DESIGN CONTROL DOCUMENT FOR THE
US-APWR
Chapter 12
Radiation Protection

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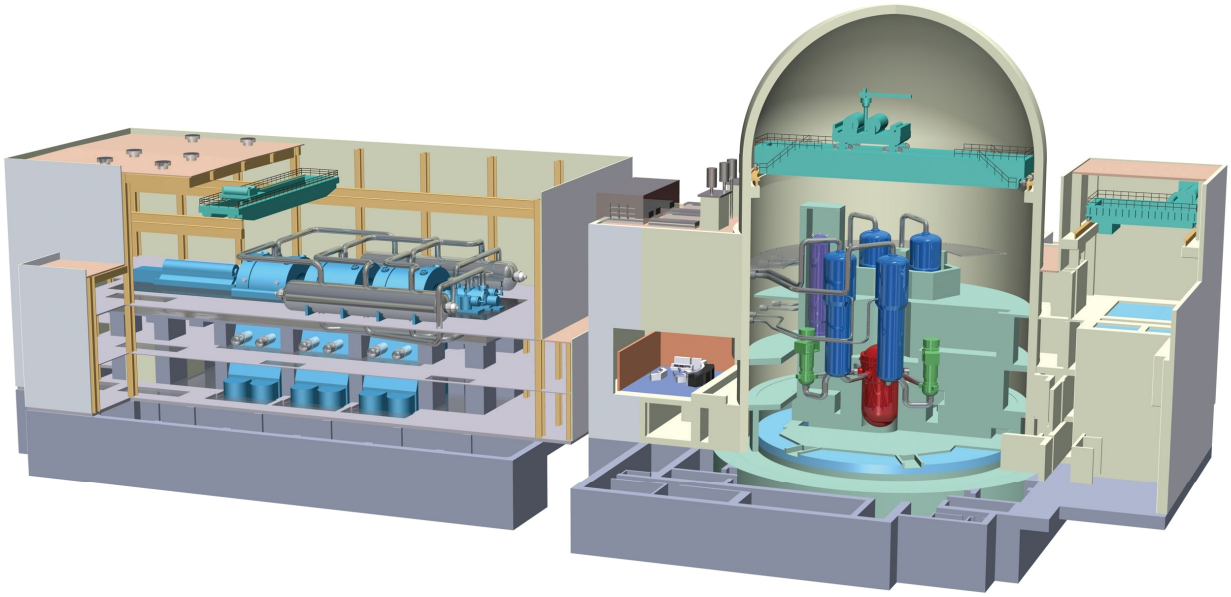

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CONTENTS

TABLES

Page

FIGURES

Page

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
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 anin.inch, inches
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
WMS	waste management system

12.0 RADIATION PROTECTION

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

US-APWR is to keep all radiation exposure of personnel within limits defined by Title 10, Code of Federal Regulations, Part 20 (Reference 12.1-1). Administrative procedures and practice in US-APWR related to maintaining radiation exposure of personnel as low as reasonably achievable (ALARA) are described below, referring to NEI 07-08 (Reference 12.1-2) submitted in August 2007 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 to personnel is kept ALARA. The organization of responsibilities for the basic design of US-APWR is intended to achieve ALARA occupational radiation exposures.

The manager of the section responsible for radiation protection engineering requires the ALARA design. The managers of related sections of Nuclear Energy Systems Engineering Center develop the design for ensuring ALARA, according to the requirement from the manager of the section that is responsible to radiation protection engineering.

12.1.1.1 Design Policies

The US-APWR is planned and designed taking into account the ALARA philosophy to reduce occupational radiation exposure in normal operation and order 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 is then reviewed, updated, and modified, as necessary, during the design phase and as experience is obtained from operating plants. Nuclear engineers with extensive experience in ALARA design and operation reviewed the plant design and 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 exposure to personnel.

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

12.1.1.2 Operation Policies

The Combined License (COL) Applicant will address the activities conducted by management personnel who have plant operational responsibility for radiation protection.

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 compliance of the US-APWR design with 10 CFR 20 (Reference 12.1-1) is ensured by the compliance of the design and operation of the facility within 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 out of the US-APWR Standard Plant scope. 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) that address 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 out of the US-APWR Standard Plant Scope. 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 out of the US-APWR Standard Plant scope. 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 othe 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 of means and adequate space to use movable shielding
- Separation of more highly radioactive equipment from less radioactive equipment and the provision of separate shielded compartments for adjacent items of radioactive equipment
- Provision for access to hatches to install or remove plant components
- Provision of design features to minimize crud buildup
- Countermeasures of design and water chemistry control to reduce radiation exposure such as the following:
  - 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 Li 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 crud behavior and buildup; (3) crud reduction method in normal operation; (4) dose rate distribution during operation and shut down; (5) revised radiation streaming behavior; (6) reduction of radiation streaming; (7) reduction of time needed time for maintenance and inspection; 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 to 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 to be informed to the engineering and/or research departments in charge of investigation for countermeasure as documents. Then the departments in charge will investigate design or operation procedure for countermeasure and review them with related departments. The approved new design or operation procedure will be reflected and applied to the plant in operation or under planning, and verified them in the field and reported back for 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 regarding ALARA design. The design procedures require that the component design engineer should 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. A review of the component designs was made for the modification of its design. 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), GDC 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 for greater volumetric flow than is supplied to that area.

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 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 tool has 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 its 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 its maintenance.

The equipment is designed with specific drainage to facilitate maintenance.

The equipment is designed with smooth surface to reduce the 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 were 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 their distribution in the piping systems, thereby reducing the potential 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 facilities to minimize the length of time spent in radiation areas include the following:

- 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 facilities to minimize radiation levels in plant access areas and near equipment requiring personnel attention include the following:

- 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 policy considerations regarding plant operations would be developed by the COL Applicant. See Subsection 12.1.4 for the COL information.

12.1.4 Combined License Information

COL 12.1 (1) The COL Applicant is to demonstrate compliance with RG 1.8 (Reference 12.1-3), 8.8 (Reference 12.1-4), and 8.10 (Subsection 12.1.1.3) (Reference 12.1-5).

COL 12.1 (2) The COL Applicant is to provide, to the level of detail provided in RG 1.70 (Reference 12.1-7), the criteria and/or conditions under which various operating procedures and techniques is to be provided to ensure that occupational radiation exposures ALARA are implemented (Subsection 12.1.3).

COL 12.1 (3) The COL Applicant is to describe how the plant follows the guidance of RG 8.2 (Reference 12.1-8), 8.4 (Reference 12.1-9), 8.6 (Reference 12.1-10), 8.7 (Reference 12.1-11), 8.9 (Reference 12.1-12), 8.13 (Reference 8.13), 8.15 (Reference 12.1-14), 8.20 (Reference 12.1-15), 8.25 (Reference 12.1-16), 8.26 (Reference 12.1-17), 8.27 (Reference 12.1-18), 8.28 (Reference 12.1-19), 8.29 (Reference 12.1-20), 8.32 (Reference 12.1-21), 8.34 (Reference 12.1-22), 8.35 (Reference 12.1-23), 8.36 (Reference 12.1-24), and 8.38 (Reference 12.1-25).

COL 12.1 (4) The COL Applicant is to describe the implementation of specific exposure control techniques.

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 07-08, Revision 0, Aug. 2007.

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 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition). RG 1.70, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, November 1978.

12.1-8 Guide for Administrative Practices in Radiation Monitoring. RG 8.2, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973.

12.1-9 Direct-Reading and Indirect-Reading Pocket Dosimeters. RG 8.4, U.S. Nuclear Regulatory Commission, Washington, DC, February 1973

12.1-10 Standard Test Procedure for Geiger-Müller Counters. RG 8.6, U.S. Nuclear Regulatory Commission, Washington, DC, May 1973

12.1-11 Instructions for Recording and Reporting Occupational Radiation Exposure Data. RG 8.7, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2005.

12.1-12 Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program RG 8.9, U.S. Nuclear Regulatory Commission, Washington, DC, July 1993.

12.1-13 Instruction Concerning Prenatal Radiation Exposure. RG 8.13, U.S. Nuclear Regulatory Commission, Washington, DC, June 1999.

