS-RE-092 A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation
RMS-RE-093 A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation
RMS-RE-094 A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation

Table 12.3-5 Airborne Radioactivity Monitors

Detector Number	Service	Type	Range $\mu Ci/cm^3$	Calibration Isotopes	Safety-Related	Quantity	Control Function
RMS-RE-049	Fuel Handling Area HVAC Radiation Gas The concentration of radioactive gas in the HVAC duct of the Fuel Handling Area	β Scint	5E-8 to 5E-4	Kr-85 Xe-133	No	1	None
RMS-RE-046	Annulus and Safeguard Area HVAC Radiation Gas The concentration of radioactive gas in the HVAC duct of the Annulus and Safeguard Area	β Scint	5E-8 to 5E-4	Kr-85 Xe-133	No	1	None
RMS-RE-048A	Reactor Building HVAC Radiation Gas The concentration of radioactive gas in the HVAC duct of the R/B Area	β Scint	5E-8 to 5E-4	Kr-85 Xe-133	No	1	None
RMS-RE-048B	Auxiliary Building HVAC Radiation Gas The concentration of radioactive gas in the HVAC duct of the A/B Area	β Scint	5E-8 to 5E-4	Kr-85 Xe-133	No	1	None
RMS-RE-048C	Sample and Lab Area HVAC Radiation Gas The concentration of radioactive gas in the HVAC duct, Sample/Lab Area	β Scint	5E-8 to 5E-4	Kr-85 Xe-133	No	1	None

Table 12.3-6 Service Conditions for the Area Radiation Monitors

Service	Temp.	Pressure	Relative Humidity	Dose Rate
MCR Area Radiation Monitor	Mild			0.25 mrem/h
Containment Air Lock Area Radiation Monitor	Mild			100 mrem/h
Radio Chemical Lab. Area Radiation Monitor	Mild			1 mrem/h
SFP Area Radiation Monitor	Mild			2.5 mrem/h
Nuclear Sampling Room Area Radiation Monitor	Mild			100 mrem/h
ICIS Area Radiation Monitor	Mild			100 mrem/h
WMS Area Radiation Monitor	Mild			100 mrem/h
TSC Area Radiation Monitor	Mild			0.25 mrem/h
Containment High Range Area Radiation Monitor	Harsh			>500 rem/h (Note)
Containment High Range Area Radiation Monitor	Harsh			>500 rem/h (Note)
Containment High Range Area Radiation Monitor	Harsh			>500 rem/h (Note)
Containment High Range Area Radiation Monitor	Harsh			>500 rem/h (Note)

(Note) Accident Condition

Table 12.3-7 Equipment Specification Limits for Cobalt Impurity Levels

Application	Maximum Mass Percent of Cobalt
Inconel and stainless steel components in fuel assembly	0.05
Inconel Tubing in Steam Generator	0.016 (Average: 0.014)
Components in region of high neutron flux such as Neutron Reflector and Lower Core Barrel	0.05
Divider Plate of Steam Generator and weld clad surfaces of Reactor Vessel, Pressurizer and Channel Head of Steam Generator	0.05
Upper Core Plate, Upper/Lower Core Support Plate and Upper Core Barrel	0.05
Main Coolant Piping,	0.15
Casings and internals of Reactor Coolant Pumps and Reactor Internals other than listed above	0.20
Bearing and hard-facing materials	Not limited (However, precipitation hardening stainless steel will be used for some valves exposed to severe depressurization conditions, and non-cobalt hard-facing material will be used for Reactor Coolant Pump.)
Auxiliary components such as valves except for listed above, piping instrumentation, tanks, and so on, including bolting materials in primary and auxiliary components	Not limited
Welding material, except where used as weld cladding	Not limited

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 1 of 76)

Fuel Storage and Handling **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	New and spent fuel storage facilities are located in the fuel handling area of the Reactor Building (R/B) which is designed to meet the Seismic Category I requirements of Regulatory Guide (RG) 1.29.	9.1.1.1 9.1.2.2.2
		The fuel storage and handling area is protected against natural phenomena. The robust concrete walls and ceiling surrounding the fuel storage and handling area are designed to withstand the loads and forces caused by wind, tornadoes, hurricanes, floods, and external missiles.	9.1.1.1 9.1.2.2.2
		The spent fuel pit is constructed of reinforced concrete lined with stainless steel plate. Similarly, the refueling canal, fuel inspection pit, and cask pit are constructed of reinforced concrete lined with stainless steel plate. The liner surface will have a 2B or higher finish, selected to minimize accumulation of corrosion and fission products, and also provide easy maintenance and decontamination. This liner surface is smooth and non-porous to avoid buildup of radioactive material.	9.1.2.2.2
		Penetrations for the drain and makeup lines are located to preclude the draining of the SFP due to a break in a line or failure of a pump to stop. The connection for the SFP pumps' suction is located below normal water level and above the level needed to provide sufficient water for shielding and for cooling of the fuel if the SFPCS is unavailable. This design feature aids in minimizing the leakages and spills (dispersion of water) from the SFP.	9.1.2.2.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 2 of 76)

Objective		System Features	DCD Reference
		The SFP is not connected to the equipment drain system to preclude unanticipated drainage.	9.1.2.2.2
		Heating, Ventilation, and Air Conditioning (HVAC) provides ventilation for the fuel handling area to provide control of the release of potentially radioactive airborne materials by maintaining exhaust airflow greater than supply airflow in this area.	9.1.1.1 9.1.2.1 9.1.2.2.2
		The SFPCS consists of one 100% capacity RWSAT (29,410 cu. ft), pumps, associated valves, piping, and instrumentation. The piping to and from the RWSAT is single-walled stainless steel that runs above ground and penetrates the building wall directly into the tank. For piping between buildings, penetration sleeves are provided to collect and direct any leakage back into the building drain for further processing. The RWSAT employs leak-tight valves to minimize leakage to the environment. This design is supplemented by operational programs, which include periodic visual inspections for piping integrity. Testing the piping segments will be included as a part of the plant routine inspections and maintenance program.	9.1.3.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The SFPCS is designed to collect system leakage; instrumentation is provided to indicate SFP water level.	9.1.3.1
		A fuel pool liner leakage collection system is provided to collect possible leakage from liner plate welds on the pit walls and floor. This system is provided with a leak detection capability and alarm. Leaked water is directed to the R/B drain sump.	9.1.2.1 9.1.2.2.2
		Two safety-related SFP water level instruments (narrow-range) are installed in the SFP. Each transmitter is interlocked with the SFP pump of that same train to avoid SFP pump cavitation and failure due to decreased SFP water level below the SFP pump suction line. Each transmitter is also interlocked with the motor operated purification line isolation valves to close on a low-low SFP water level signal. The SFP water level instruments (narrow-range) annunciate low-low water level of the SFP to the MCR and locally.	9.1.3.5.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 3 of 76)

Objective		System Features	DCD Reference
		The RWSP has a leak detection system which consists of leak detection channels that interface with the RWSP liner and are routed through the RWSP floor into standpipes between the RWSP and the PCCV. These standpipes are visually inspected during refueling outages as part of the leak monitoring operational program, in accordance with RG 4.21.	9.1.3.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 4 of 76)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	A SFP liner leakage collection system is provided to collect possible leakage from liner plate welds on the pit walls and floor. This system is provided with a leak detection capability and alarm. Leaked water is directed to the R/B drain sump.	9.1.2.1 9.1.2.2.2
		The spent fuel pit and its liner are designed to maintain their structural integrity and remain leak tight under all applicable design loads and load combinations. The walls of the SFP are an integral part of the seismic category I reactor building structure. Any leakage from liner plate welds is detected by opening the valves or caps on patrols conducted weekly. To meet the requirements of 10 CFR 20.1406, the inside of the spent fuel pit leakage collection pipes are inspected using a device such as a fiberscope approximately every refueling outage. Should materials such as accumulated boric acid residue and minerals be detected, the inside of the pipes are cleaned. The spent fuel pit leakage collection pipes are sized to allow cleaning of blockages as specified in RG 4.21.	9.1.2.2.2
		The SFP water chemistry can be checked at local sample points. If purification is required, a portion of the system flow is diverted through the SFP demineralizer and filter and returned to the pit.	9.1.3.2.2.1
		The SFPCS is designed to collect system leakage. A liner collection system to the R/B sump is provided to collect possible leakage from the SFP liner plate welds on the pit walls and floor. Leakage from the system piping is collected to the R/B sump. A leakage alarm will be installed upstream of the R/B sump for immediate detection of significant leakage levels.	9.1.3.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 5 of 76)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	The spent fuel pit purification and cooling system (SFPCS) includes the following functions: - Purifies and clarifies the SFP water - Purifies the boric acid water for the refueling water storage pit (RWSP), the reactor cavity, and the refueling water storage auxiliary tank (RWSAT) in conjunction with the Refueling Water System (RWS)	9.1.3 9.1.3.2.2.3
		Spent fuel pit demineralizers are provided with a resin-retaining screen on the backwash line, and connected to a backwash discharge line to the waste holdup tank. The screen and discharge line are designed to maintain Occupational Radiation Exposure (ORE) ALARA during resin back washing.	9.1.3.2.1.5

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 6 of 76)

Objective		System Features	DCD Reference
		The SFP water chemistry can be checked at local sample points. If purification is required, a portion of the system flow is diverted through the SFP demineralizer and filter and returned to the pit.	9.1.3.2.2.1
		The SFPCS capability is sufficient to permit the necessary operations that must be conducted in the SFP area. The SFPCS is designed to perform its purification function in accordance with the following additional criteria: - The water purification loop contains a filter vessel with a disposable cartridge filter and a mixed bed demineralizer upstream of the filter. - Local sample lines are provided at the SFP pumps discharge lines and the demineralizer outlet lines. Sampling and analysis of SFP water for gross activity and particulate concentration is conducted when the SFPCS is in continuous operation.	9.1.3.2.2.3
		Contamination spread is prevented by such means as: - Processing of water used for refueling and flushing of steel liners following spent fuel handling - Refueling equipment decontamination after use	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 7 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. Embedded and buried piping is minimized (leak chases are not considered piping). b. Process equipment items are designed with flushing capability, and to be accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces. Large tanks are designed for extended life and will not need to be replaced. If a leak develops, these tanks can be accessed for repairs in place, or can be decontaminated to remain in the cubicles until decommissioning, during which time they can be cut into smaller pieces for disposal.	—
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	Operational contamination is prevented by such means as: - Processing of water used for refueling and flushing of steel liners following spent fuel handling - Refueling equipment decontamination after use	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 8 of 76)

Water Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Essential Service Water System (ESWS)

The water in the ESWS does not normally contain radioactivity (DCD Subsection 9.2.1.2.2.6). No radioactive contamination between CCWS and ESWS occurs if the CCWS system is contaminated because the CCW plate heat exchangers are constructed to prevent intermixing of the fluids from both sides (DCD Subsection 9.2.1.2.1); therefore, the following design features are provided.

