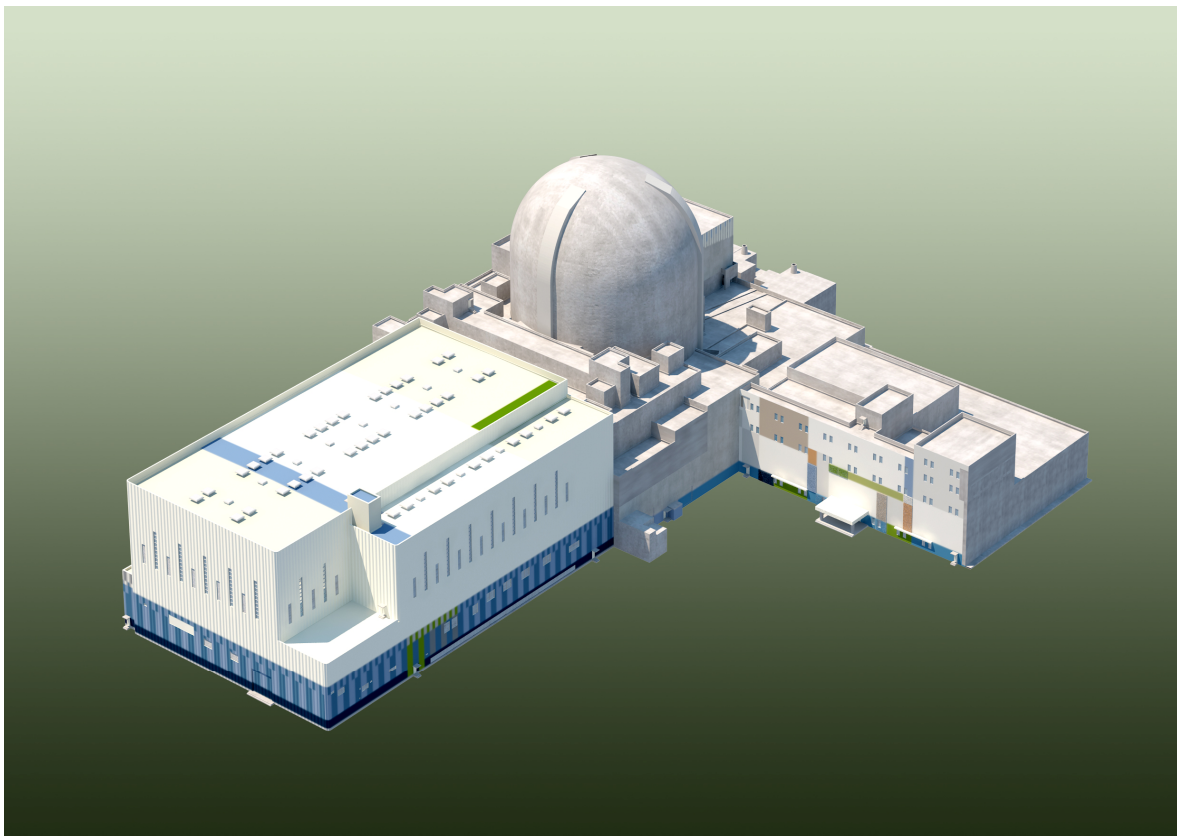


APR1400
DESIGN CONTROL DOCUMENT TIER 2

CHAPTER 12
RADIATION PROTECTION

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ACRONYM AND ABBREVIATION LIST

AB	Auxiliary Building
AC	Alternation Current
ADV	Atmospheric Dump Valve
AFAS	Auxiliary Feedwater Actuation Signal
AFW	Auxiliary Feedwater
AOO	Anticipated Operational Occurrence
APR1400	Advanced Power Reactor 1400
ARSAP	Advanced Reactor Severe Accident Program
ASI	Axial Shape Index
AST	Alternative Source Term
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CEA	Control Element Assembly
CEAC	Control Element Assembly Calculator
CEAE	Control Element Assembly Ejection
CEDE	Committed Effective Dose Equivalent
CEDM	Control Element Drive Mechanism
CEDMCS	Control Element Drive Mechanism Control System
CHF	Critical Heat Flux
CIAS	Containment Isolation Actuation Signal
CLVPS	Containment Low Volume Purge System
COL	Combined License
COLR	Core Operating Limits Report
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
CPCS	Core Protection Calculator System
CPIAS	Containment Purge Isolation Actuation Signal

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CS	Containment Spray
CSAS	Containment Spray Actuation Signal
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DCF	Dose Conversion Factor
DDE	Deep Dose Equivalent
DE	Dose Equivalent
DEG/PD	Double-ended Guillotine at the Pump Discharge
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DPS	Diverse Protection System
DVI	Direct Vessel Injection
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDE	Effective Dose Equivalent
EFPD	Effective Full Power Day
EOC	End of Cycle
EOP	Emergency Operating Procedure
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
ESF-CCS	Engineered Safety Features Component Control System
FF	Flash Fraction
FHA	Fuel Handling Accident
FHAEVAS	Fuel Handling Area Emergency Ventilation Actuation Signal
FLB	Feedwater Line Break
FMEA	Failure Modes and Effects Analysis
FTC	Fuel Temperature Coefficient

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FWCS	Feedwater Control System
GDC	General Design Criteria
GIS	Generated Iodine Spike
GSI	Generic Safety Issue
HEPA	High Efficiency Particulate Air
HFP	Hot Full Power
HPPT	High Pressurizer Pressure Trip
HSGL	High Steam Generator Level
HVAC	Heating Ventilating and Air Conditioning
HZP	Hot Zero Power
ICC	Inadequate Core Cooling
ICRP	International Commission on Radiological Protection
IOSGADV	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
IRWST	In-containment Refueling Water Storage Tank
ITC	Isothermal Temperature Coefficient
LBLOCA	Large Break Loss-of-Coolant Accident
LCO	Limiting Conditions for Operation
LDLB	Letdown Line Break
LFW	Loss of Normal Feedwater Flow
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOAC	Loss of Nonemergency AC Power
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOF	Loss of Flow
LOFW	Loss of Normal Feedwater Flow
LOOP	Loss of offsite Power
LPD	Local Power Density
LPLD	Low PZR Pressure and Low DNBR
LPZ	Low Population Zone
LSGL	Low Steam Generator Level

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LTC	Long-Term Cooling
LTOP	Low Temperature Overpressure Protection
MCR	Main Control Room
MDNBR	Minimum Departure from the Nucleate Boiling Ratio
MFS	Main Feedwater System
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PA	Postulated Accident
PAMI	Post-accident Monitoring Instrumentation
PCMI	Pellet Cladding Mechanical Interaction
PCT	Peak Cladding Temperature
PIS	Pre-accident Iodine Spike
PLCS	Pressurizer Level Control System
PLHGR	Peak Linear Heat Generation Rate
POL	Power Operating Limit
POSRV	Pilot Operated Safety and Relief Valve
PPCS	Pressurizer Pressure Control System
PPS	Plant Protection System
P-T-S	Primary-to-Secondary
PWR	Pressurized Water Reactor
RADTRAD	Radionuclide Transport, Removal, and Dose
RCFC	Reactor Containment Fan Cooler
RCGVS	Reactor Coolant Gas Vent System
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System

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RHR	Residual Heat Removal
RIA	Reactivity-initiated Accident
RMI	Reflective Metal Insulation
RMS	Radiation Monitoring System
RPCS	Reactor Power Cutback System
RPS	Reactor Protection System
RRS	Reactor Regulating System
RTO	Reactor Trip Override
RTP	Return to Power
RV	Reactor Vessel
SAFDL	Specified Acceptable Fuel Design Limit
SBCS	Steam Bypass Control System
SBLOCA	Small Break Loss-of-Coolant Accident
SCS	Shutdown Cooling System
SDC	Shutdown Cooling
SDM	Shutdown Margin
SER	Safety Evaluation Report
SF	Single Failure
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIP	Safety Injection Pump
SIRCP	Startup of an Inactive Reactor Coolant Pump
SIS	Safety Injection System
SIT-FD	Safety Injection Tank with Fluidic Device
SIT	Safety Injection Tank
SLB	Steam Line Break
SLBFP	Large Steam Line Break during Full Power Operation
SLBZP	Large Steam Line Break during Zero Power Operation

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SRM	Source Range Monitoring
SRP	Standard Review Plan
SRS	Simple Random Sampling
SRSS	Square-root-sum-of-the-squares
TBV	Turbine Bypass Valve
TEDE	Total Effective Dose Equivalent
TSC	Technical Support Center
TS	Technical Specification
UGS	Upper Guide Structure
USEPA	U.S. Environmental Protection Agency
USI	Unresolved Safety Issue
VOPT	Variable Overpower Trip

CHAPTER 12 – RADIATION PROTECTION

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

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

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

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

Radiation protection program for the APR1400 associated with maintaining occupational exposure to radiation as low as reasonably achievable (ALARA) are described below, specified in NEI 07-08A (Reference 1).

12.1.1 Policy Considerations

The design policy of the APR1400 is to maintain occupational radiation exposure ALARA to operating and construction personnel and to the public. The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and regulatory guides including NRC RGs 1.8 (Reference 2), 1.33 (Reference 3), 8.8 (Reference 4), and 8.10 (Reference 5) (COL 12.1(1)).

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The ALARA policy is applied to the total exposures (person-Sv) received by all operation and construction personnel, and to individual exposures. The proper approach and awareness of potential problems related to radiation protection are addressed through the proper training of all plant personnel as provided in NRC RG 1.8.