12.1-14 Acceptable Programs for Respiratory Protection. RG 8.15, U.S. Nuclear Regulatory Commission, Washington, DC, October 1999.

12.1-15 Applications of Bioassay for I-125 and I-131. RG 8.20, U.S. Nuclear Regulatory Commission, Washington, DC, September 1999.

12.1-16 Air Sampling in the Workplace. RG 8.25, U.S. Nuclear Regulatory Commission, Washington, DC, June 1992.

12.1-17 Applications of Bioassay for Fission and Activation Products. RG 8.26, U.S. Nuclear Regulatory Commission, Washington, DC, September 1980.

12.1-18 Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants. RG 8.27, U.S. Nuclear Regulatory Commission, Washington, DC, March 1981.

12.1-19 Audible-Alarm Dosimeters. RG 8.28, U.S. Nuclear Regulatory Commission, Washington, DC, August 1981.

12.1-20 Instruction Concerning Risks from Occupational Radiation Exposure. RG 8.29, U.S. Nuclear Regulatory Commission, Washington, DC, February 1996.

12.1-21 Criteria for Establishing a Tritium Bioassay Program. RG 8.32, U.S. Nuclear Regulatory Commission, Washington, DC, July 1988

12.1-22 Monitoring Criteria and Methods To Calculate Occupational Radiation Doses. RG 8.34, U.S. Nuclear Regulatory Commission, Washington, DC, July 1992.

12.1-23 Planned Special Exposures. RG 8.35, U.S. Nuclear Regulatory Commission, Washington, DC, June 1992.

12.1-24 Radiation Dose to the Embryo/Fetus. RG 8.36, U.S. Nuclear Regulatory Commission, Washington, DC, July 1992.

12.1-25 Control of Access to High and Very High Radiation Areas of Nuclear Plants. RG 8.38, U.S. Nuclear Regulatory Commission, Washington, DC, May 2006.

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 to design personnel protection measures and dose assessment.

12.2.1 Contained Sources

The basis of the shielding design source terms are the three 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 features for 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 1mrem/h, when radiation penetrates the bulk shielding, and less than 100rem/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 a 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. Chapter 11 estimates activity in reactor coolant with both of design and realistic base. In shielding design, sources of reactor coolant are estimated with using design base methods using 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.

Fission and corrosion product activities circulating in the RCS and out-of-core crud deposits 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, Section 11.1. 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 crud deposits 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

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-s, half-life, and a transport time of greater than 1 minute before the primary coolant goes out of the containment.

12.2.1.1.3 Chemical and Volume Control System

Radiation sources in the chemical and volume control system (CVCS) consist of 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, other than the regenerative heat

exchanger, letdown heat exchanger, and excess letdown heat exchanger are located in the A/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 tube side of the regenerative heat exchanger account for the removal of radionuclides by the CVCS demineralizers and the volume control tank.

B. CVCS demineralizer

The mixed bed demineralizer is in continuous use and removes fission products in cation and anion form. 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 saturate 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 is provided to remove 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. The source term in the B.A. evaporator is based on the intermittent processing of the coolant.

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, Section 11.5. 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) are a result of transfer of radioactive isotopes from the reactor coolant into the SFP during refueling operations.

The reactor coolant activities for fission, corrosion, and activation products are decayed for the time required to remove the reactor vessel head following shutdown. They are reduced by operation of the CVCS purification demineralizers; and are 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 in the demineralizer.

However, 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). In a few decades operation history, fuel defect has reduced and fission products in Reactor Coolant are negligible today. And activities of corrosion

products are estimated as Cobalt-60 considering gamma emission energy for each nuclide. The dose rate at SFP water surface is 15 mrem/h caused by the radiation from both of the spent fuel assembly during fuel handling and the contaminated water in SFP.

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 those tabulated in Table 11.1-2. The treatment of SG secondary side water and steam, including fission and corrosion products, 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. 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 tabulate the distribution of the radioactive gas inventory and the gamma ray source strength associated with operation of the GWMS to be 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 this equipment are derived from cold shutdown procedures during which the radioactive gases are stripped from the RCS.

Since the gases are continuously re-circulated, the gamma ray source strengths for the waste gas compressor and waste gas surge tank are identical. Tables 12.2-46 and 12.2-47 tabulate the activities and gamma ray source strengths for the recirculation equipment.

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 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 allow the shielding afforded by the concrete tank walls to result in surface dose rates less than 0.25 mrem/h.

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) and to be consistent with the recommendations of RG 8.8 (Reference 12.2-2).

The SWMS facilities process and store dry active waste. Additional radwaste facilities for dry active waste will be provided by the COL Applicant. The radiation shielding will be provided such that the dose rates comply with the requirements of 10 CFR 20 (Reference 12.2-1). 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.

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; one region is loaded for one cycle and another is loaded for two cycles. Although the cycle duration is planned as 24 months, irradiation time for a cycle in source strength calculation is assumed as 28 months for the conservativeness. 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) are the spent fuel assemblies. The source strengths employed to determine the minimum water depth above spent fuel and shielding walls around the SFP, as well as shielding of the spent fuel transfer tube, are tabulated in Table 12.2-54. For the shielding design, the SFP is assumed to contain the design 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.

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 based on specific power of 32.1 MW/MTU and burnup of 62 GWD/MTU, which is a limitation for maximum burnup for fuel rod as described in chapter 4, section 4.2.1.

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, Section 4.2, a primary source rod contains californium-252 source, 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 drive 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.

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 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 in RG 1.183 (Reference 12.2-4). 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.

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 in needed in areas where airborne concentrations exceed the limit (such as containment during normal operation.), time will be adjusted and personal doses are in compliance with 10 CFR 20 (Reference 12.2-1).

12.2.2.1 Containment Vessel Atmosphere

The detailed listing of 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 detailed listing of 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 detailed listing of 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 detailed listing of 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
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)

$C_i(t)$ = Airborne concentration of the ith radioisotope at time t in the applicable region (μCi/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.

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 is responsible for the use of any additional contained radiation sources and airborne radioactive materials that are not identified in subsection 12.2.1 and 12.2.2, including radiation sources used for instrument calibration or radiography.

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.

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

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 Pipe Side Shell Side	66.9 65.9	63.0 434.2	Homogeneous Homogeneous	Source Water Source Water 22 wt%+ Secondary Water 9wt%+ Steel 69wt%	41.6 69.2	6.1 3.4	4
Regenerative heat exchanger * Pipe Side Shell Side	8.3	23.2 140.2	Homogenous Homogenous	Water (Charging Line) Water (Letdown Line) 35 wt%+ Water (Charging Line) 6 wt%+ Steel 59 wt%	62.4 129.2	2.0	3
Letdown heat exchanger Pipe Side Shell Side	17.7	24.4 189.8	Homogenous Homogenous	Source Water Source Water 11 wt%+ Cooling water 61 wt%+ Steel 28 wt%	62.4 82.5	ignored	1
Excess letdown heat exchanger Pipe Side Shell Side	6.9	21.7 130.2	Homogenous Homogenous	Source Water Source Water 5 wt%+ Cooling water 63 wt%+ Steel 32wt%	62.4 86.9	1.8 ignored	1

* The regenerative heat exchanger consists of three shells.