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks**	
		The CCWS is the intermediate loop between the components containing radioactive fluids and the ESWS. This arrangement prevents direct leakage of the radioactive fluid to the environment through the ESWS.	9.2.1.2.1
		The water in the ESWS does not normally contain radioactivity (Subsection 9.2.1.2.2.6). Radioactive contamination only occurs if the CCWS system is contaminated and then leaks into the ESWS via the CCW HX (Subsection 9.2.1.3). Radioactive release to the environment through the ESW is therefore minimized.	9.2.1.2.2.6, 9.2.1.3
		The other components (other than the CCW HX) cooled by the ESW are the essential chiller units that do not contain radioactive fluid.	9.2.1.2.3.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 9 of 76)

Objective		System Features	DCD Reference
		Also, the effect of long-term corrosion of the piping is mitigated by adding a corrosion inhibitor. The ESW is periodically sampled and chemicals are added, as required, during power operation.	
		Design features to provide containment	
		The CCW plate heat exchangers are constructed to prevent intermixing of the fluids from both sides so that any leakage will go to the outside of the heat exchanger except when a hole is developed in the plates --a rare event with titanium plates. Gasket failure directs leakage towards the outside of the CCW heat exchanger, hence radioactive contamination of the ESWS propagating to the UHS and ultimately to the environment is not considered credible. Any leakage from the CCW heat exchangers is collected into the nonradioactive floor drains thus ensuring that no ESWS water is released directly to the environment.	9.2.1.2.1
		Radiation monitors are provided in each discharge line of CCW HX essential service water (ESW) side. The monitors alert the operator if the leaking CCW contains radioactivity so that the operator can isolate the train of the ESW which is connected to the leaking CCW HX.	9.2.1.2.1
		Prior to any radiation leakage being detected in the ESWS, however, radiation alarms in the CCWS side would have already alerted the operators of contamination in the CCWS. The affected CCWS train is immediately isolated followed by the isolation of the aligned ESWS to prevent possible contamination of the UHS and the environment.	9.2.1.2.1
		The CCWS is designed to serve as an intermediate system between components containing radioactive fluids, which are cooled by the system, and the ESWS so as to prevent direct leakage of radioactive fluid into the environment through the ESWS.	9.2.2.1.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 10 of 76)

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Radiation monitors are provided in each discharge line of CCW HX essential service water (ESW) side. The monitors alert the operator if the leaking CCW contains radioactivity so that the operator can isolate the train of the ESW which is connected to the leaking CCW HX.	9.2.1.2.1
		If, however, radioactive leakage does occur in the CCWS, radiation monitors will alarm in the MCR to enable immediate stoppage of the CCW pump and isolation of the leaking train.	9.2.1.2.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 11 of 76)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection instrumentation is described in the above rows, and is included on all four trains of the ESWS.	-
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	The ESWS draws water from the UHS [[basin]] and returns water to the UHS after passing through the CCW HXs and the essential chiller units. The UHS is the source of water to the UHS [[basin]]. The essential chiller units do not include the radioactive fluid, and CCWS is the intermediate loop between the reactor auxiliaries and the ESWS. This arrangement minimizes direct leakage of the radioactive fluid from the ESWS to the environment. In addition, radiation monitors are provided in each discharge line of CCW HX essential service water (ESW) side. The monitors alert the operator if the leaking CCW contains radioactivity so that the operator can isolate the leaking train.	9.2.1.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 12 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. Process equipment items are accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces.	-
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning.	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 13 of 76)

Water Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Component Cooling Water System (CCWS)

The CCWS is a closed loop system. It is designed to serve as an intermediate system between components containing radioactive fluids, which are cooled by the system, and the ESW so as to prevent direct leakage of radioactive fluid into the environment through the ESW system (DCD Section 9.2.2.1.2). Radioactive contamination of CCWS occurs only if there are leakages from radioactive users into the CCWS.

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The system is designed to assure that leakage of radioactive fluid from the cooled components is held within the plant.	1.2.1.5.4.4
		Design features to minimize leaks	
		Water chemistry control of CCWS is performed by adding chemicals to the CCW surge tank to prevent long term corrosion that may degrade system performance. The CCW in the surge tank is covered with nitrogen gas to maintain water chemistry.	9.2.2.3.4
		Piping joints and connections are welded, except where flanged connections are required.	9.2.2.2.1.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 14 of 76)

Objective		System Features	DCD Reference
		Design features to provide containment Radiation monitors are located downstream of the supply headers and the signal is displayed in the MCR. When the signal exceeds the setpoint, an alarm is transmitted and the CCW surge tank vent valve is closed.	 9.2.2.5.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The radiation monitors are installed to detect the leakage of radioactive materials into the CCWS. In the event of in-leakage through a RCP thermal barrier heat exchanger, the isolation valves on the RCP thermal barrier heat exchanger CCW return lines are automatically closed by the high flow rate signal, thereby preventing CCWS contamination. In addition, isolation boundary valves between the Reactor Building and Turbine and Auxiliary Buildings can be closed if contamination is detected by radiation monitors. If leakage from a higher pressure component to the CCWS should occur, the water level of the CCW surge tank increases and an alarm is transmitted to the MCR. If the in-leakage is radioactive, the radiation monitors of the CCWS also indicate in the MCR the increased radiation level and transmit an alarm when the radiation level reaches its set point. After the leak source is identified, the leak is isolated from the CCWS.	1.2.1.5.4.4 9.2.2.2.1.5 9.2.2.3.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 15 of 76)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection instrumentation is described above.	—
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	The detection and mitigation features discussed above decrease the probability of releases. In addition, CCWS makeup sources use only radiologically “clean” water as potential CCWS makeup sources.	— 9.2.2.2.1.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 16 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. Embedded piping is kept to a minimum. b. Process equipment items are accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces.	—
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning.	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 17 of 76)

Water Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Condensate Storage Facility

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The CST is installed on a steel-reinforced concrete foundation with a concrete retaining wall (dike) surrounding the tank. The foundation and wall are coated with epoxy providing smooth surfaces to facilitate draining leakage or overflow to a sump. In addition, the concrete foundation beneath the tank is sloped towards the sump within the dike. The sump has liquid detection instrumentation and alarms for operator action to initiate the collection of samples of the liquid. If the liquid is determined to be non-contaminated it will be discharged, and if it is determined to be contaminated, it will be transferred to the Liquid Waste Management System (LWMS) for treatment. In either case, the liquid is drained to a sump within the adjacent pump house to facilitate pump-out for disposal or treatment. The CST has a painted carbon steel cover that extends from the top of the tank to slightly beyond the outer diameter of the dike in order to minimize the collection of rain and snow inside the dike. Liquid inside the dike is sampled for contamination and removed for disposal or treatment.	9.2.6.2.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 18 of 76)

Objective		System Features	DCD Reference
		The underground segment of transfer piping running between the CST and the hotwell is single-walled welded stainless steel piping in a coated trench with removable but sealed covers. The coating in the trench is Service Level II as defined in RG 1.54 Revision 1 and is subject to the graded QA provisions, selection, qualification, application, testing, maintenance, and inspection provisions of RG 1.54 and standards referenced therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D4537 using the inspection plan guidance of ASTM D 5163. The trench also has inspection manholes with drain collection basins and liquid level switches. This design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity, in compliance with the guidance of RG 4.21 and industry operating experience.	9.2.6.2.4
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Piping in a coated trench with removable but sealed covers, this design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.	9.2.6.2.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 19 of 76)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	Piping in a coated trench with removable but sealed covers, this design is supplemented by periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.	9.2.6.2.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 20 of 76)

Water Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Primary Makeup Water Tanks (PMWTs)

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks** The piping to and from the PMWT is single-walled stainless steel piping designed to run aboveground and penetrates the building wall directly into the tank. For piping between buildings, penetration sleeves are provided to collect and direct any leakages back into the building for further processing. The piping may require heat tracing to protect against freezing. The PMWTs employ non-leakage type valves such as diaphragm-type valves, or leak control valves with graphite packing for handling radioactive fluid, or leak-off connection is provided to prevent leakage to environment. Similar piping is provided for the PMWTs carrying recycle water back to the A/B. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, and visual inspections to maintain piping integrity. A discussion on minimizing radioactive contamination of the system is contained in DCD Section 12.3.1.3.	9.2.6.2.6

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 21 of 76)

Objective		System Features	DCD Reference
		The tank house for the RWSAT and the two smaller PMWTs consists of a low porosity concrete foundation and concrete retaining walls around the tanks. To prevent crosscontamination, the tanks are protected by walls and a roof to prevent infiltration of rain and other precipitation. The surface of the tank house foundation is sloped to facilitate drainage of the leakage from the tanks. The leakage water flows into the piping embedded in the concrete base mat via the funnel to the valve pit located outside the PMWTs and RWSAT watertight compartments and is equipped with a normally closed valve. The piping located at the valve pit has a leak detection instrument that sends a signal to the main control room in the instance of the accumulation of leakage. The concrete foundation, the walls, and the valve pit are coated with epoxy to facilitate easy decontamination in the event of contaminated water leakage. The epoxy coating in the tank house is Service Level II as defined in RG 1.54 (Ref 9.2.11-10) and is subject to the graded QA provisions, selection, qualification, application, testing, maintenance, and inspection provisions of RG 1.54 and standards referenced therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D 4537 (Ref. 9.2.11-11) using the inspection plan guidance of ASTM D 5163 (Ref. 9.2.11-12). Segments of drain pipe are embedded into the low porosity concrete foundation. The embedded segments are kept as short as practicable to minimize unmonitored leakage from the drain pipe.	
		CSF tanks including PMWTs and CST and their associated piping are periodically tested / inspected for leakages.	9.2.6.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 22 of 76)

Process Auxiliaries **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Process and Post-Accident Sampling Systems

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The Primary Liquid Sampling System (PLSS) is designed to collect liquid samples from the Reactor Coolant System and the auxiliary systems. These samples are transported to a common location in the sampling room of the Access Building. Limiting the number of locations at which sampling analysis is performed will minimize the spread of contamination within the plant buildings.	9.3.2.2.1
		PLSS sampling is performed via manual operation on an intermittent basis in order to minimize the time during which leaks and spills can occur. The sample lines are purged before each sample is drawn and the purged liquid is returned to the low-pressure end of its own system. This liquid is therefore contained within the sampling system and the contamination of other structures, systems, or components is minimized.	9.3.2.2.1
		Leakage from the Post Accident Sampling System (PASS) outside the containment is collected by the Reactor Building sump tank to prevent the contamination of other systems and areas.	9.3.2.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 23 of 76)

Objective		**System Features**	**DCD Reference**
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The Steam Generator Blowdown Sampling System (SGBDSS) is designed to provide detection of primary-to-secondary leakage within the steam generator tubes. The samples taken from the blowdown are monitored for radioactivity as a means to indicate tube leakage. In addition, Condenser vacuum pump exhaust is equipped with radiation monitors to close the steam generator blowdown isolation valves in the event that radiation level exceeds a pre-determined setpoint.	9.3.2.2.5 10.3.3 Figure 10.4.2-1 11.5.2.4.2
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The Primary Gaseous Sampling System (PGSS) is designed to collect representative samples from the containment atmosphere during normal operation. The analysis of these samples will be used to determine the gaseous composition of the containment and will in turn indicate the presence of radioactive material in-leakage.	9.3.2.2.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 24 of 76)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	In order to reduce the source volume exposed at the sample panel of the PLSS, the components of this system which contain radioactive liquids are located in shielded compartments including the sample coolers, isolation valves, and system piping.	9.3.2.2.1
		Samples collected by the PLSS are transferred to the sample panel within a ventilated, hooded enclosure in order to confine any leakage or spillage within this system. Any contaminated liquid from leaks or spills will be collected in the sample sink and will be drained to the Waste Holdup Tank to be treated by the Liquid Waste Management System. A similar line is included for the post-accident liquid samples. This will prevent the spread of contamination from the PLSS to other plant systems or equipment.	9.3.2.2.1 9.3.2.2.3
		Samples collected by the PGSS are contained in gaseous sample vessels which are positioned within a filtered vent hood. In addition, residual dew condensation liquid collected on these gaseous sample vessels is collected and routed to the holdup tanks to be processed by the Chemical and Volume Control System. These design features will assist in decreasing the spread of potentially contaminated process fluids from the PGSS.	9.3.2.2.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 25 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. There will be no embedded or buried piping. b. The sample vessels utilized in the PLSS for high-pressure samples are a removable design equipped with quickdisconnect couplings in order to facilitate the removal of the vessels to be brought to the radiochemical laboratory for analysis. Grab sampling points are provided for liquid sample collection as needed. These points include sample vessel connections equipped with quick-disconnect couplings in order to facilitate the removal of the vessels and transport for analysis.	9.3.2.2.1 9.3.2.2.6