12.1.1.1 Design Policies

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

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

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

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

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

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

12.1.1.2 Operation Policies

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

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

The functions of the radiation protection team include the following:

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

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- b. Provide reasonable assurance that radiation exposure to plant personnel is maintained ALARA by:
 - 1) Evaluating radiological conditions in areas to be accessed by personnel and taking precautionary measures to provide reasonable assurance of minimal exposure
 - 2) Controlling personnel and equipment movement into and out of radiation control areas
 - 3) Providing instruction and supervision for the proper use and care of special protective clothing and equipment
 - 4) Posting each radiation control area with conspicuous and appropriate caution signs, and access control means (locking devices, rope) in accordance with the applicable regulations
 - 5) Administering and controlling conditions of radiation work permits for work in areas with high radiation or contamination levels in accordance with the approved procedures and the radiation protection program
- c. Determine the procedural requirements for the use of personnel monitoring devices and maintain records of personnel exposure in accordance with the applicable regulations.
- d. Control and document all radioactive material entering or leaving the plant site in accordance with the applicable regulations
- e. Establish procedures for dealing with emergency conditions
- f. Train plant staff and visitors in the radiation protection policies and procedures, as required

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- g. Include a dosimetry program as part of the site health physics program. The program includes whole body counting to measure the uptake of radiation by personnel
- h. Conduct periodic calibration for process radiation, area radiation, portable radiation, and airborne radioactivity monitors
- i. Monitor and maintain records of all radioactive effluents released from the station
- j. Collect data required in the evaluation of radiological effects from plant operation

12.1.2 Design Considerations

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

12.1.2.1 General Design Considerations for Maintaining Occupational Exposures ALARA

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

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

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

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

- a. To reduce the duration and/or frequency of required equipment maintenance through the use of industry-proven equipment and construction materials that are commensurate with the intended plant processes and services
- b. To reduce radiation levels through the use of operational controls such as frequent flushing and cleaning of equipment and the use of localized shielding where needed
- c. To reduce the time personnel spend in radiation areas where maintenance and other operational activities are necessary by providing easy access to components that require frequent maintenance and by moving components to low radiation zones to perform maintenance when possible

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

The APR1400 is also designed to minimize the production, distribution, and retention of activated corrosion products throughout the reactor coolant system. The production of corrosion products is minimized by the proper selection of materials and the implementation of chemistry control of the primary system, as described in Subsection 5.2.4.2 and Subsection 12.3.1.3. Reactor coolant leakage is controlled to minimize the distribution and retention of the corrosion products as described in Subsection 12.3.1.2(i).

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In accordance with the requirements in 10 CFR 20.1406 (Reference 6), the APR1400 design implements the design approaches to minimize, to the extent practicable, contamination of the facility and the environment; facilitates eventual decommissioning; and minimizes, to the extent practicable, the generation of radioactive waste. The design considerations that comply with these requirements are described in Subsection 12.3.1.10.

12.1.2.2 Equipment Design Considerations for Maintaining Occupational Exposures ALARA

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

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

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

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

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

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

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b. Enhanced environmental qualification of equipment

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

c. Material selection of piping and component

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

12.1.2.3 Facility Layout Design Considerations for Maintaining Occupational Exposures ALARA

The facility layout is designed so the duration of time spent by workers within radiation areas for the performance of testing and maintenance activities is minimized. Also, the separation of more highly radioactive equipment from less radioactive equipment is implemented to the greatest extent practicable in the layout design.

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

- a. Adequate spacing to facilitate the accessibility of equipment during maintenance activities and to enable the installation of temporary shielding, where necessary, during the performance of testing and maintenance activities in radiation areas.
- b. Separation of non-radioactive systems from radioactive systems to the extent practicable in order to minimize the spread of contamination.
- c. Installation of system components into separate cubicles or compartments based on their respective frequency of maintenance, operational characteristics, level of radioactivity, and inclusion of passive versus active components.

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

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

- a. Tanks are located in individual cubicles to the extent practicable and based on their levels of contamination; shield walls are sized adequately and conservatively to attenuate radiation and maintain low dose rates in areas occupied by plant personnel.
- b. Pumps and valve galleries are provided outside the cubicles containing the radioactive equipment to minimize operational exposures during operation and maintenance activities.
- c. Instruments are located in low radiation areas, and remote operation is implemented to the maximum extent practicable.
- d. Piping that contains contaminated fluids is routed inside the pipe chases to the maximum extent practicable.
- e. Shield wall penetrations are positioned to minimize radiation streaming from radioactive equipment to low radiation areas.

12.1.3 Operational Considerations

The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational radiation exposures are ALARA (COL 12.1(2)). The APR1400 radiation protection design is implemented in conjunction with operational

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procedures and programs. Radiological health and safety procedures are developed and are continually reviewed and updated as necessary to provide reasonable assurance that occupational radiation exposures are ALARA. These procedures incorporate experience gained from plant operations. Improvements suggested during design and operation are incorporated and implemented to continually update the program in order to maximize protection of the plant personnel, the public, the environment, and the equipment.

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

12.1.3.1 General Operational Considerations for Maintaining Exposures ALARA

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

Operational procedures and programs are to be developed by the COL applicant to address contamination control measures for all SSCs that have the potential to be contaminated with radiological material with the plant in accordance with NRC RG 4.21 (Reference 22). The APR1400 approach to the minimization of contamination and radioactive waste generation is described in Subsection 12.3.1.10 and COL 12.3(1).

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12.1.3.2 Design Features for ALARA Exposures During Maintenance and Inspection

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

- a. Platforms are provided around the steam generator (SG) to facilitate the handling of the reactor coolant pump (RCP) seal cartridges.
- b. Sufficient spacing is provided for ingress and egress, equipment laydown/pull areas, and removal paths for major equipment that is expected to be replaced during the plant lifetime.
- c. Removable insulations are used.
- d. The length of welds is reduced through the use of seamless piping as much as possible.
- e. An integrated head assembly (IHA) is used.

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

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

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

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- b. Integrally forged components and seamless piping are selected whenever possible to minimize the need for inservice weld inspections.
- c. Blanket-type thermal insulation with hook-and-loop fasteners is selected for components and piping containing radioactive material. This type of insulation can be removed easily by one worker for inservice weld inspection.
- d. Adequate containment or ventilation is provided to minimize the potential for the spread of airborne contamination. Subsection 12.2.2 specifies the methodology for determining the in-plant concentration of airborne contamination. In Subsection 12.3.1.4, more information on airborne contamination control is described.

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

12.1.3.3 Post-Accident Conditions

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

12.1.3.4 Decommissioning

The ALARA considerations for decommissioning are also incorporated into the system design and component layout through the provisions of individual shielded cubicles,

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adequate spacing, and ingress and egress design, and the partitioning and separation of low contamination versus high contamination areas. Other considerations for decommissioning include the provisions of decontamination capability to reduce residual contaminations, components designed with adequate capacities to reduce processing time, and the approach to maximize the use of integrated component assemblies to reduce stay time for dismantlement. In addition, decommissioning considerations for the compliance with NRC RG 4.21 and for applying to NEI 08-08A (Reference 24) are built into the design for systems and components that handle contaminated or potentially contaminated fluids. Decommissioning considerations are provided in Subsection 12.3.1.10.

12.1.4 Combined License Information

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

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

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

12.1.5 References

1. NEI 07-08A, "Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)," Rev 0, October 2009.
2. NRC RG 1.8, "Qualification & Training of Personnel for Nuclear Power Plants," Rev.3, Nuclear Regulatory Commission, May 2000.

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3. NRC RG 1.33, "Quality Assurance Program Requirements Operation," Nuclear Regulatory Commission, Rev 3, June 2013.
4. NRC RG 8.8, "Information Relevant to Ensuring the Occupational Radiation Exposures at Nuclear Power Stations will be ALARA," Rev. 3, Nuclear Regulatory Commission, June 1978.
5. NRC RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," Rev. 1-R, Nuclear Regulatory Commission, May 1977.
6. 10 CFR 20.1406, "Minimization of Contamination," June 2011.
7. NRC RG 8.2, "Guide for Administrative Practices in Radiation Monitoring," Nuclear Regulatory Commission, May 2011.
8. NRC RG 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," Nuclear Regulatory Commission, November 2005.
9. NRC RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," Nuclear Regulatory Commission, July 1993.
10. NRC RG 8.13, "Instruction Concerning Prenatal Radiation Exposure," Nuclear Regulatory Commission, June 1999
11. NRC RG 8.15, "Acceptable Programs for Respiratory Protection," Nuclear Regulatory Commission, October 1999.
12. NRC RG 8.20, "Applications of Bioassay for I-125 and I-131," Nuclear Regulatory Commission, September 1979.
13. NRC RG 8.25, "Air Sampling in the Workplace," Nuclear Regulatory Commission, June 1992.
14. NRC RG 8.26, "Applications of Bioassay for Fission and Activation Products," Nuclear Regulatory Commission, September 1980.
15. NRC RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants," Nuclear Regulatory Commission, March 1981

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16. NRC RG 8.28, "Audible-Alarm Dosimeters," Nuclear Regulatory Commission, August 1981
17. NRC RG 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure," Nuclear Regulatory Commission, February 1996.
18. NRC RG 8.34, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses," Nuclear Regulatory Commission, July 1992
19. NRC RG 8.35, "Planned Special Exposures," Nuclear Regulatory Commission, August 2010
20. NRC RG 8.36, "Radiation Dose to the Embryo/Fetus," Nuclear Regulatory Commission, July 1992
21. NRC RG 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants," Nuclear Regulatory Commission, May 2006
22. NRC RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," Nuclear Regulatory Commission, June 2008.
23. NEI 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program," Rev 0, May 2009.
24. NEI 08-08A, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," Rev 0, October 2009.

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12.2 Radiation Sources

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

12.2.1 Contained Sources

The shielding design source terms are based on full power operation with 0.25 percent fuel cladding defects and no credit for gas stripping. Sources in the primary coolant include fission products released from fuel cladding defects, activation products, and corrosion products. Throughout most of the reactor coolant system (RCS), activation products (principally nitrogen-16 (N-16)) are the primary radiation sources for shielding design during power operation. Dimensions and parameters of the radiation sources used in the shielding analyses are listed in Table 12.2-25.

12.2.1.1 Reactor Containment Building

12.2.1.1.1 Reactor Core

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

12.2.1.1.2 Reactor Coolant System

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

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

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

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

The N-16 activity at the RV outlet nozzle is 5.53×10^6 disintegrations/cm³-sec. This activity, after considering the volume expansion of coolant at the vessel outlet nozzle, is calculated by the following expression and the reactor parameters. Table 12.2-7 lists typical N-16 activities at various locations in the primary coolant loop.