Table 12.2-1 Radiation Sources Parameters (Sheet 2 of 4)

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 Pipe Side Shell Side	31.5	 56.7 264.4	 Homogenous Homogenous	 Source Water Source Water 15 wt%+ Cooling water 48 wt%+ Steel 37 wt%	 62.4 91.8	 1.8 1.2	4
Seal water heat exchanger Pipe Side Shell Side	8.4	 22.3 144.6	 Homogenous Homogenous	 Source Water Source Water 10 wt%+ Cooling water 49 wt%+ Steel 41 wt%	 62.4 97.6	ignored	1
Volume control tank Liquid Phase Vapor Phase	47.2	179.2 107.5 71.7	 Homogenous Homogenous	 Air Water	 7.6E-02 62.4	ignored	1

Table 12.2-1 Radiation Sources Parameters (Sheet 3of 4)

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
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
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
Plant Yard Area (Outside the Power Block)							
Refueling water auxiliary tank	236.2	536.6	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 4 of 4)

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 12.2-2 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 Activity

Position in Loop	Loop Transit Time (s)	Nitrogen-16 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	Activity (µCi/g)	Nuclide	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 Crud Deposits

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 ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
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 ($\mu Ci/cm^3$)	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 12.2-12 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 ($\mu Ci/cm^3$)	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	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 Specific 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	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 ($\mu Ci/cm^3$)
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	1.0E+01
0.8	3.0E+01
1	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	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 (μCi/cm^3)	Nuclide	Activity (μCi/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	5.7E-03
Refueling Water Storage Auxiliary Tank source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	3.5E-04
0.3	4.8E-03
0.8	1.3E-02
1.0	2.1E+02
1.5	3.2E+02
2.0	4.6E-03
3.0	2.3E-05

Table 12.2-51 Miscellaneous Sources – Primary Makeup Water Tank

Primary Makeup Water Tank activity	
Nuclide	Activity ($\mu Ci/cm^3$)
Co-60	9.5E-04

Primary Makeup Water Tank source strength	
Gamma Ray Energy (MeV)	Source Strength ($MeV/cm^3/sec$)
0.015	5.8E-05
0.3	8.1E-04
0.8	2.1E-03
1.0	3.5E+01
1.5	5.3E+01
2.0	7.8E-04
3.0	3.8E-06

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

Nuclide	Activity ($\mu Ci/cm^3$)	Nuclide	Activity ($\mu Ci/cm^3$)
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)

Energy Group (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)

Energy Group (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)

Energy Group (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)

Energy Group (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 (Sheet 1 of 3) Parameters and Assumptions for Calculating Airborne Radioactive Concentrations (Containment)

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) 0.01(halogen&tritium) 0.001(others)
Fuel defect	1%
Low volume purge flow rate	(in normal operation) 2000 cfm
High volume purge flow rate	(in refueling) 30000 cfm

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

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) 0.01(halogen&tritium) 0.001(others)
Fuel defect	1%
Flow rate	24000 cfm

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

Parameter/ Assumption	Value
Reactor coolant leak rate in refueling (Note)	100 lb/d (for Radiation Zone V to VI) 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) 0.01(halogen&tritium) 0.001(others)
Fuel defect	1%
Flow rate	1500 cfm(for Radiation Zone V to VI) 14000 cfm (for Radiation Zone IV) 76000 cfm (for Radiation Zone III)

(Note) Reactor Coolant leak rates were derived from leak from valves in consideration. Each Radiation Zone has deferent number of radioactive valves. Zone V or higher have many component cubicles and valve galleries, and these zones have many radioactive valves. Zone IV have relatively high radiation level corridors, and have not so many radioactive valves. Zone III have low radiation level corridors and access areas, and have fewer radioactive valves. As a result, leak rate in Zone V or higher is high, one in Zone IV is middle and one in Zone III is rather low.

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

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 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) The area in the reactor containment isn't usually entered.

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

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)
Kr-83m	1.4E-11	1E-02	Ru-106	-	5E-09
Kr-85m	1.1E-10	2E-05	Ag-110m	-	4E-08
Kr-85	2.6E-06	1E-04	Te-125m	-	2E-07
Kr-87	7.1E-16	5E-06	Te-127m	-	1E-07
Kr-88	1.2E-11	2E-06	Te-127	7.8E-13	7E-06
Xe-131m	2.2E-07	4E-04	Sb-129	-	4E-06
Xe-133m	1.4E-07	1E-04	Te-129m	1.4E-16	1E-07
Xe-133	1.6E-05	1E-04	Te-129	-	3E-05
Xe-135m	2.0E-09	9E-06	Sb-131	-	1E-05
Xe-135	9.4E-08	1E-05	Te-131m	2.2E-16	2E-07
Xe-138	-	4E-06	Te-131	-	2E-06
I-130	1.5E-13	3E-07	Te-132	3.4E-15	9E-08
I-131	3.4E-12	2E-08	Cs-132	9.2E-16	2E-06
I-132	4.2E-10	3E-06	Te-133m	-	2E-06
I-133	2.9E-12	1E-07	Te-133	-	9E-06
I-134	-	2E-05	Cs-134	8.4E-13	4E-08
I-135	3.4E-13	7E-07	Te-134	-	1E-05
Br-82	1.3E-16	2E-06	Cs-135m	-	8E-05
Br-83	-	3E-05	Cs-135	-	5E-07
Br-84	-	2E-05	Cs-136	2.1E-13	3E-07
Rb-86	8.0E-15	3E-07	Cs-137	4.8E-13	6E-08
Rb-87	-	6E-07	Cs-138	-	2E-05
Rb-88	1.3E-14	3E-05	Ba-140	-	6E-07
Rb-89	-	6E-05	La-140	3.3E-14	5E-07
Sr-89	-	6E-08	La-141	-	4E-06
Sr-90	-	2E-09	Ce-141	-	2E-07
Y-90	4.4E-14	3E-07	Ce-143	-	7E-07
Sr-91	-	1E-06	Pr-143	9.5E-16	3E-07
Y-91m	1.5E-15	7E-05	Ce-144	-	6E-09
Y-91	-	5E-08	Pr-144	1.5E-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-16	2E-06
Zr-95	-	5E-08	Cr-51	-	8E-06
Nb-95m	-	9E-07	Mn-54	-	3E-07
Nb-95	1.3E-15	5E-07	Mn-56	-	6E-06
Mo-99	8.6E-15	6E-07	Fe-55	-	8E-07
Tc-99m	5.7E-12	6E-05	Fe-59	-	1E-07
Tc-99	-	3E-07	Co-58	1.5E-16	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(Sheet 3 of 6) Airborne Radioactive Concentrations (Fuel Handling Area)

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)
Kr-83m	1.3E-11	1E-02	Ru-106	-	5E-09
Kr-85m	1.0E-10	2E-05	Ag-110m	-	4E-08
Kr-85	2.4E-06	1E-04	Te-125m	-	2E-07
Kr-87	6.5E-16	5E-06	Te-127m	-	1E-07
Kr-88	1.1E-11	2E-06	Te-127	7.2E-13	7E-06
Xe-131m	2.0E-07	4E-04	Sb-129	-	4E-06
Xe-133m	1.3E-07	1E-04	Te-129m	1.3E-16	1E-07
Xe-133	1.4E-05	1E-04	Te-129	-	3E-05
Xe-135m	1.9E-09	9E-06	Sb-131	-	1E-05
Xe-135	8.6E-08	1E-05	Te-131m	2.0E-16	2E-07
Xe-138	-	4E-06	Te-131	-	2E-06
I-130	1.3E-13	3E-07	Te-132	3.1E-15	9E-08
I-131	3.1E-12	2E-08	Cs-132	8.4E-16	2E-06
I-132	3.8E-10	3E-06	Te-133m	-	2E-06
I-133	2.7E-12	1E-07	Te-133	-	9E-06
I-134	-	2E-05	Cs-134	7.7E-13	4E-08
I-135	3.1E-13	7E-07	Te-134	-	1E-05
Br-82	1.2E-16	2E-06	Cs-135m	-	8E-05
Br-83	-	3E-05	Cs-135	-	5E-07
Br-84	-	2E-05	Cs-136	1.9E-13	3E-07
Rb-86	7.3E-15	3E-07	Cs-137	4.4E-13	6E-08
Rb-87	-	6E-07	Cs-138	-	2E-05
Rb-88	1.2E-14	3E-05	Ba-140	-	6E-07
Rb-89	-	6E-05	La-140	3.0E-14	5E-07
Sr-89	-	6E-08	La-141	-	4E-06
Sr-90	-	2E-09	Ce-141	-	2E-07
Y-90	4.1E-14	3E-07	Ce-143	-	7E-07
Sr-91	-	1E-06	Pr-143	8.8E-16	3E-07
Y-91m	1.4E-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	2.9E-16	2E-06
Zr-95	-	5E-08	Cr-51	-	8E-06
Nb-95m	-	9E-07	Mn-54	-	3E-07
Nb-95	1.2E-15	5E-07	Mn-56	-	6E-06
Mo-99	7.9E-15	6E-07	Fe-55	-	8E-07
Tc-99m	5.2E-12	6E-05	Fe-59	-	1E-07
Tc-99	-	3E-07	Co-58	1.4E-16	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	1.9E-13	5E-04			