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 26 of 76)

Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	The Post-Accident Sampling System (PASS) contains venting which is transferred to the HVAC System. This venting undergoes radiation monitoring. In the event that high radiation levels are detected, the venting stream is re-routed to a line with HEPA and charcoal filters. These HVAC design features will minimize the spread of contamination from the PASS.	9.3.2.2.3
		The SGBDSS is designed to perform the function of detecting high radiation levels in the steam generator blowdown which is an indication of tube leakage. In the event that high radiation is detected, a signal is sent which automatically isolates the steam generator blowdown lines and the steam generator sample lines. This isolation will minimize the spread of contamination from the site of the leakage within the steam generator system.	9.3.2.2.5
		The PLSS, PGSS, PASS and SGBDSS lines, which penetrate the containment, can be isolated through the manipulation of valves either by receipt of a containment isolation signal or by manual actuation. This will prevent the spread of contamination from the containment into any of the sampling sub-systems.	9.3.2.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 27 of 76)

Process Auxiliaries **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Chemical and Volume Control System (CVCS)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks**	
		- The CVCS employs diaphragm-type valves which minimize leaks, where applicable. For components which cannot structurally employ these types of valves, a leak-off connection is provided to prevent leakage to the atmosphere.	9.3.4.2.6.26
		- Heat exchangers are designed with corrosion-resistant materials and radioactive fluid is processed through the tube side.	12.3.1.1.1.2.F
		Design features to provide containment	
		- The CVCS piping that handles radioactive liquid is made of austenitic stainless steel.	9.3.4.2.6.29
		- The piping joints and connections are welded except where flanged connections are required for equipment removal for maintenance and hydrostatic testing.	9.3.4.2.6.29
		- Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The cubicles are equipped with a drainage system to direct any leakage and overflows to sumps with pumps to redirect flow to other tanks.	12.3.1.1.1.2.E

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 28 of 76)

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	- The drainage system is equipped with a liquid detection instrument which can provide early warning for leakage and/or overflow condition to initiate operator actions. - The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation, and provided with coating with non-porous material to prevent cross contamination. - On the shell side of heat exchangers, the return header has a radiation monitor to isolate the cooling water system, in the event leakage is detected.	12.3.1.1.1.2.E
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection system is incorporated in all cubicles in which the tanks contain radioactive fluid (refer to system features for objective #2 above). The tanks include: • Holdup Tanks • Volume Control Tank • Boric Acid Tanks	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 29 of 76)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	In addition to the features discussed in items above, the following design specifications and operational procedures are also implemented:	
		• Water chemistry is strictly monitored for primary and secondary systems. In particular, water is treated to control oxygen which is chemically scavenged to minimize potential for corrosion;	9.3.4.1.2.3
		• Stainless steel will be specified as the materials which are resistant to corrosion. Surface will be polished to facilitate easy decontamination; and	9.3.4.2.6
		• Suitable smooth-surface coatings facilitate the decontamination of potentially contaminated areas and equipment.	12.3.1.1.2.D
		• Floor drains with properly sloping floors are provided and radioactive and potentially radioactive drainage is separated from non-radioactive drainage.	12.3.1.1.2.D
		The summation of these design and operational features is designed to reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	
		Each of the demineralizers below are provided with a resin-retaining screen on the backwash line and connected to a backwash water discharge line to the holdup tank. The screen and discharge line are designed to maintain Occupational Radiation Exposure (ORE) ALARA during resin back washing.	
		• Mixed bed demineralizers	9.3.4.2.6.14
		• Cation bed demineralizer	9.3.4.2.6.15
		• Deborating demineralizer	9.3.4.2.6.16
		• Boric acid evaporator feed demineralizer	9.3.4.2.6.17

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 30 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. The design uses a drain header to direct potential leakage and spills to the reactor drain tank via a common header. This design minimizes embedded and buried piping within the building foundation slab. No other embedded piping is anticipated in the current design. b. Process equipment items are designed to be accessible for maintenance and replacement. Equipment is designed for extended life and will not need to be replaced. If leak develops, equipment can be accessed for repairs in place. Equipment can be decontaminated to remain in the cubicles until decommissioning, during which time they can be cut into smaller pieces for disposal.	—
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning.	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 31 of 76)

Process Auxiliaries **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Air Conditioning, Heating, Cooling, and Ventilation Systems

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks** - Air distribution ductwork is leak-tested in accordance with the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) technical manual "HVAC Air Duct Leakage Test Manual" and American Society of Mechanical Engineers, ASME N510, AG-1 Section SA and TA. - All non-safety related ductwork located in areas containing safety related components (Reactor Building, the Fuel Handing Area and the Power Source Building) shall be designed to seismic category II. They will prevent interaction with safety-related SSC's in these areas. - Ventilation zones, air distribution and airflow migrations are configured and arranged so that the ventilation air is drawn from the clean areas to areas of potentially greater radioactive contamination to a final filtration and exhaust systems discharging to the plant vent stack.	9.4.1.4 9.4.3.4 9.4.5.4 9.4.3.3.1 9.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 32 of 76)

Objective		System Features	DCD Reference
		Design features to provide containment - The penetration and the safeguard component areas supply and exhaust duct work are isolated in order that operation of the annulus emergency exhaust system can maintain a negative pressure and mitigate the release of airborne fission product to the atmosphere	9.4.3.1.1.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 33 of 76)

Objective		System Features	DCD Reference
		- The auxiliary building HVAC system discharge duct is isolated in order to prevent backflow of discharge air from the annulus emergency exhaust system into the auxiliary building HVAC system.	9.4.3.1.1.1
		- The ventilation systems in potentially contaminated areas maintain airflow from areas of low radioactivity to areas of potentially higher radioactivity.	9.4.3.1.2.1 12.3.3.3
		- The auxiliary building HVAC system controls exhaust fan airflow continuously and automatically at a predetermined value higher than the supply fan airflow to provide control of the release of potentially radioactive airborne materials from the controlled areas during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path.	9.4.3.1.2.1 12.3.3.3
		- Upon receipt of the ECCS actuation signal, the auxiliary building HVAC system discharge duct is automatically isolated by the equipment class 2, seismic category I isolation dampers in order to prevent backflow of discharge air from the annulus emergency exhaust system into the auxiliary building HVAC system.	9.4.3.3.1
		- Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.	12.3.3.3
		- Air cleaning systems are utilized.	12.3.3.3
		- The HVAC system design facilitates the replacement of the filter elements.	12.3.1.1.1.2.K
		- The tanks vent line is either open to the cubicle or connected directly to the ventilation system.	12.3.1.1.1.2.E

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 34 of 76)

Objective		System Features	DCD Reference
		- Ventilation openings, in areas where flooding might occur, are located so that water entry is not possible.,	12.3.3.3
		- HEPA filters are specified to withstand at least 20 inches wg differential pressure. Ventilation systems containing HEPA filters have fans with static pressure capacities well below 20 inches wg.	9.4.1.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The auxiliary building HVAC system controls exhaust fan airflow continuously and automatically at a predetermined value higher than the supply fan airflow to provide control of the release of potentially radioactive airborne materials from the controlled areas during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path.	9.4.3.1.2.1 12.3.3.3
		Airborne radiation monitors in the exhaust ductwork from areas like the fuel handling area, reactor building controlled areas, auxiliary building controlled areas and access control building areas will alarm in the control room. The control room operators will remotely from the control room manually isolate the supply and exhaust ductwork from the areas as needed and redirect airflow to the containment low volume purge exhaust system, filters, which are then vented through the plant vent stack, during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path.	9.4.3.2.1 12.3.4.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 35 of 76)

Objective		System Features	DCD Reference
		To minimize the buildup of radioactive contamination within the ducts, the exhaust ducts are designed and sized for the transport velocities needed to convey the radioactive contaminants without settling. Ducts for nuclear exhaust and post-accident air cleanup systems are sized for a minimum duct velocity of approximately 2,500 feet per minute (fpm). Control exhaust fan airflow is continuously and automatically set at a predetermined value to maintain a slightly negative pressure in the controlled areas within the A/B. R/B and AC/B relative to the outside atmosphere. This minimizes exfiltration from the radiological controlled areas during normal plant operation. During normal plant operation, the two air handling units and two exhaust fans are placed into operation. Upon energizing the air handling units, the isolation dampers automatically open. Upon energizing the two exhaust fans, their airflow is continuously and automatically controlled at a predetermined value to provide control of the release of potentially radioactive airborne materials from the controlled areas within A/B, R/B including the fuel handling area, and AC/B.	9.4.3.2.1 9.4.3.1.2.1 9.4.3.2.1
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The MCR HVAC system equipment and components are provided with proper access for initial and periodic inspections and maintenance during normal operation.	9.4.1.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 36 of 76)

Objective		System Features	DCD Reference
		The Auxiliary Building, Non-Class 1 E Electrical Room, Main Steam/Feedwater Piping Area, and TSC HVAC systems are designed to provide accessibility to system components for adjustment, maintenance and periodic inspection and testing of the system components to assure proper equipment function and reliability and system availability. The ESF ventilation system is designed to provide accessibility to system components for adjustment, maintenance, and periodic inspection and testing of the system components. Therefore, there are no areas where it is difficult or impossible to conduct regular inspections, testing, maintenance and adjustments.	9.4.3.1.2.1 9.4.3.1.2.2 9.4.3.1.2.3 9.4.3.1.2.4 9.4.5.1.2 9.4.5.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 37 of 76)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Upon receipt of the ECCS actuation signal, the penetration and the safeguard component areas are automatically isolated by the equipment class 2, seismic category I isolation dampers in order that operation of the annulus emergency exhaust system maintains a negative pressure and mitigates the release of airborne fission products to the atmosphere. This low probability of release decreases the need to decontaminate equipment and structures, and ensures contaminants do not spread from the source.	9.4.3.3.1 9.4.5.1.1.1 9.4.5.2.1 12.3.3.3
		The auxiliary building HVAC system controls exhaust fan airflow continuously and automatically at a predetermined value higher than the supply fan airflow to provide control of the release of potentially radioactive airborne materials from the controlled areas during normal plant operation. The exhaust airflow through the plant's vent stack is a radiologically monitored path.	9.4.3.1.2.1 9.4.3.2.1 12.3.3.3
		The containment low volume purge system during normal plant operation maintains the pressure inside containment. This air is treated through a HEPA and charcoal absorber filter and vented through the plant stack, which is a radiologically monitored path.	