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

Where:

$\Sigma\Phi$ = reaction rate (5.02×10^7 reactions/cm³-sec)

t_c = core transit time (0.81 sec)

t_t = total primary loop time (10.6 sec)

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

λ = decay constant (0.097 sec^{-1})

12.2.1.1.3 Secondary Side Systems

The rate of SG tube leakage is assumed to be 3,270 L/day (0.6 gal/min). This is assumed to be concurrent with 0.25 percent fuel cladding defects for the shielding design basis

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source term calculations. A blowdown rate of 0.2 percent of maximum steaming rate is assumed. No credit is taken for the operation of the condensate polisher demineralizers. Assumptions for calculation of main steam system (MSS) radiation sources are presented in Table 11.1-5, and results of the calculation are given in Table 12.2-18.

12.2.1.1.4 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant source of radiation in the reactor containment building after plant shutdown for refueling. A reactor operating time to reach a near-equilibrium buildup level of fission product based on the rated power operation is used in determining the source strengths. The parameters used in the spent fuel decay gamma source calculation, such as fuel enrichment, specific core power, and discharge burnup, are given in Table 12.2-8. The spent fuel decay gamma source is given in Table 12.2-9. Fuel assembly dose rates as a function of distance in the refueling pool water and time after shutdown are shown in Figures 12.2-1 and 12.2-2.

12.2.1.1.5 Processing Systems

12.2.1.1.5.1 Chemical and Volume Control System

The shielding design for chemical and volume control system (CVCS) components is based on the maximum expected radioactivity in each of the components. The component source terms are given in Tables 12.2-10 through 12.2-14.

a. Heat exchangers

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

The total radioactivity inventory for the CVCS heat exchangers is based on the tubeside water volume (including the shellside water volume for regenerative heat exchangers).

b. Ion exchangers

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

1) Purification ion exchanger

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

2) Deborating ion exchanger

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

3) Pre-holdup ion exchanger

The total radioactivity inventory is based on resin buildup for the duration of one cycle EFPDs. The DF for all nuclides except Xe, Kr, Y, Rb, and Cs is 10 with an efficiency of 90 percent. The DF for Rb and Cs is 100 with an efficiency of 99 percent. The DF for Xe, Kr, and Y is 1 with an efficiency of 0 percent. Liquid inputs processed by the pre-holdup ion exchanger include

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the letdown processed through the purification ion exchanger and the purification filter, and the inflow from the reactor drain tank (RDT) and the equipment drain tank (EDT).

4) Boric acid condensate ion exchanger

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

c. Filters

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

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

d. Tanks

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

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

e. Boric acid concentrator

The maximum values for BAC radionuclide inventories are presented in Table 12.2-14.

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

12.2.1.1.5.2 Steam Generator Blowdown System

Radiation sources in the steam generator blowdown system (SGBDS) are shown in Table 12.2-18. The sources are based on the assumed design basis primary-to-secondary (PTS) leakage rate and the assumed fuel defect percentage described in Subsection 12.2.1.1.3. The blowdown rate is assumed to be 0.2 percent of the maximum steaming rate.

12.2.1.1.5.3 Condensate Polishing System

Radiation sources in the condensate polishing system (CPS) are shown in Table 12.2-18. The sources are based on the design basis PTS leakage and the assumed fuel defect percentage described in Subsection 12.2.1.1.3. It is assumed that 65 percent of the condensate flows through the CPS and that one out of six CPS demineralizers is used to process the condensate during normal operation.

12.2.1.1.6 Gamma Sources of Irradiated Components

The components in the reactor vessel are irradiated by the fission neutrons during the core power operation and are activated. The in-core instrument (ICI) assembly, which consists of five rhodium detectors, one background detector, one core exit thermocouple, and a central member assembly, is enclosed in a protective sheath. Activated gamma sources of the irradiated ICI assembly are estimated assuming 6 years of irradiation. The activated gamma sources of the irradiated control element assembly (CEA) and the irradiated neutron source assembly (NSA) are estimated assuming 10 years of irradiation. In CEA, the neutron absorbing material is B₄C and the cladding material is Inconel 625. The NSA contains the primary neutron source of Cf²⁵² and the secondary neutron source of Sb-Be. The activated gamma source of the irradiated surveillance capsule assembly (SCA) is estimated assuming 60 years of irradiation. The irradiation calculation is performed using the ORIGEN-S computer program, and the irradiation time of each component is based on the design life time of each component. The activated gamma sources of ICI, CEA, NSA, and SCA are given in Table 12.2-15.

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

12.2.1.2 Auxiliary Building

12.2.1.2.1 Shutdown Cooling System

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

12.2.1.2.2 Component Cooling Water System

The component cooling water system (CCWS) is a closed-loop demineralized water system that can potentially become contaminated by heat exchanger leakage from other radioactive systems that are cooled by the CCWS. The CCWS is provided with radiation monitors to detect inleakage of radioactive contaminants as described in Subsection 11.5.2.3. The detection of leakage is sensitive to a level of 3.7×10^{-2} Bq/cm³ gross gamma activity in the CCWS cooling water.

In the event of a major leak in a CCWS train, the train is removed from service, a non-leaking train is used, and the contaminated water is treated through the LWMS. This design prevents the spread of contamination through the CCWS and connecting systems.

12.2.1.2.3 Spent Fuel Storage and Transfer

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

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12.2.1.2.4 Spent Fuel Pool Cooling and Cleanup System

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

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

The purification process serves two purposes:

- a. Removing the noble gases in the gas stripper avoids large activity releases to the reactor containment building following reactor vessel head removal.
- b. The ion exchange and filtration reduces dissolved fission and corrosion product concentrations in the coolant that would otherwise enter the SFP and the refueling pool.

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

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

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

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12.2.1.3 Turbine Generator Building

Radiation sources in the turbine generator building occur in the condensate polishing system (CPS) due to the design basis PTS leakage rate in the steam generator. Activity levels for all turbine generator building related sources are summarized in Table 12.2-18. The activities provided in Table 12.2-18 are based on normal operation reactor coolant activity levels and PTS leakage conditions. Radionuclide removal efficiencies of demineralizers in the CPS are assumed to be consistent with the guidance in NUREG-0017 (Reference 1).

12.2.1.4 Compound Building

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

Gaseous waste management system (GWMS) source terms are provided in Table 12.2-19. Radiation sources for each component of the GRS are calculated using the shielding basis equilibrium reactor coolant radionuclide concentrations provided in Table 12.2-5, which are based on an assumed 0.25 percent fuel defect. Activity buildup on the process gas charcoal beds is calculated assuming maximum design basis holdup times for noble gases in accordance with NUREG-0017.

The source terms for liquid waste management system (LWMS) tanks are provided in Table 12.2-20 and for LWMS processing equipment in Table 12.2-21. Source terms for the EDT and floor drain tank (FDT) are calculated using reactor coolant equilibrium radionuclide concentrations presented in Table 12.2-5 and the activity fractions in Table 11.2-2. Radionuclide concentrations in the LWMS are determined using the DIJESTER Code (Reference 2). The accumulation and decay of radionuclides in the LWMS can be modeled using this code.

The activities of LWMS demineralizers are calculated using an activity buildup and decay model. The calculation applies the process flow rates provided in Table 11.2-2, and the process fluid activity levels provided in Table 12.2-20. The demineralizer resin is assumed to have a service life of 1 year. Although the service life of filters and resins in

the LWMS may vary according to operating conditions, for radiation protection purposes, they are replaced based on the source term strength to provide reasonable assurance that occupational exposures associated with radwaste system operations remain ALARA.

Solid waste management system (SWMS) source terms are provided in Table 12.2-22. Source terms for the spent resin long-term storage tank are calculated based on the activity of CVCS demineralizer resins presented in Table 12.2-11. Source terms for the low-activity spent-resin storage tank are calculated based on the activity of LWMS demineralizer resins presented in Table 12.2-21.

12.2.1.5 Sources Resulting from Design Basis Accidents

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

12.2.1.6 Stored Radioactivity

The holdup tanks, reactor makeup water tanks (RMWTs), and boric acid storage tanks (BASTs) are the principal sources of activity outside the plant buildings. The surface dose rate of these tanks is designed so that it does not exceed 3 $\mu\text{Sv/hr}$. Administrative controls are in place to prevent personnel from occupying the immediate vicinity of the outside tanks.

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

Storage space is allocated in the compound building for the storage of spent filter cartridges, dewatered spent resins, and R/O concentrates from the LWMS. Plant areas storing radioactive wastes are shielded based on maximum stored waste volumes and average expected source strengths so that the radiation level outside the storage area is below the limit of the designated radiation zone.

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

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12.2.1.7 Pipe Routing

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

Criteria for routing radioactive piping include:

- a. Piping containing radioactive material is routed through shielded pipe chases to the extent practicable.
- b. Systems containing radioactive liquids, gases, or slurries are physically located close to interfacing systems to reduce pipe length and minimize the need for routing radioactive piping through personnel access corridors.
- c. Stagnant runs of piping are avoided to minimize the potential for crud traps. Flushing and decontamination capabilities are provided as necessary.

12.2.2 Airborne Radioactive Material Sources

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

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

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

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

- a. Maintenance of in-plant airborne concentrations of radioisotopes in normally occupied areas within the derived air concentration (DAC) prescribed in Table 1 of 10 CFR 20, Appendix B (Reference 3).
- b. Sufficient confinement and ventilation capability to prevent the spread of airborne contamination.
- c. Airborne radiation monitors in normally occupied areas where the potential for airborne contamination exists. The airborne radiation monitoring system is described in Section 11.5.

12.2.2.1 Production of Airborne Radioactive Material

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

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

12.2.2.2 Airborne Sources in Normally Accessible Areas

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

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The ventilation system is designed to control the airborne radioactivity in the laboratories, maintenance areas, and the reactor containment building.

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

12.2.2.3 Airborne Concentrations

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

Concentrations of noble gases, tritium, and iodine are calculated as a fraction of the DAC for air in the reactor containment building and normally accessible cubicles within the auxiliary building and the compound building. Except under abnormal conditions, general access areas normally have little, if any, airborne contaminants during normal operation. Table 12.2-23 provides the airborne radioactivity concentrations and DAC fractions for the main cubicles in the reactor containment, auxiliary building, and compound building.