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

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 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(Sheet 5 of 6) Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone IV)

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/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(Sheet 6 of 6) Airborne Radioactive Concentrations (Reactor Building and Auxiliary Building; Radiation Zone III)

Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/cm^3)	Nuclide	Activity (μCi/cm^3)	10CFR20 DAC (μCi/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			

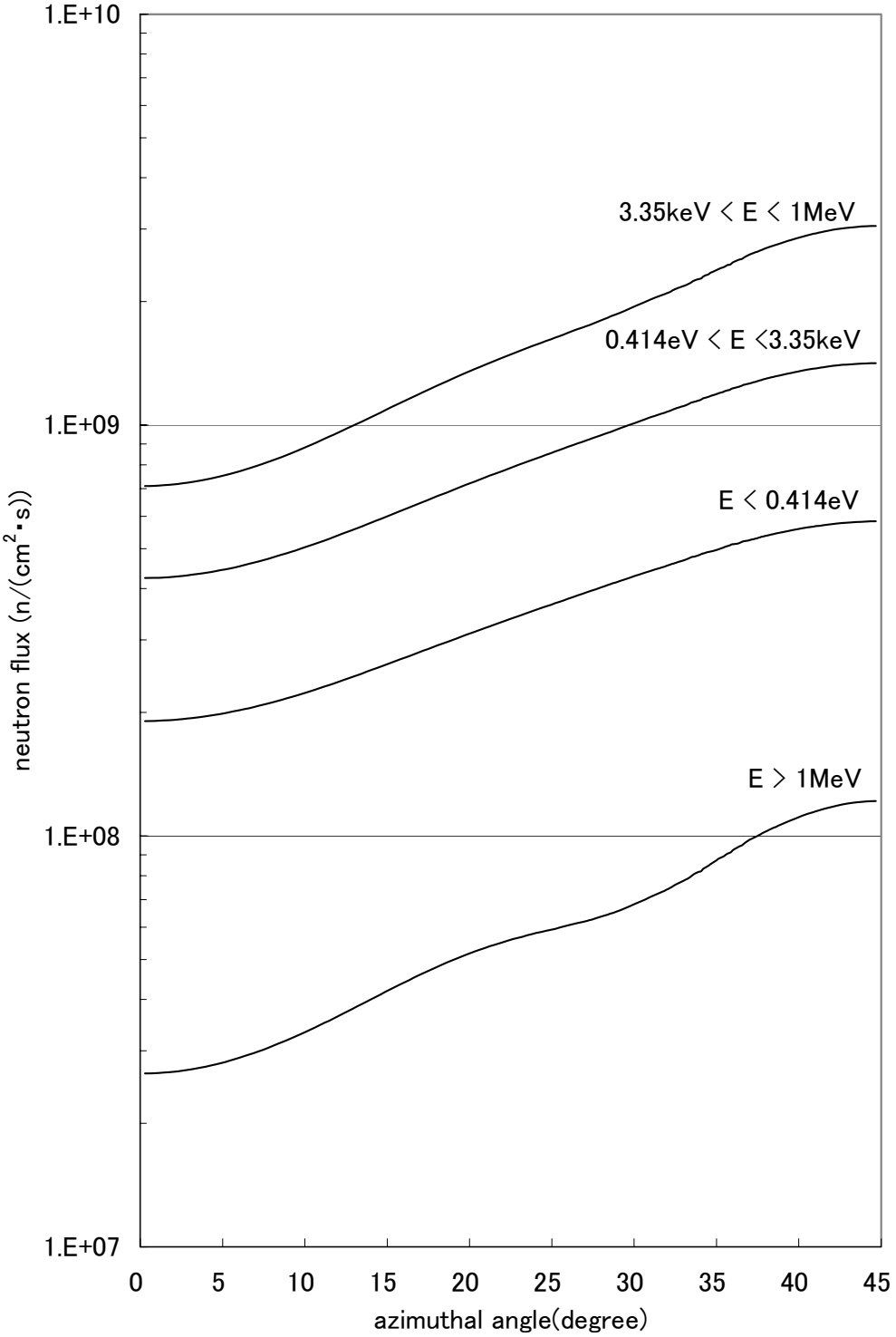

Figure 12.2-1 Azimuthal Distribution of Neutron Flux Incident on the Primary Shield at the Reactor Core Midplane

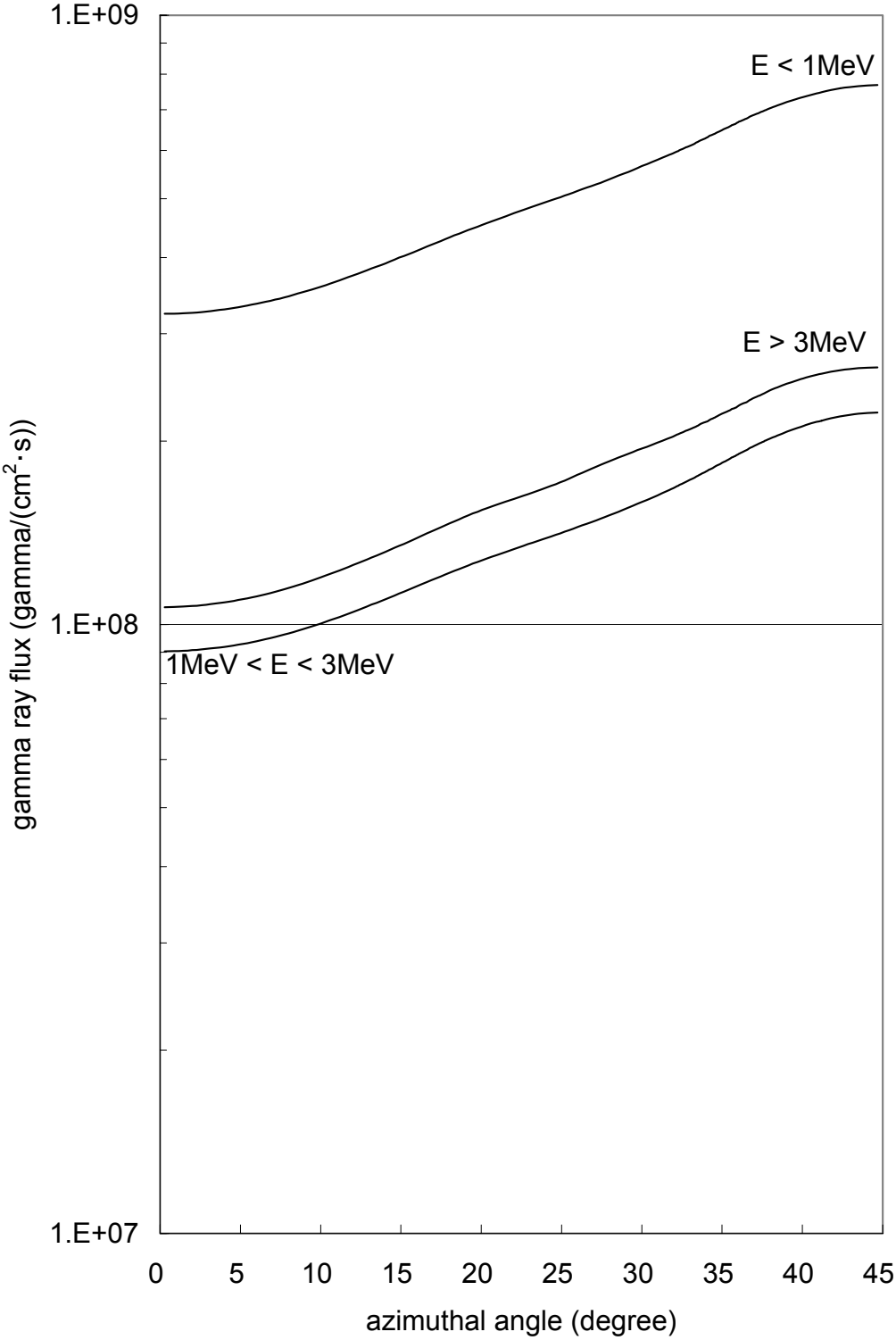

Figure 12.2-2 Azimuthal Distribution of Gamma Ray Energy Flux Incident on the Primary Shield at the Reactor Core Midplane