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 38 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. There is no embedded piping or ductwork in the HVAC systems. b. HVAC equipment and components are accessible for maintenance and replacement. These items are small in volume and may be removed and replaced with minimum cutting into small pieces.	—
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	The ventilation system produces no radioactive waste. The ventilation system is used to contain and minimize contamination and provide a means to monitor airborne contamination. All of the features listed above minimize the possibility of contamination of components and structures, and therefore minimize the volume of radioactive waste generated during operation and decommissioning. However, since the exhaust ductwork for the auxiliary building ventilation system is the major exhaust path for the plant and takes suction from all areas that are contaminated or potentially contaminated, the ductwork will be internally contaminated by the time the plant is decommissioned. The only method of minimizing the contamination of components is to use less ductwork. This will be a design consideration.	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 39 of 76)

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Liquid Waste Management System (LWMS)

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks**	
		- SSC are designed in accordance with RG 1.143, Table 1	11.2.1.2
		- SSC are specified as welded construction	11.2.1.4
		- Tanks containing liquid are designed in accordance with ANSI 55.6 with 20% safety factor and a minimum of 10% freeboard allowance	11.2.1.2
		- Piping are in accordance with ANSI/ASME B31.3, butt-welded to minimize leakage	Table 11.2-1
		Design features to minimize spills	
		- Tank levels are protected with High alarms and interlocks to prevent over fill and spills	Table 11.2-8
		- Tanks are cross-tied to allow fluid to be directed to other tanks for surge flow	11.2.1.2
		- Tanks are equipped with overflow piping to direct any overflow to drainage system	11.2.1.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 40 of 76)

Objective		System Features	DCD Reference
		Design features to provide containment - Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation. The cubicles are equipped with drainage system to direct any leakage and overflows to sumps with pumps that will redirect flow to other tanks. - In addition, the primary makeup water tanks and refueling water storage auxiliary tank, which are located in a tank house, also have same design features. The walls and floors of the tank house are coated with non-porous material. The floor is sloped towards the drainage pit or funnel. The drainage system is equipped with a liquid detection. - RWS piping from the RWSAT is connected to the R/B through the piping tunnel located at the east side of the tank house. There are construction joints (seal) between the tank house - piping tunnel and the piping tunnel - R/B. This connection is made of elastomeric and watertight material that serves as a seal between the two structures to prevent leakage from the floor of the tunnel connection. These construction joints will be preserved the seal function continuously by the appropriate installation and the periodic maintenance. - All sumps include instrumentation that alarms in the MCR and radwaste control room. The leak detection instruments for the floor drain sump and the equipment drain sump in the A/B alarm locally and also in the MCR through a representative alarm.	12.3.1.1.1.2.E 9.2.6.2.6

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 41 of 76)

Objective		**System Features**	**DCD Reference**
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	- The drainage system is equipped with a liquid detection instrument which can provide early warning for leakage and/or overflow condition. - The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation, and provided with coating with non-porous material to prevent cross contamination.	12.3.1.1.1.2.E

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 42 of 76)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The leak detection system is incorporated in all cubicles in which the tanks contain radioactive fluid (refer to system features for objective #2 above). The tanks include: • Waste Holdup Tanks • Waste Monitor Tanks • Spent Resin Storage Tanks	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 43 of 76)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	In addition to the features discussed in items above, the following design specifications and operational procedures are also implemented:	
		• Stainless steel will be specified as the material which is resistant to corrosion. Surface finish will be polished to facilitate easy decontamination; and	11.2.2.2
		• Suitable smooth-surface coatings facilitate the decontamination of potentially contaminated areas and equipment.	12.3.1.1.2.D
		• Floor drains with properly sloping floors are provided and radioactive and potentially radioactive drainage is separated from non-radioactive drainage.	12.3.1.1.2.D
		• The LWMS and the Solid Waste Management System (SWMS), which employ flexible interconnecting piping for radioactive fluids, are designed with connectors that are incompatible with the connectors for non-radioactive fluids to prevent accidental cross-contamination.	
		The summation of these design and operational features is designed to reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	
		Ion Exchange Columns are provided with a resin-retaining screen on the backwash line, and connected to a backwash discharge line to the waste holdup tank. The screen and discharge line are designed to maintain Occupational Radiation Exposure (ORE) ALARA during resin back washing.	11.2.2.2.6

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 44 of 76)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	a. The design uses a drain header to direct potential leakage and spills to the floor drain sump via a common header. This design minimizes embedded and buried piping within the building foundation slab. No other embedded piping is anticipated in the current design. b. Process equipment items (such as pumps, filters, ion exchangers in the LWMS) are designed with decontamination capability, and to be accessible for maintenance and replacement. Equipment is designed for extended life and will not need to be replaced. If leak develops, equipment can be accessed for repairs in place. Equipment can be decontaminated to remain in the cubicles until decommissioning, during which time they can be cut into smaller pieces for disposal.	—

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 45 of 76)

Objective		System Features	DCD Reference
6	Minimize the generation and volume of radioactive waste both during operation and during decommissioning (by minimizing the volume of components and structures that become contaminated during plant operation).	The treatment technologies (filtration and ion exchange) use simple but effective design that is industrial proven. These technologies concentrate contaminants in the filters and ion exchange resin and thus minimize waste generation during operation. Instrumentation is provided in the design to control the operation to ensure that the filters and ion exchange resin are efficiently used in order to minimize waste generation. The filter and ion exchange vessels are designed and specified to be stainless steel for resistant to corrosion, polished surfaces for ease of decontamination, equipped with flush water for decontamination during operation and decommissioning purpose. These equipment features greatly reduces worker doses during maintenance and decommissioning activities. Chemicals to be used for plant operation are regulated and controlled in order to minimize the generation of mixed waste. To the extent possible, non-hazardous chemicals (both listed and characteristic) are to be minimized for use. If it cannot be avoided, these and all other chemicals are tracked to prevent misuse and thus to minimize waste generation. Solid wastes generated during maintenance activities are required to be first decontaminated and then sorted at the point of generation. These procedures will be instituted in the Process Control Manual and will be used to minimize waste generation and avoid mixing different wastes into single waste classification. These wastes will be sent to specialized off-site facilities for more efficient and economic processing and disposal.	11.2.1.4

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 46 of 76)

Containment **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The US-APWR containment vessel consists of a prestressed, posttensioned concrete structure with a cylindrical wall, hemispherical dome, and flat reinforced concrete foundation slab. The inside surface of the structure is lined with carbon steel. The USAPWR reactor and reactor coolant system (RCS) are completely enclosed in the prestressed concrete containment vessel (PCCV). The PCCV is designed to assure essentially no leakage of radioactive materials to the environment, even if a major failure of the reactor coolant system were to occur.	1.1.2
		The concrete shell inner surface is lined with a minimum 1/4-in. carbon steel plate that is anchored to the concrete shell and dome to provide the required pressure boundary leak tightness. Areas around penetrations, support brackets, inner walls, and heavy components bases have thickened steel liner plates.	3.8.1
		The containment is essentially leak tight to ensure that no significant amount of radioactive material can reach the environment, even in the unlikely event of a RCS failure.	6.02
		The containment is a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat, reinforced concrete foundation slab. To ensure leak tightness during normal operation and under postulated accident conditions, the US-APWR containment is designed and built to safely accommodate an internal pressure of 68 psig.	
		The US-APWR containment is designed to permit periodic leakage rate testing. The periodic leakage rate testing program is the responsibility of any utility that references the US-APWR design for construction and licensed operation.	

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Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	Leakage testing of the RWSP liner (cladding) is performed in accordance with ASME Code requirements. Inspection criteria are delineated in ASME Code Article CC-5000. Failed inspection areas are repaired in accordance with ASME Code.	6.2.1.6

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 48 of 76)

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Reactor Coolant System Boundary and Connected Systems

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Controlled leakage shaft seals employ a well-established seal system that has been used in many operating plants. No.2 seal is designed and tested to maintain full system pressure for enough time to secure the pump. In the case of No. 1 seal failure, the No. 1 seal leak-off line is automatically closed. If the No. 1 seal fails during normal operation, the No. 2 seal minimizes leakage rates. No. 2 seal is able to withstand full system pressure and the No. 3 seal ensures the backup function. This ensures that leakage into atmosphere would not be excessive. The RCPB welds are accessible for inservice inspections (ISI) to assess the structural and leak-tight integrity (see Section 5.2). For the RV, a material surveillance program conforming to applicable codes is provided (see Chapter 5, Section 5.3).	5.4.1.4.1 3.1.2.5 Criterion 14
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	A leak monitoring system is used to provide early detection of leakage from the reactor coolant pressure boundary. Instrumentation is provided to detect significant leakage from the RCPB with indication in the MCR (see Section 5.2). The reactor coolant pressure boundary (RCPB) leak monitoring system provides a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage and monitoring leaks from the reactor coolant and associated systems. This system	1.2.1.5.2.1 3.1.2.5 Criterion 14 5.2.5

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Objective		System Features	DCD Reference
		provides information which permits the plant operators to take corrective action if a leak is evaluated as detrimental to the safety of the facility.	
		The leak monitoring system is designed in accordance with the requirements of General Design Criterion 30 and the regulatory guidance as identified below: • General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary" of Appendix A of 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," to provide a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage. • Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" (Ref.5.2-15).	5.2.5.1
		Identified leakage other than intersystem leakage, such as pump seal or valve packing, is directed to the C/V reactor coolant drain tank where it is monitored by tank pressure, temperature, and level indications.	5.2.5.3
		An important identified leakage path for reactor coolant into other systems is a flow to the secondary side of the steam generator (SG) through the SG tubes. Identified leakage through the SG (primary-to-secondary leakage) is detected by one or more of the following: • Steam generator blowdown water radiation monitor • High sensitivity main steam line monitor • Condenser vacuum pump exhaust line radiation monitor • Liquid samples taken from SG blowdown sampling line.	

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Objective		System Features	DCD Reference
		Indications of unidentified coolant leakage into the containment are provided by an air particulate radioactivity monitor, an airborne gaseous radioactivity monitor, an air cooler condensate flow rate monitoring system, and a containment sump level and flow monitoring system. The sensitivity and response time of leakage detection equipment for unidentified leakage is such that a leakage rate, or its equivalent, of 0.5 gpm can be detected in less than an hour. The methods employed for detecting leakage to the containment from unidentified sources are: • Containment sump level • Containment airborne particulate radioactivity • Condensate flow rate from air coolers. Additionally, humidity, temperature, and pressure monitoring of the containment atmosphere are used for alarms and indirect indication of leakage to the containment.	5.2.5.4
		The following leak detection systems instruments will provide the indications of reactor pressure boundary leakage in the MCR with alarms. The alarms will alert the operating personnel to monitor for leakage. Procedures for converting various indications to a common leakage equivalent will be available to operating personnel. Monitors for items A, C and D below are provided in gallon per minute leakage equivalent.	5.2.5.6 COL

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Objective		System Features	DCD Reference
		Leakage conversion procedure is to be developed as Operating and Emergency Operating Procedures described in DCD Section 13.5.2.1 to convert various indications to an identified and unidentified common leakage equivalent and leakage rate of change. A. Containment airborne particulate radioactivity monitor (Containment radiation monitor, RMS-RE-040) - airborne particulate radioactivity B. Containment air cooler condensate flow rate monitoring system - standpipe level C. Containment sump level and flow monitoring system – sump level D. Containment temperature, pressure, and humidity will only have readouts in the MCR and alarms to indicate occurrence of leakage within the containment. This method is used only to detect leaks and is not used to quantity leak rates.	5.2.5.6

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Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	E. Gross leakage detection methods - charging flow rate, letdown flow rate, pressurizer level, VCT level and reactor coolant temperatures are available as inputs for detection by RCS inventory balance. Containment sump levels and pump operation are also available. Total makeup water flow is available from the plant computer for liquid inventory.	5.2.5.6
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Leakage from the RCP is controlled by three shaft seals arranged in series, such that reactor coolant leakage to the containment is essentially zero. The No. 1 seal reduces the leak-off pressure to that of the volume control tank. The No. 2 and 3 leak-off lines route each seal leakage to the containment vessel reactor coolant drain tank (CVDT).	5.4.1.4.9

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 53 of 76)

Steam Generator Blowdown System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The major elements of the SG program in accordance with NEI 97-06 are outlined below: 5. Primary to secondary leak monitoring, which gives operators information needed to safely respond when tube integrity becomes impaired and significant leakage or tube failure occurs.	5.4.2.2.2
		Radioactive contamination of the SGBDS can occur by a primary to secondary leakage in the steam generator. The SGBDS can become contaminated due to tritium diffusion through SG tubes even without primary-to-secondary leakage. A discussion of the radiological aspects of primary-to-secondary leakage and conditions for operation is contained in Chapter 11. The isolation valve(s) in each blowdown line provides controls for reducing releases by isolating the affected steam generator blowdown line following a steam generator tube rupture. An inline radiation monitor on the common line from the steam generator blowdown sample lines, facilitate leak detection.	10.4.8.3
		The SG blowdown water radiation monitor in the blowdown sample line continuously monitors SG tube leakage. Upon detection of the significant levels of radioactivity, the blowdown flow is also isolated.	10.4.8.5