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

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

Where:

C_A = airborne concentration in each cubicle (Bq/cm³)

L = leak rate (cm³/min)

C = radioactive concentration of liquid or gas (Bq/cm³)

P = fraction of activity released to air

λ = decay constant (min⁻¹)

V = enclosed room volume (cm³)

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

The equation above represents the solution of a differential equation in equilibrium conditions in which there is no airborne radioactivity in the ventilation airflow(s) entering the area under consideration. To accommodate situations in which airborne radioactivity is in one or more ventilation airflow streams entering the area of concern, additional term(s) are added to the basic differential equation. Assumptions and parameters used for the airborne source term calculations are provided in Table 12.2-26.

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

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

Plant areas requiring post-accident occupation (vital areas) and the duration of occupation are identified in Subsections 12.3.1.8 and 12.3.1.9, respectively.

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

Initial core releases that are equivalent to those recommended in NRC RG 1.183 (Reference 5) are used. The source terms are presented in Table 12.2-24.

These source terms are used to evaluate the adequacy of shielding in post-accident conditions using shielding computer codes, which are addressed in Section 12.3, to verify that:

- a. Vital areas are accessible for operators to take the required mitigative actions during post-accident conditions.
- b. Safety-related equipment is qualified for the environmental radiation condition in the area where the equipment is located.

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

12.2.4 Combined License Information

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

12.2.5 References

1. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," Rev. 1, Nuclear Regulatory Commission, April 1985.
2. DIJESTER, "A Program to Compute Radioactive Decay in Fluid Flow Systems," Program No. 9.8. 060-1.0, D. J. Pichurski, April 1976.
3. 10 CFR 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."
4. NUREG-0737, Subsection II.B.2.3, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations," Nuclear Regulatory Commission, 1980.
5. NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Nuclear Regulatory Commission, July 2000.
6. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society, 1999 (withdrawn 2009).

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Table 12.2-1

Normal Operation Neutron Spectra Outside the Reactor Vessel

Average Neutron Energy (eV)	Neutron Spectra ⁽¹⁾ (neutrons/cm ² -s)	Average Neutron Energy (eV)	Neutron Spectra ⁽¹⁾ (neutrons/cm ² -s)
1.58E+07	3.84E+05	2.40E+05	1.88E+09
1.32E+07	1.04E+06	1.47E+05	1.90E+09
1.11E+07	3.78E+06	8.92E+04	1.23E+09
9.30E+06	6.69E+06	5.41E+04	9.14E+08
8.01E+06	9.76E+06	3.64E+04	3.04E+08
6.74E+06	2.03E+07	2.89E+04	1.82E+08
5.52E+06	2.79E+07	2.51E+04	5.46E+08
4.32E+06	5.00E+07	2.30E+04	3.50E+08
3.35E+06	4.00E+07	1.85E+04	5.96E+08
2.87E+06	3.17E+07	1.11E+04	7.74E+08
2.60E+06	3.94E+07	5.23E+03	7.96E+08
2.42E+06	2.00E+07	2.47E+03	6.63E+08
2.36E+06	6.20E+06	1.02E+03	1.03E+09
2.29E+06	3.13E+07	3.34E+02	5.40E+08
2.08E+06	8.73E+07	1.58E+02	5.33E+08
1.79E+06	1.22E+08	6.93E+01	6.54E+08
1.50E+06	2.07E+08	2.40E+01	7.57E+08
1.18E+06	4.96E+08	7.86E+00	4.19E+08
9.12E+05	5.06E+08	3.45E+00	5.34E+08
7.82E+05	2.31E+08	1.37E+00	3.89E+08
6.75E+05	1.04E+09	6.45E-01	3.55E+08
5.53E+05	1.01E+09	2.57E-01	7.95E+08
4.33E+05	9.84E+08	5.00E-02	3.06E+09
3.33E+05	1.53E+09		

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

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Table 12.2-2

Normal Operation Gamma Spectra Outside the Reactor Vessel

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

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

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Table 12.2-3

Shutdown Gamma Spectra Outside the Reactor Vessel

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

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

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Table 12.2-4

Basis for Reactor Coolant Fission Product Activities⁽¹⁾

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

(1) Shielding Design Source Term (0.25% Fuel Defect, DAMSAM Code Input)

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Table 12.2-5

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

Nuclide	Activity (Bq/g)	Nuclide	Activity (Bq/g)
Kr-85m	1.04E+04	Co-60	2.10E+01 ⁽¹⁾
Kr-85	4.44E+04	Zn-65	2.02E+01 ⁽¹⁾
Kr-87	8.14E+03	Sr-89	3.26E+01
Kr-88	2.26E+04	Sr-90	2.18E+00
Xe-131m	4.44E+04	Sr-91	4.81E+01
Xe-133m	2.70E+03	Y-91m	2.81E+01
Xe-133	2.89E+06	Y-91	4.81E+00
Xe-135m	5.92E+03	Y-93	1.15E+00
Xe-135	5.92E+04	Zr-95	1.54E+01 ⁽¹⁾
Xe-137	1.37E+03	Nb-95	5.18E+00
Xe-138	5.18E+03	Mo-99	2.81E+03
Br-84	1.96E+02	Tc-99m	1.63E+03
I-131	2.48E+04	Ru-103	1.74E+00
I-132	6.66E+03	Ru-106	7.40E-01
I-133	3.52E+04	Ag-110m	5.15E+01 ⁽¹⁾
I-134	4.07E+03	Te-129m	5.92E+01
I-135	2.00E+04	Te-129	6.29E+01
Rb-88	2.29E+04	Te-131m	2.81E+02
Cs-134	3.52E+03	Te-131	1.11E+02
Cs-136	4.81E+02	Te-132	1.96E+03
Cs-137	4.07E+03	Ba-137m	3.70E+03
N-16	8.22E+06 ⁽²⁾	Ba-140	4.07E+01
H-3	1.30E+05	La-140	1.37E+01
Na-24	1.81E+03 ⁽¹⁾	Ce-141	1.52E+00
Cr-51	5.48E+02	Ce-143	4.07E+00
Mn-54	6.34E+01 ⁽¹⁾	Ce-144	4.44E+00
Fe-55	4.75E+01 ⁽¹⁾	W-187	9.70E+01 ⁽¹⁾
Fe-59	1.19E+01 ⁽¹⁾	Np-239	8.62E+01 ⁽¹⁾
Co-58	1.82E+02 ⁽¹⁾		

(1) ANSI/ANS-18.1 basis source terms (Reference 6)

(2) At the reactor vessel outlet nozzle

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Table 12.2-6

Deposited Crud Activities
(SG Tubing and Reactor System Piping)

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

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Table 12.2-7

N-16 Activity

Location	Activity (Bq/g)
Vessel outlet nozzle	8.22E+06
Vessel outlet line (midpoint)	8.13E+06
Steam generator (midpoint)	5.68E+06
Reactor coolant pump (midpoint)	3.78E+06
Vessel inlet line (midpoint)	3.61E+06

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Table 12.2-8

Parameters Used in Spent Fuel Decay Gamma Source Calculation

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

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Table 12.2-9

Spent Fuel Gamma Source

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

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Table 12.2-10 (1 of 2)

CVCS Heat Exchanger Inventories, Maximum Values (Bq)

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

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Table 12.2-10 (2 of 2)

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

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Table 12.2-11 (1 of 2)

CVCS Ion Exchanger Inventories, Maximum Values (Bq)

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

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Table 12.2-11 (2 of 2)

Nuclide	Purification	Deborating	Pre-holdup	Boric Acid Condensate
Te-129m	1.3E+12	4.1E+09	2.6E+09	1.9E+05
Te-129	1.9E+09	1.7E+07	2.2E+06	8.1E-01
I-131	1.2E+14	9.6E+11	2.2E+11	9.3E+06
Te-131m	2.1E+11	1.9E+09	2.9E+08	2.5E+03
Te-131	1.2E+09	1.1E+07	1.4E+06	1.9E-01
Te-132	4.0E+12	3.6E+10	6.3E+09	1.3E+05
I-132	4.0E+11	3.6E+09	4.6E+08	3.5E+02
I-133	1.9E+13	1.7E+11	2.5E+10	1.6E+05
I-134	9.2E+10	8.4E+08	1.1E+08	3.1E+01
Cs-134	3.5E+14	2.0E+09	6.1E+12	1.6E+04
I-135	3.5E+12	3.2E+10	4.2E+09	9.2E+03
Cs-136	2.3E+12	2.7E+08	3.9E+10	1.3E+03
Cs-137	5.0E+14	2.3E+09	8.6E+12	1.9E+04
Ba-140	3.2E+11	9.2E+05	6.6E+08	2.4E+02
La-140	1.4E+10	3.1E+05	2.3E+07	1.4E+01
Ce-141	3.1E+10	3.4E+04	6.7E+07	1.3E+01
Ce-143	3.5E+09	9.2E+04	5.4E+06	3.4E+00
Ce-144	5.4E+11	1.0E+05	1.2E+09	4.9E+01
Na-24	7.0E+11	4.1E+07	1.0E+09	6.7E+02
Cr-51	9.4E+12	7.2E+10	2.1E+10	4.5E+03
Mn-54	8.0E+12	1.1E+10	1.9E+10	7.0E+02
Fe-55	8.3E+12	8.0E+09	1.9E+10	5.3E+02
Fe-59	3.3E+11	1.7E+09	7.4E+08	1.1E+02
Co-58	8.0E+12	2.8E+10	1.8E+10	1.8E+03
Co-60	4.0E+12	3.6E+09	9.2E+09	2.4E+02
Zn-65	2.3E+12	4.5E+05	5.2E+09	2.2E+02
Ba-137m	5.0E+14	2.3E+09	8.6E+12	1.9E+04
W-187	6.0E+10	2.2E+06	8.9E+07	5.8E+01
Np-239	1.3E+11	1.9E+06	2.1E+08	1.2E+02