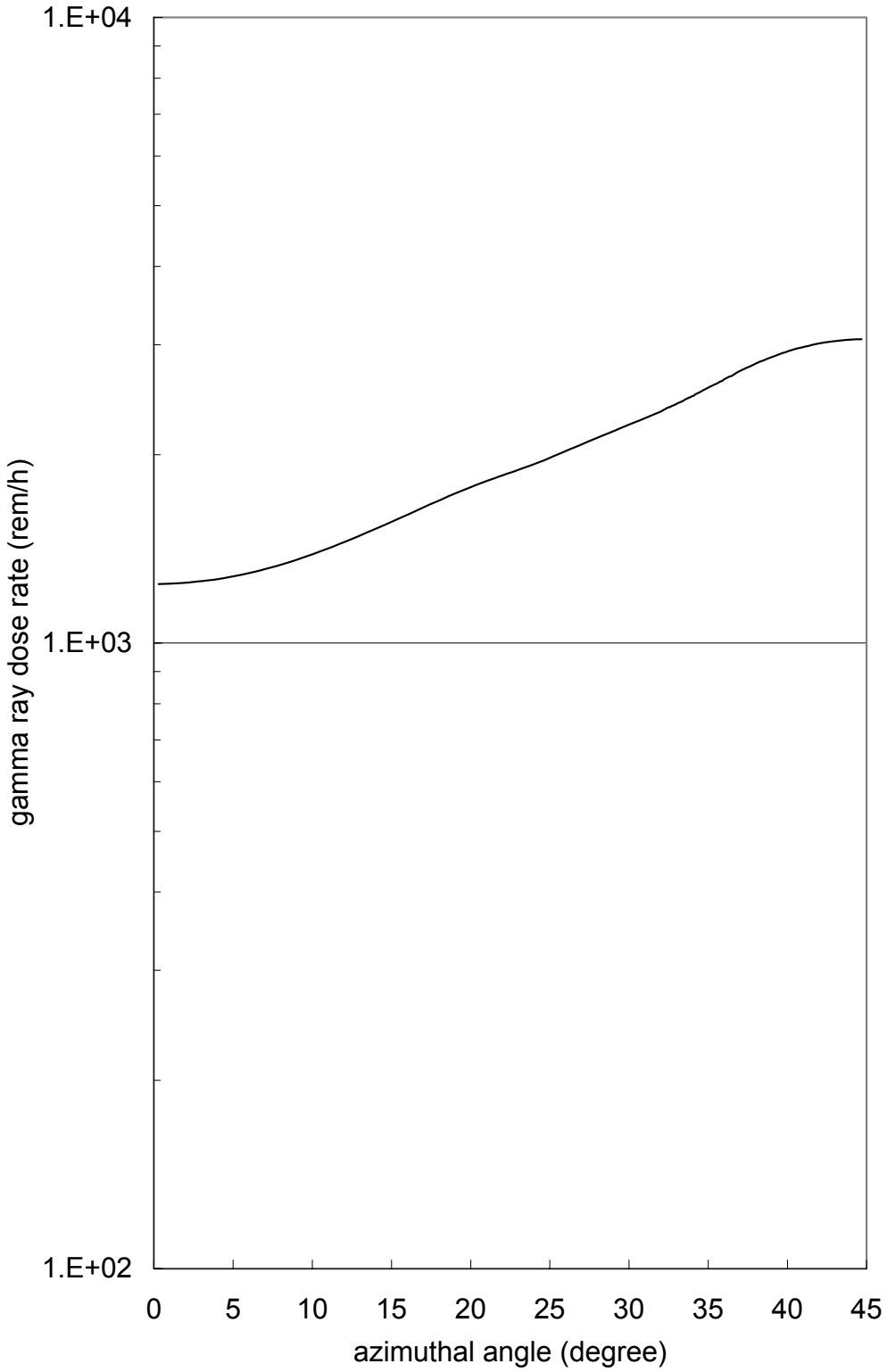

Figure 12.2-3 Azimuthal Distribution of Gamma Ray Dose Rate Incident on the Primary Shield at the Reactor Core Midplane

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.

12.3.1.1.1.2 Balance of Plant Equipment.

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.

E. Tanks

Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines 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.

F. Heat Exchangers

Heat exchangers 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.

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 on 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. Metal diaphragm or bellows seat valves are used on those systems where essentially no leakage can be tolerated.

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.

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 are used in conjunction with the general equipment described in Subsection 12.3.1.1.1, and include that 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 12.4).

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, and 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 compartment 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.

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 12.4). 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/shutdown conditions are shown in Figure 12.3-1.

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 12.4.

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 in. 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 in. 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 or from any surface that the radiation penetrates 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 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 limitation to and from radiologically controlled areas (RCAs):

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 Control 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) Principles of access control:

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.

(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, contamination is checked for radioactive materials on body surfaces with a radiation monitor, and persons are required to report to the radiation control engineer and follow instructions if any contamination is found.

e. When carrying materials out of the RCA, it is verified by measurement that the external radiation and the surface contamination of the material does not exceed the values related to controlled areas

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 12.4.

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), postaccident 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 mission dose for the vital areas at various times after an accident are given in Table 12.3-3. 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 12.4. 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.

12.3.2 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) 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 Chemistry composition of Type04 that are recommended in ANSI/ANS 6.4-1997.

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 containment wall is a reinforced, pre-stressed concrete structure, surrounding the nuclear steam supply system. The wall is of a minimum thickness of 4 ft 4 in. and the dome is of a minimum thickness of 3 ft 8 in.

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 X.

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 mass of steel plate reinforced concrete that surrounds the reactor vessel and extends upward from the containment floor. The minimum concrete thickness of the primary shield is 9 ft 2 in. The primary shield meets the following objectives:

- In conjunction with the secondary shield to reduce the radiation level from sources within the reactor vessel and the RCS to allow a limited access to the containment during normal, full-power operation.
- The gap between the reactor vessel and the primary shield wall has been designed to minimize neutron streaming to the containment vessel free volume. In addition, this gap region includes a labyrinth arrangement to minimize further streaming.
- After shutdown, limits of the radiation level from sources within the reactor vessel, allowing remote inspection through penetration and limited access to the RCS equipment.
- Limits of neutron activation of component and structural materials.

The secondary shield consists of steel plate reinforced concrete that surrounds 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 ft.

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 (Section 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 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 pit and the fuel transfer canal. 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.

12.3.2.2.3 Reactor Building Shielding Design

During normal operations, the major components in the reactor containing radioactivity are the RHR, containment spray, safety injection, and charging systems. Under accident conditions, these will contain high levels of radioactivity. 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 Section 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 ft 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 fuel racks to the minimum water level in the spent fuel pit is 29 ft. The dose rate with 29 ft 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 ft 1 in. in the refueling cavity and 11 ft 1 in. in the fuel transfer canal and spent fuel pool. 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 ft 1 in.-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 shielding (Chapter 9, Section 9.1) 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.

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. Control room ventilation system parameters are provided in Chapter 6, Section 6.4. 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.

Administration control of inspection access for fuel transfer tube and access control the access area near seismic gap below fuel transfer tube is to be explained in section 4 of the COL document.

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. Access to the outside storage tanks that have a contact dose rate greater than 0.25 mrem/h is restricted by a fence so that dose rates to personnel in the plant yard areas are limited to less than 0.25 mrem/h.

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 under the plant operating conditions 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) will be satisfied in control room following those hypothetical accidents described in Chapter 15.
- 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.

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.

- 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 minimizes the amount of uncontrolled exfiltration 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 1, Section 1.9, 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 1, Section 1.9.