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Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	Sample blowdown water for chemistry and detect primary-tosecondary leakage with the SG blowdown water radiation monitor. The blowdown samples are used to check the water chemistry of the blowdown water and to detect leakage or failure of a steam generator tube by radiation monitoring. The SGBDS is automatically isolated from the steam generator by closing the isolation valves in the event of an abnormal condition.	1.2.1.5.3.5
		Ground water conditions and issues relative to the US-APWR are site-specific, including monitoring and safeguards requirements to be implemented to the design and operational requirements in RG 4.21.	COLA
		The specification of low cobalt tubing for the US-APWR steam generator design is an important feature of the design; not only in terms of reduced exposure relative to the steam generator, but to the total plant radiation source term. The cobalt content is controlled to not be more than 0.016 mass percent, with an average of 0.014 mass percent for the US-APWR steam generator tubing.	12.3.1.1.1.1.D
		Maximum mass percent of cobalt content for the Upper Core Plate, Upper/Lower Core Support Plate and Upper Core Barrel is 0.05.	Table 12.3-7
		Maximum mass percent of cobalt content for the Main Coolant Piping is 0.15.	Table 12.3-7

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Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	SG blowdown demineralizers are provided with a resin-retaining screen on the backwash line, and connected to a backwash discharge line to the waste holdup tank or the [[waste water system (WWS)]], depending on whether any radioactivity is detected. If the radioactivity from SG tube leakage is detected in the blowdown water, the discharge is diverted to the waste holdup tank in the Liquid Waste Management System for processing. The screen and discharge line are designed to maintain Occupational Radiation Exposure (ORE) ALARA during resin back washing.	10.4.8.2.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 56 of 76)

Engineered Safety System Design **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	*Prestressed Concrete Containment Vessel (PCCV)* - The PCCV is designed to completely enclose the reactor and RCS and assure that essentially no leakage of radioactive materials to the environment would result even if a major failure of the RCS were to occur.	1.2.1.5.4.1

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Flood Protection from External Sources **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Provisions included in the section are as follows; Below grade, the US-APWR nuclear island and other seismic category I and II structures are primarily protected against exterior flooding and the intrusion of ground water by virtue of their thick reinforced concrete walls and base mats. As recommended by NUREG-0800, SRP 14.3.2 (Reference 3.4-4), the external walls below flood level are equal to or greater than two feet thick to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features. Construction joints in the exterior walls and base mats are provided with water stops to prevent seepage of ground water. The COL Applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of ground water into seismic category I buildings and structures. Below-grade exterior wall penetrations such as for piping and conduits have been minimized to reduce the risk of in-leakage and flooding. Where below-grade piping penetrations are necessary, they are designed to preclude water intrusion.	3.4.1.2
		Water-tight doors are used as protective barriers to prevent flood waters from spreading to adjacent divisions in various buildings and elevations. Water-tight doors have position indication for closure verification and are periodically inspected and tested to ensure proper functionality.	3.4.1.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 58 of 76)

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Flood protection from the failure of the plant systems such as the outside storage tanks and yard piping is achieved using dikes, levees, retention basins, component location, and/or sited grading and drainage. Dikes, levees, and retention basins are provided to retain leaks and spills due to postulated failures of tanks and vessels, when appropriate. Alternatively, external tanks and piping are located sufficiently far away so that their failure does not jeopardize safety-related equipment. This is accomplished by locating external flood sources so that any spillage or leakage is directed away from safety-related equipment by virtue of the site grading and drains, and by locating these items away from exterior doors that could act as a pathway for flood waters. In addition, buried yard piping is located either in pipe tunnels or sufficiently far away so that cracks or breaks will not result in soil erosion that undermines safety related structures or components. In summary, the US-APWR seismic category I and II structures provide hardened protection as defined in RG 1.59 (Reference 3.4-5) against external flooding through such design features as sloped roofs, thick reinforced concrete with special porosity reducing additives, waterproofing, and special sealing of joints and penetrations.	3.4.1.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 59 of 76)

Flood Protection from Internal Sources **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
4	Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, reducing any amounts released, and decreasing the spread of the contaminant from the source.	Water-tight doors are used as protective barriers to prevent flood waters from spreading to adjacent divisions in various buildings and elevations. Water-tight doors have position indication for closure verification and are periodically inspected and tested to ensure proper functionality.	3.4.1.3

Buried Seismic Category I Piping, Conduits, and Tunnels

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		System Features	DCD Reference
5	Facilitate decommissioning by (1) minimizing embedded and buried piping, and (2) designing the facility to facilitate the removal of any equipment and/or components that may require removal and/or replacement during facility operation or decommissioning.	Buried seismic category I piping, conduits, and tunnels are not present in the US-APWR standard plant design.	3.7.3.7

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Residual Heat Removal System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The reactor coolant discharged from the CS/RHR pump is circulated through the tube side of the CS/RHR heat exchanger, while cooling is provided by circulating CCW through the shell side. The tubes are welded to the tube sheet to prevent leakage of the reactor coolant.	5.4.7.2.2.2
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The RHRS is provided with a leakage detection system to minimize the leakage from those portions of the RHR system outside of the containment that contain or may contain radioactive material following an accident.	5.4.7.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 61 of 76)

Containment Isolation Systems **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Systems that are including remote manual valve for containment isolation are followings: • Safety injection system. • Containment spray system • Residual heat removal system • Emergency feedwater system • Main steam system • Seal water injection • Component cooling water system • Post-accident sampling return line • Fire protection water supply system The condition in which containment isolation is needed in safety injection system, containment spray system and residual heat removal system is when leak occurs in these systems. These systems are located in safeguard component area. Leak detection system is installed in each system. Level instruments are installed in each pump compartment sump. In addition, if leak is occurred, operators can notice by pump suction/discharge pressure and pump flow rate.	6.2.4.2
		The piping systems penetrating the containment are provided with leak detection, isolation, and containment capabilities. These piping systems are designed with the capability to test, periodically, the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits.	6.2.4.3
		Regulatory Position: **LEAKAGE CONTROL OUTSIDE CONTAINMENT**	Table 6.3-1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 62 of 76)

Objective		System Features	DCD Reference
		Leakage detection and leakage control program outside of containment following an accident shall be discussed. US-APWR Design: A pit (sump) with a leak detector installed in each pump compartment and alarms to MCR to prevent significant leakage of radioactive recirculation water from the high head injection system to the reactor building. The high head injection system is designed to have sufficient redundancy and independence to prevent loss of core cooling function during an accident assuming the isolation of the leaked train after leakage is detected.	

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 63 of 76)

Steam and Power Conversion System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Radioactivity Protection** Under normal operating conditions, the system may become contaminated through steam generator tube leakage and/or tritium diffusion. Radiological monitoring of the main condenser air removal system, the gland seal system, the steam generator blowdown system, and the main steam lines is used to detect contamination and alarm high concentrations. A discussion of the radiological aspects of primary-to-secondary system leakage and limiting conditions for operation is contained in Chapter 11. One of the functions of the steam generator blowdown system described in Subsection 10.4.8 is to monitor the radioactivity level in the secondary cycle.	10.1.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 64 of 76)

Main Steam Supply **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Radioactive contamination of the MSS can occur by a primary side to secondary side leak in the SG. The MSS can also become contaminated due to tritium diffusion through SG tubes even without primary-to-secondary leakage. A discussion of the radiological aspects of primary-to-secondary system leakage and conditions for operation is contained in Chapter 11. The MSIVs provide controls for reducing releases by isolating the affected main steam line following a steam generator tube rupture (SGTR).	10.3.3
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	Radiation monitors on each steam line, condenser vacuum pump exhaust line, GSS exhaust fan discharge line and the SG blowdown line facilitate primary-to-secondary leak detection.	10.3.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 65 of 76)

Circulating Water System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Large CWS leaks due to pipe/expansion joint failures is indicated and alarmed in the control room by a loss of vacuum in the condenser shell. The effects of flooding due to a CWS failure, such as the rupture of an expansion joint, assumes that the flow into the T/B comes from both the upstream and downstream side of the break and also conservatively assumes that one system isolation valve does not fully close. This does not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the T/B and the base slab of the T/B is located at grade elevation. Any leakage from the CWS due to tube leakage into the main condenser is detected by the secondary sampling system (SSS).	10.4.5.3.4.1
		Also, the TCS is maintained at a higher pressure than the non-ESW system (which draws water from CWS) to prevent leakage of the non-ESW into the TCS.	10.4.5.3.4.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 66 of 76)

Objective		System Features	DCD Reference
3	Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult or impossible to conduct regular inspections (such as for spent fuel pit, tanks that are in contact with the ground, and buried, embedded, or subterranean piping) to avoid release of contamination of the environment.	The CWS is automatically isolated in the event of gross leakage into the turbine building (T/B) condenser area to prevent flooding of the T/B.	10.4.5

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 67 of 76)

Condensate and Feedwater System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The CFS is designed to permit appropriate in-service and functional testing of the system and components to ensure structural integrity and leak-tightness.	10.4.7.1.2

Emergency Feedwater System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The EFWS is designed to permit appropriate in-service and functional testing of the system and components to ensure their structural integrity and leak-tightness.	10.4.9.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 68 of 76)

Auxiliary Steam Supply System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	The condensate piping from the ASSS drain tank is a single-walled carbon steel pipe run above ground in pipe chases from the A/B to the T/B, and is then connected to double-walled welded carbon steel piping through the T/B wall penetration to the auxiliary boiler. Since this is not a high traffic area, this segment of pipe is run above ground and is slightly sloped so that any leakage is collected in the outer pipe and drained to the auxiliary boiler building. At the auxiliary boiler building end, a leak detection instrument is provided to monitor leak. A drain pipe is provided to direct any drains to the building sump. The facility floor has an epoxy coating and the sump has a liquid level detection instrument alarm for operator action. The sump drain line pumps the liquid contents to the T/B sump for collection and analysis. The sump drain line inside the auxiliary boiler building is constructed of single-walled carbon steel pipe. From the auxiliary boiler building wall penetration to the T/B outside wall, the drain line is constructed of double-walled piping run above ground. Inside the T/B, the pipe is single-walled as it is routed to the T/B sump. The double-walled segment of the sump drain line is sloped towards the turbine building and is equipped with leak detection instrumentation. The steam piping is jacketed with insulation and heat protection. The Auxiliary Boiler is designed with a blowdown connection from the boiler drum to the building sump. The boiler blowdown is drained directly into the sump for transfer into the Turbine Building sump. The T/B sump contents are then pumped to the Waste Holdup Tanks in the LWMS for processing. This design is supplemented by operational programs which includes periodic hydrostatic or pressure testing of pipe segments, instrument calibration, and when required, visual inspection and maintenance of piping, trench and instrument integrity.	10.4.11.2.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 69 of 76)

Auxiliary Steam Supply System **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
2	Provide for adequate leak detection capability to provide prompt detection of leakage for any structure, system, or component which has the potential for leakage.	The auxiliary steam drain monitors the leakage of the radioactive materials from the boric acid evaporator to the condensed water of the ASSS.	10.4.11.1.2
		Monitoring the leakage from the primary side of the evaporator, the radiation monitor is attached to the downstream of the auxiliary steam drain pump. The high alarm of the monitor isolates the pump discharge line and steam supply line from main steam and trips the pump.	10.4.11.2.1
		Leakage of radioactive materials from primary side in the B.A. evaporator.	10.4.11.2.3

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 70 of 76)

Objective		System Features	DCD Reference
		If there is leakage of radioactive materials from the primary side in the B.A. evaporator, the auxiliary steam drain tank pump discharge isolation valve is closed and the auxiliary steam drain pumps are tripped by the auxiliary steam drain monitor high alarm. The high signal is alarmed to the main control room.	