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Table 12.2-12 (1 of 2)

CVCS Filter Inventories, Maximum Values (Bq)

Nuclide	Seal Injection	Reactor Drain	Boric Acid	Purification	Reactor Makeup Water
H-3	9.8E+08	1.7E+09	9.7E+08	2.0E+09	1.9E+09
N-16	0.0E+00	0.0E+00	0.0E+00	1.8E+08	0.0E+00
Kr-85m	7.7E+07	6.9E+07	0.0E+00	1.6E+08	7.1E-03
Kr-85	3.4E+08	5.7E+08	2.1E+03	6.7E+08	1.7E+02
Kr-87	5.6E+07	5.3E+07	0.0E+00	1.2E+08	4.6E-04
Kr-88	1.6E+08	1.5E+08	0.0E+00	3.4E+08	6.2E-03
Xe-131m	3.4E+08	4.9E+08	2.3E+02	6.7E+08	4.8E+01
Xe-133m	2.0E+07	2.3E+07	1.6E-02	4.1E+07	2.5E-01
Xe-133	2.2E+10	2.9E+10	1.0E+03	4.4E+10	8.2E+02
Xe-135m	3.4E+07	3.8E+07	0.0E+00	9.0E+07	1.5E-05
Xe-135	4.4E+08	4.1E+08	0.0E+00	9.0E+08	1.7E-01
Xe-137	6.3E+06	8.8E+06	0.0E+00	2.1E+07	2.4E-07
Xe-138	2.9E+07	3.3E+07	0.0E+00	7.8E+07	1.0E-05
Br-84	1.2E+04	1.3E+06	0.0E+00	3.0E+06	1.3E-06
Rb-88	6.6E+07	1.5E+08	0.0E+00	3.5E+08	3.4E-04
Sr-89	4.9E+03	4.0E+05	3.7E+04	4.9E+05	2.3E+00
Sr-90	3.3E+02	2.8E+04	1.9E+04	3.3E+04	3.7E-01
Sr-91	7.2E+03	3.3E+05	1.1E-12	7.3E+05	1.2E-03
Y-91m	1.9E+05	1.8E+05	0.0E+00	4.3E+05	7.6E-04
Y-91	3.6E+04	6.0E+04	5.9E+05	7.3E+04	3.4E+01
Y-93	8.6E+03	8.0E+03	1.1E-11	1.7E+04	4.4E-03
Zr-95	2.3E+03	1.9E+05	2.4E+04	2.3E+05	1.3E+00
Nb-95	7.8E+02	6.3E+04	3.5E+03	7.8E+04	2.8E-01
Tc-99m	2.4E+05	1.1E+07	0.0E+00	2.5E+07	1.6E-02
Mo-99	4.2E+05	2.5E+07	5.5E+02	4.3E+07	3.5E+00
Ru-103	2.6E+02	2.1E+04	1.4E+03	2.6E+04	1.0E-01
Ru-106	1.1E+02	9.5E+03	4.3E+03	1.1E+04	1.1E-01
Ag-110m	7.8E+03	6.6E+05	2.5E+05	7.8E+05	6.9E+00

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Table 12.2-12 (2 of 2)

Nuclide	Seal Injection	Reactor Drain	Boric Acid	Purification	Reactor Makeup Water
Te-129m	4.5E+03	7.2E+05	3.5E+04	9.0E+05	2.9E-01
Te-129	4.3E+03	4.1E+05	0.0E+00	9.5E+05	1.9E-06
I-131	1.9E+06	2.6E+08	5.9E+05	3.8E+08	2.1E+01
Te-131m	2.1E+04	2.2E+06	2.9E-02	4.3E+06	5.9E-03
Te-131	6.8E+03	7.2E+05	0.0E+00	1.7E+06	4.6E-07
Te-132	1.5E+05	1.8E+07	9.1E+02	3.0E+07	3.1E-01
I-132	4.8E+05	4.4E+07	0.0E+00	1.0E+08	8.3E-04
I-133	2.6E+06	2.6E+08	3.7E-02	5.3E+08	3.8E-01
I-134	2.7E+05	2.6E+07	0.0E+00	6.2E+07	7.3E-05
Cs-134	1.3E+07	4.5E+07	1.0E+07	5.3E+07	2.2E+02
I-135	1.5E+06	1.4E+08	0.0E+00	3.0E+08	2.2E-02
Cs-136	1.8E+06	5.4E+06	2.1E+04	7.3E+06	4.0E+00
Cs-137	1.5E+07	5.2E+07	1.5E+07	6.2E+07	2.8E+02
Ba-140	6.2E+03	4.6E+05	3.9E+03	6.2E+05	7.5E-01
La-140	2.1E+03	1.1E+05	5.5E-02	2.1E+05	6.3E-03
Ce-141	2.3E+02	1.8E+04	9.1E+02	2.3E+04	7.8E-02
Ce-143	6.1E+02	3.2E+04	2.2E-03	6.2E+04	1.3E-03
Ce-144	6.7E+02	5.7E+04	2.3E+04	6.7E+04	6.1E-01
Na-24	2.7E+05	1.3E+07	4.6E-06	2.7E+07	1.1E-01
Cr-51	5.7E+10	1.8E+10	5.9E+08	8.6E+12	5.2E+05
Mn-54	4.9E+10	1.6E+10	6.0E+09	7.4E+12	1.4E+06
Fe-55	5.0E+10	1.7E+10	8.5E+09	7.6E+12	1.6E+06
Fe-59	2.0E+09	6.4E+08	4.3E+07	3.0E+11	2.7E+04
Co-58	4.8E+10	1.6E+10	1.9E+09	7.3E+12	8.5E+05
Co-60	2.4E+10	8.0E+09	4.4E+09	3.6E+12	8.0E+05
Zn-65	3.1E+03	2.6E+05	9.7E+04	3.1E+05	2.7E+00
Ba-137m	1.5E+07	5.2E+07	1.5E+07	6.2E+07	2.8E+02
W-187	1.5E+04	7.4E+05	9.4E-04	1.5E+06	1.6E-02
Np-239	1.3E+04	7.5E+05	5.5E+00	1.3E+06	7.8E-02

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Table 12.2-13 (1 of 2)

CVCS Tank Inventories, Maximum Values (Bq)

Nuclide	Reactor Drain	Equipment Drain	Volume Control	Holdup	Reactor Makeup Water	Boric Acid Storage
H-3	1.6E+12	1.7E+11	2.0E+12	1.9E+14	1.8E+14	4.1E+13
N-16	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Kr-85m	1.4E+11	2.3E+10	3.8E+12	8.4E+09	3.8E+03	0.0E+00
Kr-85	9.1E+12	4.0E+11	1.7E+13	1.9E+11	1.3E+08	1.3E+09
Kr-87	6.7E+10	1.1E+10	2.6E+12	1.9E+09	2.5E+02	0.0E+00
Kr-88	2.5E+11	4.1E+10	8.0E+12	1.2E+10	3.3E+03	0.0E+00
Xe-131m	3.8E+12	6.0E+10	9.5E+12	1.5E+10	1.3E+07	4.6E+07
Xe-133m	1.2E+11	2.1E+10	5.8E+11	9.0E+09	7.2E+04	3.1E+03
Xe-133	1.9E+14	3.4E+12	6.2E+14	5.7E+11	2.3E+08	2.0E+08
Xe-135m	3.7E+10	6.4E+09	6.7E+11	1.3E+08	4.3E+00	0.0E+00
Xe-135	8.6E+11	1.4E+11	1.2E+13	4.4E+10	4.9E+04	5.8E-09
Xe-137	8.2E+09	1.4E+09	7.1E+10	9.0E+05	6.9E-02	0.0E+00
Xe-138	3.2E+10	5.6E+09	5.5E+11	1.0E+08	3.0E+00	0.0E+00
Br-84	1.2E+09	2.0E+08	2.6E+07	1.1E+06	1.2E-01	0.0E+00
Rb-88	1.4E+11	2.4E+10	1.4E+11	1.0E+08	3.2E+01	0.0E+00
Sr-89	3.7E+08	4.2E+07	1.0E+07	1.6E+08	2.2E+05	1.6E+09
Sr-90	2.6E+07	2.9E+06	6.9E+05	1.3E+07	3.4E+04	8.1E+08
Sr-91	3.1E+08	5.1E+07	1.5E+07	1.3E+07	1.2E+02	4.8E-08
Y-91m	1.7E+08	2.9E+07	3.9E+08	1.5E+08	7.2E+01	0.0E+00
Y-91	5.5E+07	6.2E+06	7.6E+07	2.5E+09	3.2E+06	2.5E+10
Y-93	7.4E+06	1.2E+06	1.8E+07	7.6E+07	4.1E+02	4.6E-07
Zr-95	1.8E+08	2.0E+07	4.8E+06	7.7E+07	1.2E+05	1.0E+09
Nb-95	5.8E+07	6.5E+06	1.6E+06	2.4E+07	2.7E+04	1.5E+08
Tc-99m	1.0E+10	1.7E+09	5.0E+08	2.3E+08	1.5E+03	5.8E-15
Mo-99	2.3E+10	3.1E+09	8.8E+08	5.0E+09	3.3E+05	2.3E+07
Ru-103	2.0E+07	2.2E+06	5.5E+05	8.2E+06	9.8E+03	5.9E+07
Ru-106	8.8E+06	9.8E+05	2.3E+05	4.1E+06	1.0E+04	1.8E+08
Ag-110m	6.1E+08	6.8E+07	1.6E+07	2.8E+08	6.5E+05	1.1E+10

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Table 12.2-13 (2 of 2)