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, as noble gases and iodine in the reactor coolant, and is partially released into the air when a reactor coolant leak occurs in the reactor 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 (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 1, Section 1.9.

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 not exceed dose limit for access area. Therefore, no shielding is provided and occupational exposure for the filter elements replacement will not be level of sufficient magnitude.
- During and post accident conditions, filters and absorbers in air purification system of control room and technical support center, and filters in annulus air purification system are in operation. The level of radioactivity and dose rate developed during 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, so 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.
- HEPA filter banks that are more than three filter units high, where each filter is 2 ft by 2 ft have a platform to facilitate access to the upper filters.
- 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 Systems (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 ARMS and Airborne Radioactivity Monitoring System supplement the personnel and area radiation survey provisions of the plant health physics program described in Section 12.4 and assure compliance with the personnel radiation protection guidelines of 10 CFR 20 (Reference 12.3-2), 10 CFR 50 (Reference 12.3-7), 10 CFR 70 (Reference 12.3-21), and 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) and ANSI N13.1-1999 (Reference 12.3-25).

The design of the fuel pool 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).

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 ray 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-91, RMS-RE-92, RMS-RE-93, and RMS-RE-94). 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 and occasionally accessible where 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 and detecting the presence of fission products due to design basis accident need to be detected

- 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 area monitors are installed in the following locations:

(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)
(i) TSC

Furthermore, portable ARMSs are installed in the following locations during the work:

(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-91, RMS-RE-92, RMS-RE-93, RMS-RE-94.

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 and, with the exception of the containment area monitors, indicated and alarmed to the personnel near the detector location and the WMS Area Monitor, which alarmed to the personnel in the A/B control room. Monitors RMS-RE-91, RMS-RE-92, RMS-RE-93, RMS-RE-94, 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.

Alarm setpoints are controlled by plant procedures.

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 operators 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 air exhausted from cubicles HVAC exhaust ducts
- 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. 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

The sampling points of the airborne radioactivity monitors are shown in the Figure 12.3-10. Details of design will be developed and, it is to be described in Combined License Information and verified it in ITAAC.

12.3.4.2.3 General System Description

The system description of airborne radioactivity monitors is 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 no local annunciation.

12.3.4.2.6 Power Supplies

Each channel-installed rack is provided with a common power supply. However, 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-49)

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-46)

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-48A)

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-48B)

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-48C)

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

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, the plant operator (COL Applicant) will have a radiation protection program in place to assure that radiation exposures will be within limits and ALARA (see Section 12.4).

In-plant radiation exposures during normal operation and anticipated operational occurrences are incurred from the following activities (Reference 12.3-29):

- 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 12.3-30). The latest edition of NUREG-0713 (Reference 12.3-30) (Vol. 27) includes data through 2005. Table 4.2 of NUREG-0713 (Reference 12.3-30) 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 has led to fewer refueling outages. Prior to 1990, annual refueling outages were not uncommon. The US-APWR will be on a 24-month cycle.
- 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 12.3-7 thorough 12.3-14 are based on RG 8.19 "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates," Rev 0, 1978 (Reference 12.3-29). Table 12.3-14 is a summation of the US-APWR's individual dose estimates in person-rem/year. The total annual station exposure is 68.63 person-rem, which is substantially less than the 100 person-rem annual value cited in NUREG -0713 (Reference 12.3-30).

Because of extended fuel cycles (18-24 month), 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 12.3-30), 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 12.3-15. 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 75.76 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 should 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 12.3-30), RG 8.19 (Reference 12.3-29), the Comanche Peak Steam Electric Station data, the US-APWR design, and the COL Applicant's ALARA program, it is expected that operation of the US-APWR will result in radiation exposures of less than 100 person-rem per year.

12.3.5.1 Occupational Radiation Exposure

Radiation exposures to operating personnel are restricted to the limits of 10 CFR 20 (Reference 12.3-2). The health physics program described in Section 12.4 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 12.3-2).

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 12.3-29):

- 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.

12.3.5.1.1 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 12.3-7 and 12.3-8 provide a breakdown of the collective doses for routine and non-routine operations and surveillance respectively. The occupational radiation exposure above operation floor in the containment vessel is reduced by the labyrinth structure between reactor vessel and primary shielding.

12.3.5.1.2 Routine Maintenance

Routine maintenance is required for mechanical and electrical components. Table 12.3-9 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.

12.3.5.1.3 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 12.3-10.

12.3.5.1.4 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 12.3-11 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 12.3-26), are not needed. Long-term refueling cycle of US-APWR (maximum 24 month) reduce the annual occupational radiation exposure.

12.3.5.1.5 In-service Inspection

American Society of Mechanical Engineers Code, Section XI (Reference 12.3-31) 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 12.3-12 provides the dose estimates for ISI activities. The

insulation with shielding materials is used for 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.

12.3.5.1.6 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 12.3-13 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.

12.3.5.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the seven activity categories discussed above are summarized in Table 12.3-14.

12.3.5.1.8 Post-Accident Actions

Requirements of 10 CFR 52.79(b) (Reference 12.3-32) relative to the plant area access and the post-accident sampling, 10 CFR 50.34(f)(2)(viii) (Reference 12.3-33), 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-dependency of the area dose rates and the required post-accident access times. 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.

12.3.5.2 Radiation Exposure at the Site Boundary

12.3.5.2.1 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.

12.3.5.2.2 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.

12.3.6 Combined License Information

COL 12.3(1) The COL Applicant is responsible for 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.

COL 12.3(2) The COL Applicant is to provide more detailed air controlling process and airborne radioactivity monitor in Figure 12.3-10.

COL 12.3(3) 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).

12.3.7 References

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12.3-9 Concrete Radiation Shields for Nuclear Power Plants. RG 1.69, U.S. Nuclear Regulatory Commission, Washington, DC, December 1973.

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, 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.

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12.3-24 Guide for Administrative Practices in Radiation Monitoring. Regulatory Guide 8.2, 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 Nastional Standard Institute N13.1-1997.

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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'-4"	2'-0"
50'-2''	Regenerative heat exchanger Room	1'-2"	2'-0"	4'-0"	2'-0"	1'-4"	2'-4"
50'-2"	Excess letdown heat exchanger Room	2'-0"	2'-0"	4'-0"	2'-0"	1'-4"	2'-0"
Reactor Building							
-26'-4"	A-Charging pump Room	3'-8"	3'-8"	3'-2"	3'-2"	Ground	14'-0"
-26'-4"	B-Charging pump Room	3'-8"	3'-2"	3'-2"	3'-2"	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"	-	3'-4"	3'-2"	1'-10"	3'-2" 3)

1) Face to Safeguard component area AHU room
2) Upstairs of B-Charging pump room
3) Face to corridor