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 71 of 76)

Refueling Water Storage Auxiliary Tank (RWSAT) **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		System Features	DCD Reference
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks** The SFPCS consists of one 100% capacity RWSAT (29,410 cu.ft), pumps, associated valves, piping, and instrumentation. The piping to and from the RWSAT is single-walled stainless steel that runs above ground and penetrates the building wall directly into the tank. For piping between buildings, penetration sleeves are provided to collect and direct any leakage back into the building drain for further processing. The RWSAT employs leak-tight valves to minimize leakage to environment. This design is supplemented by operational programs, which include periodic visual inspections for piping integrity. Testing the piping segments will be included as a part of the plant routine inspections and maintenance program.	9.1.3.2

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 72 of 76)

US-APWR Equipment Design Considerations for Keeping Radiation Exposure ALARA

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks** Consistent with the requirements of 10 CFR 20.1406 (Reference 12.1-6), the design criteria strive to minimize the possibilities for contamination of the facility and environment, to facilitate eventual decommissioning, and to reduce the generation of radwaste. Examples of practical means for system design to minimize the possibilities for contamination are described below. • The basic plant layout is planned to minimize the spread of contamination. • Radioactive and potentially radioactive drains are separated from non-radioactive drains. • The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination. • Ventilation systems are designed for minimizing the spread of airborne contamination • In building compartments with a potential for contamination, the exhaust is designed forgreater volumetric flow than the air intake into that area.	12.1.2.2.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 73 of 76)

US-APWR Equipment Design Considerations for Keeping Radiation Exposure ALARA

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		**System Features**	**DCD Reference**
		Additionally, examples of practical means for equipment design to minimize the possibilities for contamination are described below. • Overflow lines of tanks are directed to the waste collection system to control any contamination within plant structures. • Tank vents are hard-piped to heating, ventilation, and air conditioning (HVAC) ducts, not to open room spaces. • Equipment vents and drains from highly radioactive systems are piped directly to the collection system. • All-welded piping systems are employed in contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at the joints. • The valves in some radioactive systems are provided with leak-off connections piped directly to the collection system. • Floor drains are provided to recover radioactive leakage. • Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts. • Refueling tool has smooth surfaces to reduce contamination. Collection piping: In all potentially radioactive areas, the collection system piping for the liquid waste is stainless steel. Potentially radioactive laboratory and decontamination waste, regeneration waste, and detergent waste collection system piping is stainless steel. Non-radioactive collection piping is generally made of carbon in despite of partially stainless steel. The collection piping is double-walled piping with leak detection instruments to prevent contamination discharge to the environment at the lowest elevation floor of the A/B and the R/B. Collection sumps: The strategically located sumps are used to collect radioactive and non-radiactive liquid waste. The non-radioactive collection sumps are constructed of concrete with the corrosion resistant coating or liner.	

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 74 of 76)

Component Cooling Water (CCW) Isolation Valves

(Note: The "System Features" column consists of excerpts/summary from the DCD)

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	**Design features to minimize leaks** The CCW system supplies cooling water to components located in the nonseismic Category I buildings (turbine building and auxiliary building). Each CCW supply line (A2 and C2) has two in-series air operated isolation valves. These valves close automatically to isolate the non-seismic Category I portion of the CCW system upon receipt of a S signal, P signal or surge tank low-low level signal (See Figure 9.2.2-1, Sheets 1, 2 of 9). The open/close positions of the valves are displayed in the MCR. These valves automatically close to protect against CCW leakage upon detection of sudden pressure drop and/or operator initialed demand signals. The valves also fail close. In-series check valves are provided on the CCW return lines from the nonseismic Category I portion of the CCW system (See Figure 9.2.2-1, Sheets 1, 2 of 9).	9.2.2.2.1.5
		If leakage from a higher pressure component to the CCWS should occur, the water level of CCW surge tank increases and an alarm is transmitted to the MCR. If the in-leakage is radioactive, the radiation monitors of the CCWS will provide an indication in the MCR of an increased radiation level and transmit an alarm when the radiation level reaches its set point. After the leak source is identified, the leak is isolated from the CCWS. In the event that the in-leakage is through the RCP thermal barrier HX, the isolation valves on the RCP thermal barrier HX CCW return line are automatically closed by the high flow rate signal, thereby preventing further CCWS contamination.	9.2.2.3.1

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 75 of 76)

Tanks **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		**System Features**	**DCD Reference**
1	Minimize leaks and spills and provide containment in areas where such events may occur.	Tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the floor drains to control and minimize the potential for the spread of contamination within plant structures. Tanks containing radioactive fluids are designed to have overflow lines at least equal in size to the largest inlet line. The tank vent line is either equipped with open vents to the cubicle or connected directly to the ventilation system. The spent resin tank vents are equipped with a break-pot, which separates the air from the moisture and any entrained resin, which are subsequently sent to the A/B sump, and vents the air to the exhaust ductwork. These measures minimize the possible contamination of the area and the ductwork. Tanks containing radioactive particulate material shall have smooth welds and mixing, flushing and cleaning capabilities to prevent retention of the radioactive particulate material. Tanks containing radioactive particulate material shall include one or more of the features mentioned below: • Purification of radioactive fluids (filtration or ion exchange) prior to discharge • Sloped or cone-shaped tank bottom • Smooth surfaces and smooth internal welds to minimize crud traps • Tank flushing capability • Agitation by recirculation flow capability • Lancing or chemical cleaning capability	12.3.1.1.1.2.E

Table 12.3-8 Regulatory Guide 4.21 Design Objectives and Applicable DCD Subsection Information for Minimizing Contamination and Generation of Radioactive Waste (Sheet 76 of 76)

Tanks **(Note: The "System Features" column consists of excerpts/summary from the DCD)**

Objective		**System Features**	**DCD Reference**
		Tank cubicles are coated with non-porous material up to a wall height to contain the entire tank content. The floors of cubicles that may contain radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation. The cubicle coating for these tanks is an epoxy coating which is Service Level II as defined in RG 1.54, and is subject to the graded QA provisions, selection, qualification, application, testing, maintenance, and inspection provisions of RG 1.54 and standards referenced therein, as applicable to Service Level II coatings. The cubicles are equipped with a drainage system to direct any leakage and overflows to sumps with pumps that will redirect flow to other tanks. The drainage system is equipped with a liquid detection instrumentation which can provide early warning for leakage and/or overflow condition to initiate operator actions. The floors of these cubicles containing radioactive fluid are sloped to facilitate faster drainage and to minimize liquid accumulation, and provided with coating with non-porous material to prevent cross contamination.	

Table 12.3-9 Projected Dose Rates for the Access Areas 1 week after an Accident

POST ACCIDENT Access Areas	Dose Rate 1 week after an Accident
CS/RHR, SI Pump Discharge Flow Access Area	≦ 15 mrem/h
CS/RHR Pump Minimum Flow Access Area	≦ 15 mrem/h

Table 12.3-10 Mission Dose for the Access Areas access route 1 week after an Accident

Access Area	Task description	Time when access required [h]	Max dose rate [rem/h]	Mission dose [rem]	Access route zone map No.
CS/RHR, SI Pump Discharge Flow Access Area	Access to A-CS/RHR,SI Pump Discharge Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 1,2,3
		6.1	1.5E-02	9.1E-02	
		Total		9.1E-02	
	Access to B-CS/RHR,SI Pump Discharge Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 1,2,3
		6.1	1.5E-02	9.2E-02	
		Total		9.2E-02	
	Access to C-CS/RHR,SI Pump Discharge Flow Access Area from AC/B (round trip)	5.5E-02	2.5E-03	1.4E-04	Figure 12.3-11 Sheet 1,2,3
		6.0	1.5E-02	9.1E-02	
		Total		9.1E-02	
	Access to D-CS/RHR,SI Pump Discharge Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 1,2,3
		6.1	1.5E-02	9.1E-02	
		Total		9.1E-02	
CS/RHR Pump Minimum Flow Access Area	Access to A-CS/RHR, Pump Minimum Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 3
		6.0	1.5E-02	9.1E-02	
		Total		9.1E-02	
	Access to B-CS/RHR, Pump Minimum Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 3
		6.1	1.5E-02	9.1E-02	
		Total		9.1E-02	
	Access to C-CS/RHR, Pump Minimum Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 3
		6.0	1.5E-02	9.1E-02	
		Total		9.1E-02	
	Access to D-CS/RHR, Pump Minimum Flow Access Area from AC/B (round trip)	3.0E-02	2.5E-03	7.4E-05	Figure 12.3-11 Sheet 3
		6.0	1.5E-02	9.0E-02	
		Total		9.0E-02	

(Note) Walk speed is usually about 13000 ft/h (4 km/h) and stairs are about 6500 ft/h (2 km/h).
Replacement, calibration or repair time is conservatively assumed to require 6 hours.

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 1 of 34)
Site

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 2 of 34)
Reactor Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 3 of 34)
Reactor Building Sectional View B-B

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 4 of 34)
Reactor Building at Elevation -26'-4"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 5 of 34)
Reactor Building at Elevation -8’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 6 of 34)
Reactor Building at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 7 of 34)
Reactor Building at Elevation 13’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 8 of 34)
Reactor Building at Elevation 25’-3”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 9 of 34)
Reactor Building at Elevation 35'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 10 of 34)
Reactor Building at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 11 of 34)
Reactor Building at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 12 of 34)
Reactor Building at Elevation 101'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 13 of 34)
Reactor Building at Elevation 115'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 14 of 34)
Auxiliary Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 15 of 34)
Auxiliary Building at Elevation -26'-4"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 16 of 34)
Auxiliary Building at Elevation -8'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 17 of 34)
Auxiliary Building at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 18 of 34)
Auxiliary Building at Elevation 15’-9”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 19 of 34)
Auxiliary Building at Elevation 25'-3"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 20 of 34)
Auxiliary Building at Elevation 35'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 21 of 34)
Auxiliary Building at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 22 of 34)
Auxiliary Building at Elevation 76'-5"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 23 of 34)
Auxiliary Building at Elevation 89’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 24 of 34)
Turbine Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 25 of 34)
Turbine Building at Elevation -18’-0”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 26 of 34)
Turbine Building at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 27 of 34)
Turbine Building at Elevation 34'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 28 of 34)
Turbine Building at Elevation 61’-0”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 29 of 34)
Turbine Building at Elevation 88'-10"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 30 of 34)
Turbine Building at Elevations 108’-4” and 113’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 31 of 34)
Turbine Building at Roof Elevation

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 32 of 34)
Access Building Sectional View A-A