Nuclide	Reactor Drain	Equipment Drain	Volume Control	Holdup	Reactor Makeup Water	Boric Acid Storage
Te-129m	6.7E+08	7.5E+07	9.3E+06	2.5E+08	2.8E+04	1.5E+09
Te-129	3.8E+08	6.6E+07	9.0E+06	9.6E+05	1.8E-01	0.0E+00
I-131	2.4E+11	2.9E+10	3.9E+09	7.0E+10	2.0E+06	2.5E+10
Te-131m	2.0E+09	3.0E+08	4.4E+07	2.5E+08	5.5E+02	1.2E+03
Te-131	6.6E+08	1.2E+08	1.4E+07	4.4E+05	4.3E-02	0.0E+00
Te-132	1.7E+10	2.2E+09	3.1E+08	3.6E+09	2.9E+04	3.8E+07
I-132	4.0E+10	7.0E+09	9.9E+08	2.3E+08	7.8E+01	0.0E+00
I-133	2.4E+11	3.7E+10	5.5E+09	2.2E+10	3.6E+04	1.6E+03
I-134	2.4E+10	4.2E+09	5.7E+08	4.4E+07	6.9E+00	0.0E+00
Cs-134	4.2E+10	4.7E+09	2.8E+10	7.6E+09	2.1E+07	4.4E+11
I-135	1.3E+11	2.1E+10	3.1E+09	2.3E+09	2.0E+03	1.3E-11
Cs-136	5.0E+09	5.8E+08	3.8E+09	7.2E+08	3.8E+05	9.0E+08
Cs-137	4.9E+10	5.4E+09	3.2E+10	8.9E+09	2.6E+07	6.2E+11
Ba-140	4.2E+08	4.9E+07	1.3E+07	1.5E+08	7.0E+04	1.6E+08
La-140	1.0E+08	1.5E+07	4.3E+06	1.8E+07	5.9E+02	2.3E+03
Ce-141	1.7E+07	1.9E+06	4.8E+05	6.9E+06	7.3E+03	3.8E+07
Ce-143	3.0E+07	4.4E+06	1.3E+06	4.5E+06	1.2E+02	9.3E+01
Ce-144	5.3E+07	5.9E+06	1.4E+06	2.5E+07	5.7E+04	9.8E+08
Na-24	1.2E+10	1.9E+09	5.6E+08	8.8E+08	1.1E+04	1.9E-01
Cr-51	6.1E+09	6.9E+08	1.7E+08	2.4E+09	2.3E+06	1.1E+10
Mn-54	7.5E+08	8.4E+07	2.0E+07	3.5E+08	8.3E+05	1.5E+10
Fe-55	5.7E+08	6.3E+07	1.5E+07	2.7E+08	7.1E+05	1.5E+10
Fe-59	1.4E+08	1.5E+07	3.7E+06	5.7E+07	7.3E+04	4.8E+08
Co-58	2.1E+09	2.4E+08	5.7E+07	9.2E+08	1.5E+06	1.3E+10
Co-60	2.5E+08	2.8E+07	6.6E+06	1.2E+08	3.2E+05	7.2E+09
Zn-65	2.4E+08	2.7E+07	6.4E+06	1.1E+08	2.5E+05	4.1E+09
Ba-137m	4.9E+10	5.4E+09	3.2E+10	8.9E+09	2.6E+07	6.2E+11
W-187	6.8E+08	1.0E+08	3.0E+07	8.1E+07	1.5E+03	4.0E+01
Np-239	6.9E+08	9.4E+07	2.7E+07	1.4E+08	7.3E+03	2.3E+05

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Table 12.2-14 (1 of 2)

Boric Acid Concentrator Package Inventories, Maximum Values (Bq)

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

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Table 12.2-14 (2 of 2)

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

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Table 12.2-15

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

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

(1) Includes detector and associated cable.

Table 12.2-16

Shutdown Cooling System Specific Source Strength

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

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Table 12.2-17

Fission and Corrosion Product Activities in the Spent Fuel Pool

Nuclide	Activity (Bq/g)	Nuclide	Activity (Bq/g)
H-3	4.0E+04	Te-129	0.0E+00
N-16	0.0E+00	I-131	1.6E+02
Kr-85m	0.0E+00	Te-131m	7.3E-01
Kr-85	0.0E+00	Te-131	0.0E+00
Kr-87	0.0E+00	Te-132	9.9E+00
Kr-88	0.0E+00	I-132	9.4E-05
Xe-131m	0.0E+00	I-133	6.1E+01
Xe-133m	0.0E+00	I-134	0.0E+00
Xe-133	0.0E+00	Cs-134	8.1E+01
Xe-135m	0.0E+00	I-135	1.7E+00
Xe-135	0.0E+00	Cs-136	6.2E+00
Xe-137	0.0E+00	Cs-137	1.1E+02
Xe-138	0.0E+00	Ba-140	2.8E-01
Br-84	0.0E+00	La-140	4.9E-02
Rb-88	0.0E+00	Ce-141	1.1E-02
Sr-89	2.6E-01	Ce-143	1.2E-02
Sr-90	8.5E-02	Ce-144	8.4E-02
Sr-91	1.6E-02	Na-24	1.8E+00
Y-91m	0.0E+00	Cr-51	4.0E+00
Y-91	6.5E-01	Mn-54	1.3E+00
Y-93	1.7E-03	Fe-55	1.5E+00
Zr-95	1.3E-01	Fe-59	9.4E-02
Nb-95	3.9E-02	Co-58	1.6E+00
Tc-99m	8.1E-02	Co-60	7.4E-01
Mo-99	1.3E+01	Zn-65	3.5E-01
Ru-103	1.3E-02	Ba-137m	1.1E+02
Ru-106	1.6E-02	W-187	2.1E-01
Ag-110m	9.1E-01	Np-239	3.8E-01
Te-129m	4.4E-01		

Table 12.2-18 (1 of 3)

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

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

Table 12.2-18 (2 of 3)

Isotope	SG Water (Bq/g)	SG Steam (Bq/g)	Blowdown Mixed Bed (Bq)	Blowdown Pre-Filter (Bq)	Blowdown Post-Filter (Bq)	Condensate (Bq/g)	CPS Cation Bed (Bq)	CPS Mixed Bed (Bq)
Cs-134	4.77E+00	2.39E-02	1.55E+12	0.00E+00	0.00E+00	1.08E-02	5.27E+08	5.63E+08
Cs-136	6.50E-01	3.25E-03	2.36E+10	0.00E+00	0.00E+00	1.48E-03	6.65E+07	3.75E+07
Cs-137	5.52E+00	2.76E-02	1.94E+12	0.00E+00	0.00E+00	1.25E-02	6.10E+08	6.61E+08
Cr-51	6.78E-01	3.39E-03	5.20E+10	4.73E+10	4.73E+08	1.54E-03	7.23E+07	5.60E+07
Mn-54	7.85E-02	3.93E-04	2.28E+10	2.07E+10	2.07E+08	7.14E-05	3.46E+06	3.63E+06
Fe-55	5.88E-02	2.94E-04	1.95E+10	1.77E+10	1.77E+08	5.35E-05	2.60E+06	2.79E+06
Fe-59	1.47E-02	7.36E-05	1.73E+09	1.57E+09	1.57E+07	1.34E-05	6.37E+05	5.57E+05
Co-58	2.25E-01	1.13E-03	3.71E+10	3.37E+10	3.37E+08	2.05E-04	9.82E+06	9.28E+06
Co-60	2.60E-02	1.30E-04	8.90E+09	8.09E+09	8.09E+07	2.36E-05	1.15E+06	1.24E+06
Zr-95	1.91E-02	9.53E-05	2.96E+09	2.69E+09	2.69E+07	2.70E-05	1.29E+06	1.21E+06
Zn-65	2.50E-02	1.25E-04	6.90E+09	6.27E+09	6.27E+07	2.27E-05	1.10E+06	1.15E+06
N-16	7.38E-01	3.69E-03	7.09E+03	0.00E+00	0.00E+00	6.71E-04	5.52E+01	5.52E+00
Na-24	2.08E+00	1.04E-02	3.64E+09	0.00E+00	0.00E+00	1.89E-03	2.72E+07	2.83E+06
Sr-89	4.03E-02	2.02E-04	5.35E+09	0.00E+00	0.00E+00	1.46E-04	6.98E+06	6.28E+06
Sr-90	2.70E-03	1.35E-05	9.48E+08	0.00E+00	0.00E+00	1.00E-05	4.87E+05	5.27E+05
Sr-91	5.33E-02	2.67E-04	6.01E+07	0.00E+00	0.00E+00	1.93E-04	1.85E+06	1.87E+05
Y-91m	1.47E-02	7.36E-05	1.43E+06	0.00E+00	0.00E+00	5.28E-05	4.39E+04	4.39E+03
Y-91	5.95E-03	2.98E-05	8.64E+08	0.00E+00	0.00E+00	2.13E-05	1.02E+06	9.35E+05
Y-93	1.28E-03	6.41E-06	1.52E+06	0.00E+00	0.00E+00	4.50E-06	4.54E+04	4.58E+03
Nb-95	6.41E-03	3.20E-05	6.11E+08	0.00E+00	0.00E+00	2.29E-05	1.08E+06	8.98E+05

Table 12.2-18 (3 of 3)