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"	3'-6"	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'-10"	3'-10"	3'-4"	3'-10"
3'-7"	Piping Area 3)	3'-4"	3'-10"	3'-10"	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"	5'-3"	2'-4"	5'-3"
25'-3"	Piping Area 6)	3'-0"	7'-1"	3'-10"	3'-10"	2'-4"	3'-10"
25'-3"	Main Control Room	3'-4"	3'-4"	3'-4"	3'-4"	3'-4"	3'-4"
25'-3"	A-Piping Penetration Area	3'-10"	3'-10"	4'-2"	4'-4"	3'-4"	3'-10"
25'-3"	B-Piping Penetration Area	4'-2"	3'-10"	4'-2"	3'-4"	3'-4"	3'-10"
25'-3"	C-Piping Penetration Area	4'-2"	2'-8"	3'-10"	3'-10"	3'-4"	3'-10"
25'-3"	D-Piping Penetration Area	3'-4"	-	4'-2"	3'-10"	3'-4"	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) Adjacent to Volume control tank 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	2’-0”	4’-0”	4’-2”	3’-4”	Ground	4’-2”
-26’-4”	B-Holdup tank Room	2’-0”	4’-0”	2’-0”	3’-4”	Ground	4’-2”
-26’-4”	C-Holdup tank Room	4’-0” 1)	4’-0”	2’-0”	3’-4”	Ground	4’-2”
-26’-4”	A-Waste holdup tank Room	2’-0”	3’-4”	2’-6”	2’-0”	Ground	2’-8”
-26’-4”	B-Waste holdup tank Room	2’-0”	3’-4”	2’-0”	2’-0”	Ground	2’-8”
-26’-4”	C-Waste holdup tank Room	2’-0”	2’-6” 1)	2’-6”	3’-4”	Ground	2’-10”
-26’-4”	D-Waste holdup tank Room	3’-4”	2’-0”	2’-0”	3’-4”	Ground	2’-10”
-26’-4”	Charcoal bed Room (A)	2’-6”	2’-6”	4’-4”	2’-6”	Ground	2’-6”
-26’-4”	Charcoal bed Room (B)	2’-6”	2’-6”	4’-4”	2’-6”	Ground	2’-6”
-26’-4”	Waste gas surge tank Room (A)	3’-4”	3’-4”	3’-11” 1)	2’-8”	Ground	2’-8”
-26’-4”	Waste gas surge tank Room (B)	3’-4”	2’-8”	3’-11” 1)	3’-4”	Ground	2’-8”
-26’-4”	A-Spent resin storage tank Room	3’-2”	2’-8”	3’-4”	4’-6”	Ground	3’-2”
-26’-4”	B-Spent resin storage tank Room	4’-6”	2’-8”	3’-2”	4’-6”	Ground	3’-2”
-26’-4”	Valve Area 2)	2’-2”	2’-6”	2’-6”	2’-6”	Ground	2’-11” 1)
-26’-4”	Valve Area 3)	2’-7”	3’-4”	2’-8”	3’-4”	Ground	2’-8”
-26’-4”	Valve Area 4)	3’-6”	3’-6”	3’-4”	3’-2”	Ground	3’-2”
-8’-7”	Piping Area 4)	3’-6”	3’-6”	4’-6” 5)	3’-4”	3’-6”	3’-6”
-8’-7”	Piping Area 6)	3’-4”	3’-4”	3’-4”	4’-0”	2’-8”	2’-8”

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

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'-2" 1)	4'-8"
3'-7"	Cation-bed demineralizer Room	3'-4"	2'-10"	2'-10"	4'-0"	2'-10"	4'-8"
3'-7"	Spent fuel pit demineralizer Room	2'-10"	2'-0"	2'-0"	3'-4"	3'-2"	3'-4"
3'-7"	Valve Area 2)	2'-10"	4'-2"	4'-2" 3)	3'-4" 4)	3'-2" 5)	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	2'-0"	2'-8"	2'-0"	2'-0"	2'-8"	2'-2"
3'-7"	A-Spent fuel filter Room	1'-6"	2'-2"	2'-0"	2'-0"	2'-8"	2'-2"
3'-7"	B-Spent fuel filter Room	1'-6"	2'-2"	1'-6"	2'-0"	2'-8"	2'-2"
3'-7"	Steam generator blowdown demineralizer Room	2'-10"	2'-2"	3'-4"	2'-6"	2'-8" 3)	3'-4"
3'-7"	Valve Area 6)	2'-10"	2'-2"	3'-4"	2'-6"	2'-8" 3)	3'-4"
3'-7"	A,B-Waste demineralizer Room	3'-8"	2'-5"	2'-5"	2'-5"	2'-6"	4'-0"
3'-7"	C,D-Waste demineralizer Room	3'-2"	2'-5"	1'-10"	3'-0"	2'-6"	4'-0"
3'-7"	Valve Area 7)	2'-5" 8)	2'-8"	2'-8"	2'-8"	2'-6"	3'-4"
13'-6"	Piping Area 9)	2'-6"	3'-4"	2'-1"	3'-4"	2'-8"	3'-4"
13'-6"	Hold up Tank Piping Area	3'-4"	4'-8" 4)	3'-4"	2'-8" 3)	2'-8"	3'-4"
13'-6"	Hold up Tank Valve Area	3'-4"	4'-8" 4)	3'-4"	4'-0"	2'-8"	3'-4"

1) Face to Spent resin storage tank Room
2) Adjacent to Mix bed demineralizer Room
3) Face to area of Zone III
4) Face to A,B-Mix bed demineralizer Room
5) Face to Valve Area
6) Adjacent to Steam generator blowdown demineralizer Room
7) Adjacent to Waste demineralizer Room
8) Face to A,B-Waste demineralizer Room
9) East side of demineralizer Rooms

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

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

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.	2.4E-02	1.0E-03	2.4E-05	Figure 12.3-3 Sheet 3,4,5
		7.7E-03	1.0	7.7E-03	
		Total		7.7E-03	
	Return to MCR from Radiochemical laboratory	4.6E-02	2.5E-03	1.2E-04	Figure 12.3-3 Sheet 3,4,5
		7.7E-03	1.0	7.7E-03	
		Total		7.8E-03	
Technical support center (TSC)	Access to TSC from AC/B for operation.	2.3E-02	1.0E-03	2.3E-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)	3.3E-02	2.5E-03	8.1E-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 (sheet 3 of 3) Mission Dose for the Vital Areas access route after an Accident (1day to 1mounth after)

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.	2.4E-02	1.0E-03	2.4E-05	Figure 12.3-4 Sheet 3,4,5
		7.7E-03	1.5E-02	1.2E-04	
		Total		1.4E-04	
	Return to MCR from Radiochemical laboratory	4.6E-02	2.5E-03	1.2E-04	Figure 12.3-4 Sheet 3,4,5
		7.7E-03	1.5E-02	1.2E-04	
		Total		2.4E-04	
Technical support center (TSC)	Access to TSC from AC/B for operation.	2.3E-02	1.0E-03	2.3E-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)	3.3E-02	2.5E-03	8.1E-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 12.3-4 Area Radiation Monitors

Detector	Type	Service	Nominal Range	Safety-Related	Quantity	Control Function
RMS-RE-1	Gamma ray	MCR Area Radiation Monitor	1E-5 to 1E+0 R/h	No	1	None
RMS-RE-2	Gamma ray	Containment Air Lock Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-3	Gamma ray	Radio Chemical Lab. Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-5	Gamma ray	SFP Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-6	Gamma ray	Nuclear Sampling Room Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-7	Gamma ray	ICIS Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-8	Gamma ray	WMS Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-9	Gamma ray	TSC Area Radiation Monitor	1E-4 to 1E+1 R/h	No	1	None
RMS-RE-91A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation
RMS-RE-92 A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation
RMS-RE-93 A,B	Gamma ray	Containment High Range Area Radiation Monitor	1E+0 to 1E+7 R/h	Yes	1	Containment Ventilation Isolation
RMS-RE-94 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-49	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-46	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-48A	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-48B	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-48C	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 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)
Walking	0.2	0.5	2	1/shift	**0.22**
Checking:					
Containment cooling system	1	1	1	1/day	**0.36**
Accumulators	1.5	1	1	1/day	**0.54**
Pressurizer valves	10	0.2	1	1/day	**0.73**
Boron acid (BA) 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					**2.37**

Table 12.3-8 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 12.3-9 Occupational Dose Estimates During Routine Maintenance

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rem/y)
Mechanical: Changing filters:					
Waste filter	100	0.5	1	6/year	**0.3**
Laundry filter	100	0.5	1	10/year	**0.5**
Boron Acid filter	100	0.5	1	2/year	**0.1**
Mechanical components	10	100	5	1/2year	**2.5**
Electrical components	5	200	6	1/2year	**3**
Decompositions of valves	50	40	4	1/2year	**4**
Reactor Coolant pump	10	100	4	1/2year	**2**
Other pumps	0.25	100	4	1/2year	**0.05**
Others:					
Radiation management	5	1	2	1/day	**3.65**
Washing / decontamination	1	1	3	1/day	**1.1**
Operation of equipment	1	1	2	1/day	**0.73**
Total					**17.93**

Table 12.3-10 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 12.3-11 Occupational Dose Estimates During Refueling

Activity	Average dose rate (mrem/h)	Exposure time (h)	Number of workers	Frequency	Dose (person-rems/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**

* MHI has estimated the total dose for **reactor pressure vessel head and internals-removal and installation including** servicing of the top-mounted ICIS as 16.0 person-rems, which on an 24-month refueling cycle results in an equivalent person-rem/year of 1/2 x 16.0 = 8.0 person-rems.