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 33 of 34)
Access Building at Elevations -26’-4”, -8’-0” and 3’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-1 Radiation Zones for Normal Operation/Shutdown (Sheet 34 of 34)
Access Building at Elevations 17'-9", 30'-2" and 48'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 1 of 10)
Power Block at Elevation -26'-4"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 2 of 10)
Power Block at Elevation -8’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 3 of 10)
Power Block at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 4 of 10)
Power Block at Elevation 13'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 5 of 10)
Power Block at Elevation 25'-3"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 6 of 10)
Power Block at Elevation 35’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 7 of 10)
Power Block at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 8 of 10)
Power Block at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 9 of 10)
Power Block at Elevation 101’-0”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-2 General Plant Arrangement with Post Accident Vital Areas (Sheet 10 of 10)
Power Block at Elevation 115'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 1 of 10)
Power Block at Elevation -26’-4”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 2 of 10)
Power Block at Elevation -8’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 3 of 10)
Power Block at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 4 of 10)
Power Block at Elevation 13’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 5 of 10)
Power Block at Elevation 25’-3”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 6 of 10)
Power Block at Elevation 35’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 7 of 10)
Power Block at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 8 of 10)
Power Block at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 9 of 10)
Power Block at Elevation 101’-0”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-3 Post Accident Radiation Zone MAP:1hour After Accident (Sheet 10 of 10)
Power Block at Elevation 115’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 1 of 10)
Power Block at Elevation -26’-4”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 2 of 10)
Power Block at Elevation -8’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 3 of 10)
Power Block at Elevation 3’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 4 of 10)
Power Block at Elevation 13’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 5 of 10)
Power Block at Elevation 25’-3”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 6 of 10)
Power Block at Elevation 35'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 7 of 10)
Power Block at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 8 of 10)
Power Block at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 9 of 10)
Power Block at Elevation 101'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-4 Post Accident Radiation Zone MAP:1day After Accident (Sheet 10 of 10)
Power Block at Elevation 115'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 1 of 10)
Power Block at Elevation -26’-4”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 2 of 10)
Power Block at Elevation -8'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 3 of 10)
Power Block at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 4 of 10)
Power Block at Elevation 13'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 5 of 10)
Power Block at Elevation 25'-3"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 6 of 10)
Power Block at Elevation 35’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 7 of 10)
Power Block at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 8 of 10)
Power Block at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 9 of 10)
Power Block at Elevation 101'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-5 Post Accident Radiation Zone MAP:1week After Accident (Sheet 10 of 10)
Power Block at Elevation 115'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 1 of 10)
Power Block at Elevation -26'-4"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 2 of 10) Power Block at Elevation -8'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 3 of 10)
Power Block at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 4 of 10)
Power Block at Elevation 13’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 5 of 10)
Power Block at Elevation 25’-3”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 6 of 10)
Power Block at Elevation 35’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 7 of 10)
Power Block at Elevation 50’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 8 of 10)
Power Block at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 9 of 10)
Power Block at Elevation 101’-0”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Post Accident Radiation Zone MAP:1month After Accident (Sheet 10 of 10)
Power Block at Elevation 115'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-7 Isometric View of Main Control Room Shielding

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-8 Labyrinth for radiation protection around Fuel Transfer Tube

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-9 The typical layout of air handling unit

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-10 The sampling point of the airborne radioactivity monitors

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 1 of 10)
Power Block at Elevation -26'-4"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 2 of 10)
Power Block at Elevation -8’-7”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 3 of 10) Power Block at Elevation 3'-7"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 4 of 10)
Power Block at Elevation 13’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 5 of 10)
Power Block at Elevation 25'-3"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 6 of 10)
Power Block at Elevation 35’-2”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 7 of 10)
Power Block at Elevation 50'-2"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 8 of 10)
Power Block at Elevation 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 9 of 10)
Power Block at Elevation 101’-0”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Post Accident Radiation Zone MAP: 1week After Accident (Sheet 10 of 10)
Power Block at Elevation 115’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-12 Layout Drawing of the Tank House

12.4 Dose Assessment

Radiation exposures in the plant are primarily due to direct radiation from components and equipment containing radioactive material. In addition, in some areas of the plant, there can be radiation exposure to personnel due to the presence of airborne radionuclides. This section addresses the anticipated occupational radiation exposure due to normal operation and anticipated inspection and maintenance.

The radiation source terms and the shielding design that determine direct radiation dose rates are discussed in Section 12.2 and Subsection 12.3.2. The plant layout of equipment and shielding is shown in Figure 12.3-1. Radiation exposure to personnel due to the presence of airborne radionuclides is discussed in Subsection 12.2.2.

The plant is designed to keep radiation exposures within limits and ALARA (see Sections 12.1 and 12.3). In addition, a radiation protection program in place to assure that radiation exposures is to be within limits and ALARA is described in Section 12.5.

In-plant radiation exposures during normal operation and anticipated operational occurrences are incurred from the following activities (Reference 12.4-1):

- Operations and surveillance
- Routine maintenance
- Waste processing
- Refueling operations
- ISI
- Special maintenance

The radiation exposure from the above activities will vary from plant to plant. The NRC compiles and publishes annual occupational radiation exposures from its licensees in NUREG-0713, “Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities” (Reference 12.4-2). The latest edition of NUREG-0713 (Reference 12.4-2) (Vol. 27) includes data through 2005. Table 4.2 of NUREG-0713 (Reference 12.4-2) lists the average annual exposures from US pressurized-water reactors (PWRs) for the years 1973 through 2005. Prior to 1988, the reported exposure per reactor routinely exceeded 300 person-rem per year. After 1988, the exposures dropped from 300 person-rem per year to less than 100 person-rem per year in 2000. The total average exposure per reactor has remained less than 100 person-rem per year since that time.

See the following items for reasons for the drop in radiation exposures:

- Better nuclear fuel performance. The amount of fission products “leaked” from fuel elements is far smaller than the “design basis” 1% value.

- Better fuel performance supports operation for longer cycles compared to the annual refueling outages commonly used up to abut 1990. The US-APWR will support reload cycles up to 24 months.

- ALARA awareness, ALARA training, and ALARA planning included by plant operators in refueling and maintenance activities further reduced unnecessary exposures.

- Refueling and other major outage planning that includes ALARA considerations has led to shorter outages and, therefore, less exposure.

- Plants have also undertaken Cobalt reduction programs to minimize the Cobalt-60 source of radioactivity.

Tables 12.4-1 thorough 12.4-8 are based on RG 8.19 "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates," Rev 1, 1979 (Reference 12.4-1). Use was made of certain data from Revision 0 of this Regulatory Guide because the data from Revision 0 are considered to be reasonable estimates of annual exposure since the data generally agrees with more current similar industry averages. Table 12.4-8 is a summation of the US-APWR's individual dose estimates in person-rem/year. Annual dose estimates are based on 24-month refueling cycle as a typical example. The total annual station exposure is 71.03 person-rem, which is substantially less than the 100 person-rem annual value cited in NUREG -0713 (Reference 12.4-2).

Because of extended fuel cycles (18-24 months), some activities performed concurrently with refueling outages (e.g., major maintenance and ISIs) may not be done annually. The annual doses for a particular reactor may vary around the 100 person-rem average. For this reason, the NRC is also compiling a rolling 3-year average as a better metric to account for the longer fuel cycle. The information in NUREG-0713, Table 4.6 (Reference 12.4-2), lists the 3-year rolling average exposure for US PWRs for the years 2003-2005. The 3-year average exposures at 42 sites (69 PWR reactors) range from 43 person-rem (Seabrook) to 195 person-rem (Palisades). Of the 42 sites, only eight exceeded 100 person-rem per year.

Specific current data from a two-unit Westinghouse four-loop PWR plant site, the Comanche Peak Steam Electric Station, for seven consecutive years is presented in Table 12.4-9. There is a rise in 2002 and 2005 to approximately 120 person-rem per reactor unit due to maintenance and refueling. The average over the 3-year period of 2000 through 2002 and 2003 through 2005 is 75.67 and 75.09 person-rem respectively. This data exemplifies the NRC's reasoning that a three-year rolling average may be a better representation of expected dose rates, and indicates that the expected radiation exposures during the similar operation of the US-APWR may be less than 100 person-rem per year.

The conservative source term and shielding, the nuclear fuel performance, the facility layout, and the equipment and piping layout and design incorporated in the US-APWR design will contribute to lower plant exposures (See Section 12.3). Based on PWR experience cited above from NUREG-0713 (Reference 12.4-2), RG 8.19 (Reference

12.4-1), the Comanche Peak Steam Electric Station data, and the US-APWR design, it is expected that operation of the US-APWR results in radiation exposures of less than 100 person-rem per year.

12.4.1 Occupational Radiation Exposure

Radiation exposures to operating personnel are restricted to the limits of 10 CFR 20 (Reference 12.4-3). The Operational health physics program described in Section 12.5 and the radiation protection features described in Section 12.3 together maintain occupational radiation exposures ALARA. The airborne concentration is shown in Table 12.2-61. The airborne dose is less than the dose limit of 10 CFR 20 (Reference 12.4-3).

In the analysis of occupational radiation exposure data from operating plants of a design similar to the US-APWR, that is, domestic plants having Westinghouse-designed nuclear steam supply systems, the best operating plant performance is 0.1 rem per megawatt electrical per year of electricity produced. Major factors contributing to this level of occupational radiation exposure include low plant radiation fields, good layout and access provisions, and operational practices and procedures that minimize time spent in radiation fields. As discussed, the US-APWR design incorporates features to reduce occupational radiation exposure that goes beyond the designs provided for plants currently in operation.

The estimated annual occupational radiation exposures are developed within the following categories (Reference 12.4-1):

- Routine operations and surveillance
- Non-routine operations and surveillance
- Routine maintenance
- Waste processing
- Refueling operations
- ISI
- Special maintenance

Exposure data obtained from operating plants have been reviewed to obtain a breakdown of the doses incurred within each category. For several routinely performed operations, this information has been used to develop detailed dose predictive models. These models identify the various steps that are included in the operation, radiation zones, required number of workers, and the time to perform each step. This information has been used to develop dose estimates for each of the preceding categories. There is no separate determination of doses due to airborne activity. Experience demonstrates that the dose from airborne activity is not a significant contributor to the total doses and wearing a respiratory mask and installation of a temporary area exhaust equipment protect workers from airborne contamination caused by drying RCS internal components exposed to air.

12.4.1.1 Operations and Surveillance

To support plant operations, the performance of various systems and components is monitored. In addition, the operation of some of the manual valves requires personnel to enter radiation fields.

Examples of activities in this category are as follows:

- Routine inspections of the plant components and systems
- Unidentified leak checks
- Operation of the manual valves
- Reading of the instruments
- Routine health physics patrols and surveys
- Decontamination of the equipment or plant work areas
- Calibration of the electrical and mechanical equipment
- Chemistry sampling and analysis

When the plant is at power, the containment radiation fields are significantly higher than at plant shutdown. The frequency and duration of at-power containment entries is dependent on the plant operator. Tables 12.4-1 and 12.4-2 provide a breakdown of the collective doses for routine and non-routine operations and surveillance respectively. The occupational radiation exposure above the operation floor in the containment vessel is reduced by the labyrinth structure between reactor vessel and primary shielding.

12.4.1.2 Routine Maintenance

Routine maintenance is required for mechanical and electrical components. Table 12.4-3 provides a breakdown of the collective doses for routine maintenance. Exposures can be minimized by having good access to equipment and large work area for adequate maintenance activities, a characteristic of the US-APWR layout.

12.4.1.3 Waste Processing

The US-APWR radwaste system designs incorporate an uncomplicated approach to waste processing. Elimination of high maintenance components contributes significantly to lower anticipated doses due to waste processing activities. Estimated annual doses from waste processing operations appear in Table 12.4-4.

12.4.1.4 Fuel Handling

The refueling process is labor intensive. Detailed planning and the coordination of effort are essential in order to maintain personnel doses ALARA. The incorporation of advanced technology into the refueling process also reduces doses. Table 12.4-5 provides dose estimates for the various refueling activities.

The design of the new fuel handling and storage areas precludes criticality under all postulated normal and accident conditions. Therefore, criticality monitors, as stated in 10 CFR 50.68 (Reference 12.4-4), are not needed. The long-term refueling cycle of the US-APWR (maximum 24 month) reduces the annual occupational radiation exposure.

12.4.1.5 In-service Inspection

American Society of Mechanical Engineers Code, Section XI (Reference 12.4-5) requires periodic ISI on plant safety-related components. The Code defines the ISI interval as a 10-year period and sets requirements for each one-third interval (each 40 months). In general, at least 25% (with credit for no more than 33-1/3 percent) of the specified inspections must be performed in each 40-month testing interval. The amount of inspection required for an area varies according to the category but is explicitly defined in the Code. Table 12.4-6 provides the dose estimates for ISI activities. Insulation with shielding materials is used on the primary coolant piping. The inspection of reactor vessel are remote-operated with the UT (Ultrasonic Test) machine which MHI developed. These effects reduce the occupational radiation exposure.