Isotope	SG Water (Bq/g)	SG Steam (Bq/g)	Blowdown Mixed Bed (Bq)	Blowdown Pre-Filter (Bq)	Blowdown Post-Filter (Bq)	Condensate (Bq/g)	CPS Cation Bed (Bq)	CPS Mixed Bed (Bq)
Mo-99	3.42E+00	1.71E-02	2.68E+10	0.00E+00	0.00E+00	1.23E-02	4.25E+08	8.24E+07
Tc-99m	1.70E+00	8.48E-03	1.18E+09	0.00E+00	0.00E+00	6.30E-03	3.75E+07	3.75E+06
Ru-103	2.15E-03	1.08E-05	2.28E+08	0.00E+00	0.00E+00	7.91E-06	3.75E+05	3.19E+05
Ru-106	9.16E-04	4.58E-06	2.74E+08	0.00E+00	0.00E+00	3.38E-06	1.64E+05	1.73E+05
Ag-110m	6.38E-02	3.19E-04	1.77E+10	0.00E+00	0.00E+00	5.80E-05	2.81E+06	2.93E+06
Te-129m	7.32E-02	3.66E-04	6.78E+09	0.00E+00	0.00E+00	2.66E-04	1.26E+07	1.03E+07
Te-129	3.91E-02	1.96E-04	5.24E+06	0.00E+00	0.00E+00	1.43E-04	1.63E+05	1.63E+04
Te-131m	3.35E-01	1.68E-03	1.17E+09	0.00E+00	0.00E+00	1.20E-03	2.89E+07	3.60E+06
Te-131	3.69E-02	1.85E-04	1.79E+06	0.00E+00	0.00E+00	1.34E-04	5.59E+04	5.59E+03
Te-132	2.39E+00	1.20E-02	2.18E+10	0.00E+00	0.00E+00	8.62E-03	3.12E+08	6.71E+07
Ba-137m	1.65E-01	8.25E-04	8.17E+05	0.00E+00	0.00E+00	6.29E-04	2.66E+04	2.66E+03
Ba-140	5.02E-02	2.51E-04	1.79E+09	0.00E+00	0.00E+00	1.78E-04	8.02E+06	4.48E+06
La-140	1.65E-02	8.25E-05	7.74E+07	0.00E+00	0.00E+00	6.08E-05	1.71E+06	2.44E+05
Ce-141	1.88E-03	9.40E-06	1.70E+08	0.00E+00	0.00E+00	6.66E-06	3.14E+05	2.56E+05
Ce-143	4.87E-03	2.44E-05	1.87E+07	0.00E+00	0.00E+00	1.82E-05	4.62E+05	5.98E+04
Ce-144	5.50E-03	2.75E-05	1.57E+09	0.00E+00	0.00E+00	1.91E-05	9.28E+05	9.71E+05
W-187	1.15E-01	5.73E-04	3.19E+08	0.00E+00	0.00E+00	1.04E-04	2.16E+06	2.48E+05
Np-239	1.05E-01	5.23E-04	6.86E+08	0.00E+00	0.00E+00	9.51E-05	3.10E+06	5.34E+05

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APR 1400 DCD TIER 2

Table 12.2-19

Gaseous Radwaste System Source Terms (0.25 % Fuel Defect)

Nuclide	At Inlet (Bq/cm ³)	Buildup Activity on Charcoal Bed (Bq)				At Outlet (Bq/cm ³)
		1st Delay Bed	2nd Delay Bed	3rd Delay Bed	4th Delay Bed	
Kr-85m	3.22E+05	3.67E+12	1.43E+11	5.53E+09	2.15E+08	7.32E-01
Kr-85	1.38E+06	5.32E+13	5.32E+13	5.32E+13	5.32E+13	1.38E+06
Kr-87	2.48E+05	8.35E+11	8.93E+06	9.54E+01	1.02E-03	3.24E-15
Kr-88	6.99E+05	5.22E+12	3.10E+10	1.84E+08	1.09E+06	8.65E-04
Xe-131m	1.37E+06	4.99E+14	2.59E+14	1.35E+14	6.98E+13	9.98E+04
Xe-133m	8.24E+04	1.11E+13	3.16E+11	8.99E+09	2.55E+08	5.37E-02
Xe-133	8.88E+07	2.29E+16	5.18E+15	1.17E+15	2.65E+14	2.32E+05
Xe-135m	1.82E+05	1.23E+11	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	1.83E+06	4.40E+13	5.02E+04	5.73E-05	6.54E-14	3.10E-30
Xe-137	4.19E+04	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	1.55E+05	9.69E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	7.37E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	9.16E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	2.44E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	1.32E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.54E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	7.44E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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Table 12.2-20 (1 of 3)

Liquid Radwaste System Tank Source Terms (Bq/cm³)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
Br-84	6.27E+01	8.62E+01	1.96E+00	8.88E-04
I-131	7.94E+03	1.09E+04	2.48E+02	9.66E+00
I-132	2.13E+03	2.93E+03	6.66E+01	1.92E-01
I-133	1.13E+04	1.55E+04	3.52E+02	5.79E+00
I-134	1.30E+03	1.79E+03	4.07E+01	3.02E-02
I-135	6.40E+03	8.80E+03	2.00E+02	1.15E+00
Rb-88	7.33E+03	1.01E+04	2.29E+02	2.90E-01
Cs-134	1.13E+03	1.55E+03	3.52E+01	7.81E+00
Cs-136	1.54E+02	2.12E+02	4.81E+00	9.84E-01
Cs-137	1.30E+03	1.79E+03	4.07E+01	9.04E+00
Na-24	5.79E+02	7.96E+02	1.81E+01	2.25E-02
Cr-51	1.75E+02	2.41E+02	5.48E+00	2.34E-02
Mn-54	2.03E+01	2.79E+01	6.34E-01	2.81E-03
Fe-55	1.52E+01	2.09E+01	4.75E-01	2.11E-03
Fe-59	3.81E+00	5.24E+00	1.19E-01	5.16E-04
Co-58	5.82E+01	8.01E+01	1.82E+00	7.96E-03
Co-60	6.72E+00	9.24E+00	2.10E-01	9.32E-04
Zn-65	6.46E+00	8.89E+00	2.02E-01	8.93E-04

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Table 12.2-20 (2 of 3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
Sr-89	1.04E+01	1.43E+01	3.26E-01	1.42E-03
Sr-90	6.98E-01	9.59E-01	2.18E-02	9.68E-05
Sr-91	1.54E+01	2.12E+01	4.81E-01	3.96E-04
Y-91m	8.99E+00	1.24E+01	2.81E-01	2.52E-04
Y-91	1.54E+00	2.12E+00	4.81E-02	2.22E-04
Y-93	3.68E-01	5.06E-01	1.15E-02	9.97E-06
Zr-95	4.93E+00	6.78E+00	1.54E-01	6.73E-04
Nb-95	1.66E+00	2.28E+00	5.18E-02	2.43E-04
Mo-99	8.99E+02	1.24E+03	2.81E+01	8.68E-02
Tc-99m	5.22E+02	7.17E+02	1.63E+01	7.79E-02
Ru-103	5.57E-01	7.66E-01	1.74E-02	7.52E-05
Ru-106	2.37E-01	3.26E-01	7.40E-03	3.28E-04
Ag-110m	1.65E+01	2.27E+01	5.15E-01	2.28E-03
Te-129m	1.89E+01	2.60E+01	5.92E-01	2.55E-03
Te-129	2.01E+01	2.77E+01	6.29E-01	1.66E-03
Te-131m	8.99E+01	1.24E+02	2.81E+00	5.93E-03
Te-131	3.55E+01	4.88E+01	1.11E+00	1.10E-03
Te-132	6.27E+02	8.62E+02	1.96E+01	6.36E-02
Ba-137m	1.18E+03	1.63E+03	3.70E+01	8.45E+00
Ba-140	1.30E+01	1.79E+01	4.07E-01	1.66E-03

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Table 12.2-20 (3 of 3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
La-140	4.38E+00	6.03E+00	1.37E-01	1.09E-03
Ce-141	4.86E-01	6.69E-01	1.52E-02	6.53E-05
Ce-143	1.30E+00	1.79E+00	4.07E-02	9.10E-05
Ce-144	1.42E+00	1.95E+00	4.44E-02	1.96E-04
W-187	3.10E+01	4.27E+01	9.70E-01	1.76E-03
Np-239	2.76E+01	3.79E+01	8.62E-01	2.50E-03

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Table 12.2-21 (1 of 3)

Liquid Radwaste System Component Source Terms (Bq)

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

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Table 12.2-21 (2 of 3)

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

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Table 12.2-21 (3 of 3)

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

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Table 12.2-22

Solid Radwaste System Tank Source Terms (Bq)

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank	Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank
Na-24	7.01E+11	7.16E+10	Rh-106	0.00E+00	2.77E+09
Cr-51	9.49E+12	1.86E+11	Ag-110m	9.29E+12	1.92E+10
Mn-54	1.45E+13	2.37E+10	Ag-110	0.00E+00	2.50E+08
Fe-55	3.40E+13	1.79E+10	Te-129m	1.31E+12	2.05E+10
Co-58	8.28E+12	6.60E+10	Te-129	1.92E+09	1.32E+10
Fe-59	3.34E+11	4.20E+09	I-129	0.00E+00	1.03E+01
Co-60	2.38E+13	7.90E+09	Te-131m	2.12E+11	2.20E+10
Zn-65	3.57E+12	7.53E+09	Te-131	1.21E+09	4.10E+09
Br-84	2.73E+09	3.46E+08	I-131	4.04E+11	6.61E+12
Rb-88	1.15E+11	2.27E+10	Te-132	4.04E+12	3.37E+11
Sr-89	1.01E+12	1.16E+10	I-132	4.04E+11	3.78E+11
Y-89m	0.00E+00	1.16E+06	I-133	1.92E+13	1.95E+12
Sr-90	3.96E+12	8.23E+08	I-134	9.30E+10	1.08E+10
Y-90	0.00E+00	4.97E+08	Cs-134	1.20E+15	1.32E+12
Sr-91	1.20E+10	1.23E+09	I-135	3.54E+12	3.58E+11
Y-91m	9.40E+07	7.83E+08	Cs-136	2.34E+12	1.45E+11
Y-91	1.63E+07	1.84E+09	Cs-137	4.60E+15	1.53E+12
Y-93	3.80E+06	3.10E+07	Ba-137m	4.60E+15	1.43E+12
Zr-93	0.00E+00	3.13E-01	Ba-140	3.21E+11	1.23E+10
Zr-95	6.34E+11	5.56E+09	La-140	1.40E+10	1.07E+10
Nb-95m	0.00E+00	5.63E+07	Ce-141	3.11E+10	5.23E+08
Nb-95	1.10E+11	2.25E+09	Ce-143	3.51E+09	3.50E+08
Mo-99	4.81E+12	4.37E+11	Pr-143	0.00E+00	1.01E+08
Tc-99m	2.50E+11	4.03E+11	Ce-144	9.18E+11	1.66E+09
Tc-99	0.00E+00	2.33E+04	Pr-144	0.00E+00	1.65E+09
Ru-103	4.32E+10	6.10E+08	W-187	6.01E+10	6.07E+09
Rh-103m	0.00E+00	6.07E+08	Np-239	1.30E+11	1.18E+10
Ru-106	2.01E+11	2.77E+09			

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Table 12.2-23 (1 of 4)