Table 12.3-12 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 12.3-13 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	50	40	10	1/2year	**10.0**
Inspection of RCP internal	100	90	4	1/10year	**3.6**
Servicing of detectors around core	10	10	3	1/2year	**0.15**
Total					**13.75**

Table 12.3-14 Annual Personnel Doses per Activity Categories

Category	Reference tables	Estimated Annual Person-rem Exposure
Occupational Dose Estimates During Routine Operations and Surveillance	Table 12.3-7	2.37
Occupational Dose Estimates During Nonroutine Operations and Surveillance	Table 12.3-8	8.61
Occupational Dose Estimates During Routine Maintenance	Table 12.3-9	17.93
Waste processing	Table 12.3-10	5.63
Refueling	Table 12.3-11	8.74
Inservice Inspection	Table 12.3-12	11.6
Special maintenance	Table 12.3-13	13.75
Total		**68.63**

Table 12.3-15 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	139.064	**241.947**
Three-Year Average per plant											**75.092**
2006	1184	25.807	559	5.134	370	8.011	14	.115	629	22.002	**61.069**

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Figure 12.3-1 (Sheet 1 of 34) Radiation Zones Normal Operation/Shutdown (Site)

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Figure 12.3-1 (Sheet 2 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building Sectional View A-A)

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Figure 12.3-1 (Sheet 3 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building Sectional View B-B)

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Figure 12.3-1 (Sheet 4 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation -26'-4")

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Figure 12.3-1 (Sheet 5 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation -8'-7")

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Figure 12.3-1 (Sheet 6 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 3’-7”)

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Figure 12.3-1 (Sheet 7 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 13'-6")

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Figure 12.3-1 (Sheet 8 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 25’-3”)

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Figure 12.3-1 (Sheet 9 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 35'-2")

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Figure 12.3-1 (Sheet 10 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 50’-2”)

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Figure 12.3-1 (Sheet 11 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 76'-5")

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Figure 12.3-1 (Sheet 12 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 101'-0")

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Figure 12.3-1 (Sheet 13 of 34) Radiation Zones for Normal Operation/Shutdown (Reactor Building at Elevation 115'-6")

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Figure 12.3-1 (Sheet 14 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building Sectional View A-A)

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Figure 12.3-1 (Sheet 15 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation -26'-4")

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Figure 12.3-1 (Sheet 16 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation -8'-7")

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Figure 12.3-1 (Sheet 17 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 3'-7")

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Figure 12.3-1 (Sheet 18 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 13'-6")

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Figure 12.3-1 (Sheet 19 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 25'-3")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-1 (Sheet 20 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 35'-2")

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Figure 12.3-1 (Sheet 21 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 50’-2“)

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Figure 12.3-1 (Sheet 22 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 76'-5")

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Figure 12.3-1 (Sheet 23 of 34) Radiation Zones for Normal Operation/Shutdown (Auxiliary Building at Elevation 89'-7")

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Figure 12.3-1 (Sheet 24 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building Sectional View A-A)

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Figure 12.3-1 (Sheet 25 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Elevation -18-0”)

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Figure 12.3-1 (Sheet 26 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Elevation 3'-7")

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Figure 12.3-1 (Sheet 27 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Elevation 34’-0”)

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Figure 12.3-1 (Sheet 28 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Elevation 61'-0")

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Figure 12.3-1 (Sheet 29 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Elevation 88'-10")

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Figure 12.3-1 (Sheet 30 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Elevations 108'-4" and 113'-6")

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Figure 12.3-1 (Sheet 31 of 34) Radiation Zones for Normal Operation/Shutdown (Turbine Building at Roof Elevation)

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Figure 12.3-1 (Sheet 32 of 34) Radiation Zones for Normal Operation/Shutdown (Access Building Sectional View A-A)

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Figure 12.3-1 (Sheet 33 of 34) Radiation Zones for Normal Operation/Shutdown (Access Building at Elevations -26'-4", -11'-4" and 3'-7")

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Figure 12.3-1 (Sheet 34 of 34) Radiation Zones for Normal Operation/Shutdown (Access Building at Elevations 17'-9", 30'-2" and 48'-2")

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Figure 12.3-2 (Sheet 1 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation -26'-4")

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Figure 12.3-2 (Sheet 2 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation -8'-7")

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Figure 12.3-2 (Sheet 3 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 3'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-2 (Sheet 4 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 13'-6")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-2 (Sheet 5 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 25'-3")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-2 (Sheet 6 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 35'-2")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-2 (Sheet 7 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 50’-2”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-2 (Sheet 8 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 76’-5”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-2 (Sheet 9 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 101'-0")

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Figure 12.3-2 (Sheet 10 of 10) General Plant Arrangement with Post Accident Vital Areas (Power Block at Elevation 115’-6”)

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Figure 12.3-3 (Sheet 1 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation -26'-4")

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Figure 12.3-3 (Sheet 2 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation -8'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 3 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 3'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 4 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 13'-6")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 5 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 25'-3")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 6 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 35'-2")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 7 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 50'-2")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 8 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 76'-5")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 9 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 101'-0")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-3 (Sheet 10 of 10) Post Accident Radiation Zone MAP:1hour After Accident (Power Block at Elevation 115'-6")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 1 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation -26'-4")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 2 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation -8’-7”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 3 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 3'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 4 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 13'-6")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 5 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 25'-3")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 6 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 35'-2")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 7 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 50’-2”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 8 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 76’-5”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 9 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 101'-0")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-4 (Sheet 10 of 10) Post Accident Radiation Zone MAP:1day After Accident (Power Block at Elevation 115’-6”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 1 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation -26'-4")

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Figure 12.3-5 (Sheet 2 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation -8’-7”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 3 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 3'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 4 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 13’-6”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 5 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 25’-3”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 6 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 35’-2”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 7 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 50’-2”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 8 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 76'-5")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 9 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 101'-0")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-5 (Sheet 10 of 10) Post Accident Radiation Zone MAP:1week After Accident (Power Block at Elevation 115'-6")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 1 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation -26'-4")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 2 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation -8'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 3 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 3'-7")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 4 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 13'-6")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 5 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 25’-3”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 6 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 35'-2")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 7 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 50’-2”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 8 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 76’-5”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 9 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 101'-0")

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-6 (Sheet 10 of 10) Post Accident Radiation Zone MAP:1month After Accident (Power Block at Elevation 115’-6”)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-7 Isometric View of Main Control Room Shielding

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-8 Labyrinth for radiation protection around Fuel Transfer Tube

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-9 The typical layout of air handling unit

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 12.3-10 The sampling point of the airborne radioactivity monitors

12.4 Operational Radiation Protection Program

12.4.1 Operational Radiation Protection Program Guidelines

The operational protection radiation program is being developed, implemented and maintained as described in the Nuclear Energy Institute Technical Report, NEI 07-03 (Reference 12.4-1). The specific CFR criteria referenced in NEI 07-03 shall be met and strictly adhered to. All recommendations and guidance referenced in NEI-07-03 will be addressed and implemented as applicable to the US-APWR and the plant site.

It is expected that NEI-07-03 will be accepted as the operational radiation protection program standard by the end of 2007. NEI submitted revision 0 of the Topical Report NEI 07-03 to the NRC in March 2007. The NRC responded with 16 requests for more information and 23 suggested editorial changes in August 2007. NEI incorporated what the NRC requested on 13 of the 16 requests for more information and incorporated all of the editorial suggestions. Regarding the remaining three questions, NEI did not believe that they should be incorporated, and responded to the NRC with sound justification as to why the remaining three requests for more information do not need to be incorporated. NEI 07-03 Revision 3 was submitted to the NRC on October 9, 2007.

12.4.2 Operational Radiation Protection Program Contents

The program shall consist of the following as described in Reference 12.4-1:

- 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
- 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

12.4.3 References

12.4-1 Generic FSAR Template Guidance for Radiation Protection Program Description. NEI Technical Report 07-03 Revision 3, Oct. 2007.