12.4.1.6 Special Maintenance

Maintenance that goes beyond the routine scheduled maintenance is considered as special maintenance. This category includes both the modification of equipment to upgrade the plant and repairs to failed components. Dose estimates assume no significant equipment upgrade efforts. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will be lower for the US-APWR than for current plant designs because of the reduced radiation fields, increased equipment reliability, and the reduced number of components relative to currently operating plants. Table 12.4-7 provides the estimated doses due to special maintenance operations. In the inspection of SG heat-transfer tube, the occupational radiation exposure can be reduced by the remote operation using the ECT (Eddy Current Test) machine and its remote attachment equipment, both of which MHI developed.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the seven activity categories discussed above are summarized in Table 12.4-8.

12.4.1.8 Post-Accident Actions

Requirements of 10 CFR 50.34(f)(2)(viii) (Reference 12.4-6), are included in Chapter 1, Subsection 1.9.3. If procedures are followed, the design prevents radiation exposures to any individual from exceeding 5 rem to the whole body or 50 rem to the extremities. Figure 12.3-2 shows the general plant arrangement with the vital areas that must be accessed in the post-accident environment identified. Figures 12.3-3 through 12.3-6 contain radiation zone maps for plant areas including those areas requiring post-accident access. This figure shows projected radiation zones in areas requiring access and access routes or ingress, egress, and performance of actions at these locations. The radiation zone maps reflect maximum radiation fields over the course of an accident. The analyses that confirm that the individual personnel exposure limits following an accident are not exceeded reflect the time-dependence of the area dose rates and the required post-accident access times. The post-accident doses on the radiation zone maps are determined by adding the upper limit dose on the radiation zone maps under normal conditions (from Figure 12.3-1) to the gamma dose from the airborne radioactive materials in containment after LOCA. The doses are calculated by modeling the outer shield and containment as a cylinder with the containment free volume. Then the outer shield is ignored in the penetration areas dose calculation although other shields having sufficient shielding effect are considered. The areas that require post-accident accessibility are as follows:

- Main Control Room (MCR)
- Technical Support Center (TSC)
- Postaccident sampling system (PASS)
- Radiochemistry laboratory (sample analysis)
- Hot counting room

Accident parameters and sources are discussed and evaluated in Chapter 15, Subsection 15.6.5.5.

12.4.1.9 Dose to Construction Workers

For multiunit plants, the COL Applicant is to provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s).

12.4.2 Radiation Exposure at the Site Boundary

12.4.2.1 Direct Radiation

The direct radiation from onsite contained sources is described in this subsection. The direct radiation from the containment and other plant buildings is negligible.

12.4.2.2 Doses Due to Airborne Radioactivity

Doses at the site boundary due to releases of airborne radioactivity are given in this subsection and in Chapter 11, Subsection 11.3.3.

12.4.3 Combined License Information

COL 12.4(1) For multiunit plants, the COL Applicant is to provide estimated annual doses to construction workers in a new unit construction area, as a result of radiation from onsite radiation sources from the existing operating plant(s).

12.4.4 References

12.4-1 Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants – Design Stage Man-Rem Estimates. RG 8.19, Rev 0, May 1978, Rev 1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1979.

12.4-2 Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities – 2005. Thirty-Eighth Annual Report, 1981, NUREG-0713, Vol 27, U.S. Nuclear Regulatory Commission, 2006.

12.4-3 "Standards for Protection against Radiation," Energy. Title 10 Code of Federal Regulations Part 20, U.S. Nuclear Regulatory Commission, Washington, DC.

12.4-4 "Critically accident requirements," Energy. Title 10 Code of Federal Regulations Part 50.68, U.S. Nuclear Regulatory Commission, Washington, DC.

12.4-5 "Rules for Inservice Inspection of Nuclear Power Plant Components," Boiler and Pressure Vessel Code – 2007 Edition. Section XI, American Society of Mechanical Engineers Code, Washington, DC.

12.4-6 "Contents of construction permit and operating license applications; technical information." Energy. Title 10 Code of Federal Regulations Part 50.34 (f) (2) (viii), U.S. Nuclear Regulatory Commission, Washington, DC.

Table 12.4-1 Occupational Dose Estimates During Routine Operations and Surveillance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Routine patrols	0.2	0.5	2	1/shift*1	**0.22**
Checking:					
Containment cooling system	1	1	1	1/month	**0.01**
Accumulators	1.5	1	1	1/month	**0.02**
Boric acid makeup system	5	0.2	1	1/day	**0.36**
Fuel pool system	1	0.25	1	1/day	**0.09**
RHR pump	1	0.2	1	1/day	**0.07**
Total					**0.77**

*1 There are 3 shifts per day.

Table 12.4-2 Occupational Dose Estimates During Nonroutine Operations and Surveillance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Operation of equipment:					
Safety injection system	5	1	1	1/month	**0.06**
Instrument calibration	2	1	1	1/day	**0.73**
Collection of radioactive samples:					
Liquid system	10	0.5	1	1/day	**1.83**
Gas system	5	0.5	1	1/month	**0.03**
Solid system	10	0.5	1	4/year	**0.02**
Radiochemistry	1	1	2	1/day	**0.73**
Radwaste operation	3	8	3	1/week	**3.75**
Health physics	1	2	2	1/day	**1.46**
Total					**8.61**

Table 12.4-3 Occupational Dose Estimates During Routine Maintenance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Changing filters:					
Waste filter	100	0.5	1	6/year	**0.3**
Laundry filter	100	0.5	1	10/year	**0.5**
Boric acid filter	100	0.5	1	2/year	**0.1**
Mechanicals:					
Mechanical components *1	10	100	5	1/2year	**2.5**
Electrical components *1	5	200	6	1/2year	**3**
Refurbishing of valves *1	50	40	4	1/2year	**4**
Reactor Coolant pump *1	10	100	4	1/2year	**2**
Other pumps *1	0.25	100	4	1/2year	**0.05**
Others:					
Radiation surveillance by RP personnel *2	5	1	2	1/day	**3.65**
Washing / decontamination *2	1	1	3	1/day	**1.1**
Operation of equipment *2	1	1	2	1/day	**0.73**
Total					**17.93**

*1 The dose for this activity is taken from data from operating Japanese PWRs. The dose has been adjusted to account for the 24-month refueling cycle currently assumed for the US-APWR.

*2 This activity is part of routine maintenance that is done intermittently and not truly a daily activity. The total yearly dose for this activity is taken from data from operating Japanese PWRs. The remaining quantities were adjusted to match the total yearly dose given the assumption of the frequency being once per day.

Table 12.4-4 Occupational Dose Estimates During Waste Processing

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Control room	0.1	3000	1	1/year	**0.3**
Sampling and filter changing	10	4	1	1/week	**2.1**
Panel operation, inspection, and testing	1	2	1	1/day	**0.73**
Operation of waste processing and packaging equipment	2	12	2	1/week	**2.5**
Total					**5.63**

Table 12.4-5 Occupational Dose Estimates During Refueling

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Reactor pressure vessel head and internals-removal and installation*	100	20	8	1/2year	**8.0**
Fuel preparation	10	24	2	1/2year	**0.24**
Fuel handling	2.5	100	4	1/2year	**0.5**
Total					**8.74**

* The dose for this activity including servicing of the top-mounted ICIS has been estimated as 16.0 person-rem by MHI for the US-APWR ICIS design. Given the 24-month refueling cycle currently assumed for the US-APWR, the dose has been adjusted to account for this activity occurring only every 2 years.

Table 12.4-6 Occupational Dose Estimates During ISI

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Providing access: installation of platforms, ladders, etc., removal of thermal insulation	40	30	4	1/2year	**2.4**
Inspection of welds	40	100	3	1/2year	**6.0**
Follow up: installation of thermal insulation platform removal and cleanup	40	40	4	1/2year	**3.2**
Total					**11.6**

Table 12.4-7 Occupational Dose Estimates During Special Maintenance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Inspection of SG heat-transfer tube *1	50	40	10	1/2year	10.0
Inspection of RCP internal *2	100	90	4	1/10year	3.6
Servicing of detectors around core *1	10	10	3	1/2year	0.15
Inspection of ICIS guide thimble *3	40	50	8	1/4year	4.0
Total					**17.75**

*1 The dose for this activity is taken from data from operating Japanese PWRs. The dose has been adjusted to account for the 24-month refueling cycle currently assumed for the US-APWR.

*2 The dose for this activity is taken from data from operating Japanese PWRs. This activity is assumed to occur only once every ten years. The data for this dose has been adjusted to account for this activity occurring only every 10 years (every 5 cycles).

*3 The dose for this activity has been estimated as 20.0 person-rem by MHI for the US-APWR ICIS design. This activity is assumed to occur only every other refueling. Given the 24-month refueling cycle currently assumed for the US-APWR, the dose has been adjusted to account for this activity occurring only every 4 years (every 2 cycles).

Table 12.4-8 Annual Personnel Doses per Activity Categories

Category	Reference tables	Estimated Annual Person-rem Exposure
Occupational Dose Estimates During Routine Operations and Surveillance	Table 12.4-1	0.77
Occupational Dose Estimates During Nonroutine Operations and Surveillance	Table 12.4-2	8.61
Occupational Dose Estimates During Routine Maintenance	Table 12.4-3	17.93
Waste processing	Table 12.4-4	5.63
Refueling	Table 12.4-5	8.74
Inservice Inspection	Table 12.4-6	11.6
Special maintenance	Table 12.4-7	17.75
Total		**71.03**

Table 12.4-9 Annual Occupational Doses Received At Comanche Peak Steam Electric Station – All Categories and Job Functions

	Maintenance		Operations		Health Physics		Supervisory		Engineering		Total person-rem
Year	Number of Personnel	person-rem	Number of Personnel	person-rem	Number of Personnel	person-rem	Number of Personnel	person-rem	Number of Personnel	person-rem	
2000	997	34.436	599	9.544	303	11.801	26	.055	559	31.423	**87.259**
2001	1082	47.968	562	8.935	368	12.312	18	.217	691	55.162	**124.594**
2002	1519	76.39	751	15.73	513	26.991	30	.329	922	122.747	**242.187**
Three-Year Average per plant											**75.673**
2003	845	24.243	533	7.629	289	8.552	17	.212	497	31.605	**72.241**
2004	1014	31.381	563	8.267	370	16.102	14	.17	655	80.447	**136.367**
2005	1666	67.556	652	11.607	536	23.43	20	.29	1222	139.065	**241.948**
Three-Year Average per plant											**75.093**
2006	1184	25.807	559	5.134	370	8.011	14	.115	629	22.002	**61.069**

12.5 Operational Radiation Protection Program

The COL Applicant is to describe the operational radiation protection program for ensuring that occupational radiation exposures are ALARA. This Combined Information is addressed in Subsection 12.1.4. This program will be based on RG 1.206 and any additional guidance developed by the industry and approved by the NRC.

The program consists of the following:

- A detailed management policy
- An organizational structure with clearly defined responsibilities
- Definition and description of all facilities, including laboratories and office spaces
- Definition and description of the monitoring instrumentation and equipment (note)
- Definition and description of the personnel protective clothing and equipment, including the necessary inventory of supplies
- Definition and description of other protective equipment, such as portable ventilation systems, temporary shielding, etc.
- Procedures on radiological surveillance
- Procedures on methods to maintain exposures ALARA
- Procedures on posting and labeling
- Procedures on access control
- Procedures on radiation work permits
- Procedures on personnel monitoring
- Procedures on dose control
- Procedures on contamination control
- Procedures on respiratory protection
- Procedures on radioactive material control
- Procedures on radiation protection training
- Quality assurance programs in effect

Note: This includes the personnel monitoring and radiation survey equipment, and laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.