Airborne Radioactivity Concentrations

Reactor Containment Building (Normal Operation)

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

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Table 12.2-23 (2 of 4)

Reactor Containment Building (48 hrs after Shutdown)

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

Table 12.2-23 (3 of 4)

Auxiliary Building Cubicles (Normal Operation)

Cubicle	Airborne Radioactivity Concentration (Bq/cm ³)			Derived Air Concentration (DAC) Fraction			
	Kr, Xe	Br, I	H-3	Kr, Xe	Br, I	H-3	Total
CS Pump and Miniflow Hx Rm	1.46E-01	9.74E-07	1.94E-02	3.93E-02	1.03E-02	2.63E-02	6.66E-02
SI Pump Rm	4.52E-02	3.05E-07	5.99E-03	1.22E-02	3.18E-04	8.09E-03	2.06E-02
Floor Drain Sump Pump Rm	1.84E+00	5.34E-05	4.10E-02	7.99E-01	2.61E-02	5.54E-02	8.81E-01
Pipe Chase and Valve Rm (055-A14C)	9.06E-02	5.91E-07	1.22E-02	2.44E-02	6.37E-04	1.65E-02	4.15E-02
Shutdown Cooling Hx Rm	3.49E-02	2.28E-07	4.70E-03	9.38E-03	2.45E-04	6.36E-03	1.60E-02
Charging Pump Rm	1.59E+00	2.80E-07	3.56E07	6.81E-01	1.40E-04	4.82E-02	7.29E-01
Charging Pump Miniflow Hx Rm	4.49E-01	1.23E-05	1.01E-02	1.81E-01	6.37E-03	1.37E-02	2.01E-01
Equipment Drain Tank Rm	8.48E-02	1.77E-06	3.59E-04	3.62E-02	1.03E-03	4.85E-04	3.77E-02
Reactor Drain Pump Rm	2.70E+00	8.76E-06	2.11E-03	7.95E-01	5.20E-03	2.85E-03	8.04E-01
Gas Stripper Rm	2.14E+00	6.11E-07	3.53E-07	9.02E-01	3.04E-04	4.76E-07	9.02E-01
Filter and Demi. Valve Area	1.90E+00	4.85E-05	3.67E-02	7.92E-01	2.37E-02	4.96E-02	8.65E-01
SFP Clean up Pump Rm	6.23E-02	4.29E-07	7.75E-03	1.68E-02	4.50E-04	1.05E-02	2.77E-02
Reactor Makeup Water Pump Rm	2.49E-07	1.36E-12	1.20E-02	6.57E-08	1.81E-09	1.63E-02	1.63E-02
Holdup Pump Rm	5.29E-04	5.88E-08	1.19E-02	2.95E-04	6.30E-04	1.61E-02	1.65E-02
Volume Control Tank Rm	1.06E+00	1.62E-07	2.41E-021	3.88E-01	9.22E-05	3.26E-02	4.20E-01
Valve Rm (120-A23A)	1.81E-01	5.27E-06	5.01E-03	7.89E-02	2.58E-03	6.78E-03	8.83E-02
Fuel Handling Area (Normal Operation)			2.40E-02			3.20E-02	3.20E-02
Fuel Handling Area (Refueling)			5.70E-02			7.70E-02	7.70E-02

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Table 12.2-23 (4 of 4)

Compound Building Cubicles (Normal Operation)

Cubicle	Airborne Radioactivity Concentration (Bq/cm ³)			Derived Air Concentration (DAC) Fraction			
	Kr, Xe	Br, I	H-3	Kr, Xe	Br, I	H-3	Total
Valve Rm (063-P07)	1.62E+00	1.03E-04	8.29E-02	6.66E-01	5.24E-02	1.12E-01	8.31E-01
Equipment Waste Pump Rm	1.74E+00	5.01E-05	3.89E-02	7.47E-01	2.47E-02	5.25E-02	8.24E-01
Equipment Waste Tank Rm	1.16E-01	3.30E-06	4.89E-04	4.89E-02	1.64E-03	6.61E-04	5.12E-02
Floor Drain Pump Rm	1.79E+00	5.16E-05	4.01E-02	7.67E-01	2.55E-02	5.42E-02	8.47E-01
Normal Sump Pump Rm	9.49E-02	2.78E-06	2.12E-03	4.17E-02	1.35E-03	2.86E-03	4.59E-02
Chemical Waste Pump Rm	4.66E-02	1.35E-06	1.04E-03	2.02E-02	6.61E-04	1.41E-03	2.22E-02
Floor Drain Tank Rm	4.74E-01	1.30E-05	1.07E-02	1.92E-01	6.73E-03	1.44E-02	2.13E-01
Chemical Waste Tank Rm	1.07E-02	2.83E-07	2.42E-04	4.15E-03	1.51E-04	3.27E-04	4.62E-03
RLS Drain Tank and Pump Rm	0.00E+00	5.85E-08	1.02E-05	0.00E+00	3.33E-05	1.38E-05	4.71E-05
Chemical Drain Sump Pump Rm	1.00E-01	2.92E-06	2.24E-03	4.37E-02	1.42E-03	3.02E-03	4.81E-02
Monitor Tank Rm	0.00E+00	1.08E-08	0.00E+00	0.00E+00	1.00E-05	0.00E+00	1.00E-05
Monitor Tank Pump Rm	0.00E+00	2.07E-07	0.00E+00	0.00E+00	1.84E-04	0.00E+00	1.84E-04
Valve Rm (085-P06)	0.00E+00	1.98E-05	1.62E-02	0.00E+00	1.02E-02	2.19E-02	3.20E-02
Valve Rm (085-P15)	0.00E+00	4.61E-05	3.64E-02	0.00E+00	2.31E-02	4.92E-02	7.23E-02
Valve Rm (085-P16)	2.06E-01	5.96E-06	4.59E-03	8.91E-02	2.92E-03	6.21E-03	9.82E-02

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Table 12.2-24

Source Terms for Post-Accident Shielding Analysis

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

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

(2) Nuclide Group 1: Xe, Kr

Nuclide Group 2: I, Br

Nuclide Group 3: Cs, Rb

Nuclide Group 4: Te, Sb, Se

Nuclide Group 5: Sr, Ba

Nuclide Group 6: Co, Ru, Rh, Pd, Mo, Tc

Nuclide Group 7: Am, Cm, La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y

Nuclide Group 8: Ce, Pu, Np

Table 12.2-25 (1 of 3)

Radioactive Source Dimensions and Parameters Used in Shielding Analysis

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm ³)	Material	Thickness (cm)
Reactor Containment Building	Pressurizer	Cylinder	244.48	—	700.66	Water: 100 %	0.59	Steel	12.38
	Reactor Coolant Pump	Cylinder	185.00	—	126.74	Water: 100 %	0.75	Steel	14.00
	Reactor Drain Tank	Cylinder	Liquid: 162.90 Vapor: 99.70	—	528.57	Water: 27 % Vapor: 73 %	1.00 0.001293	Not considered	
	Regenerative HX	Cylinder	24.69	—	400.69	Water: 85 % Steel: 15 %	0.85 1.18	Steel	2.22
	Letdown HX	Cylinder	45.72	—	341.36	Water: 88 % Steel: 12 %	0.88 0.95	Steel	2.54
	Steam Generator	Annular Cylinder	O.D: 497.80 I.D: 415.80	—	969.57	Water: 100 %	0.70	Steel	12.86
		Semisphere	472.60						
Auxiliary Building.	SC HX	Cylinder	68.58	—	803.15	Water: 94 % Steel: 6 %	0.94 0.54	Steel	1.27
	SC Miniflow HX	Cylinder	33.66	—	173.43	Water: 93 % Steel: 7 %	0.93 0.59	Steel	0.95
	Charging Pump Miniflow HX	Cylinder	38.10	—	298.70	Water: 94 % Steel: 6 %	0.94 0.50	Steel	1.27
	CS HX	Cylinder	129.54	—	701.04	Water: 94 % Steel: 6 %	0.94 0.49	Steel	1.59

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Table 12.2-25 (2 of 3)

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm ³)	Material	Thickness (cm)
Auxiliary Building	CS MiniFlow HX	Cylinder	31.75	—	186.06	Water: 94 % Steel: 6 %	0.94 0.45	Steel	0.95
	Equipment Drain Tank	Cylinder	193.59	—	610.87	Water: 50 % Vapor: 50 %	1.00 0.001293	Not considered	
	Boric Acid Concentrator	Cylinder	Liquid: 193.53 Vapor: 206.58	—	180.52	Water: 47 % Vapor: 53 %	1.00 0.001293	Not considered	
	SC HX	Cylinder	137.16	—	803.15	Water: 94 % Vapor: 6 %	0.942 0.453	Steel	1.27
	SFP Clean-up Demi.	Cylinder	145.70	—	144.17	Water: 100 %	1.00	Not considered	
	Boric Acid Condensate IX	Cylinder	74.60	—	206.17	Water: 100 %	1.00	Not considered	
	Deborating IX	Cylinder	105.08	—	104.49	Water: 100 %	1.00	Not considered	
	Pre-Holdup IX	Cylinder	52.54	—	104.49	Water: 100 %	1.00	Not considered	
	Purification IX	Cylinder	52.54	—	104.49	Water: 100 %	1.00	Not considered	
	SFP Cooling HX	Rectangular Parallelepiped	31.19	134.16	198.28	Water: 67 % Steel: 33 %	0.67 2.63	Not considered	
	Volume Control Tank	Cylinder	120.72	—	218.09	Water: 40 % Vapor: 60 %	1.00 0.001293	Not considered	
	SGBD Flash Tank	Cylinder	152.40	—	455.96	Water: 100 %	1.00	Not considered	
	SGBD HX	Cylinder	42.43	—	487.68	Water: 86 % Steel: 14 %	0.90 1.12	Steel	1.27

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Table 12.2-25 (3 of 3)

Building	Component	Source Dimension				Source Characteristic		Housing	
		Shape	Diameter (or Width) (cm)	Length (cm)	Height (cm)	Material	Partial Density (g/cm ³)	Material	Thickness (cm)
Auxiliary Building	Spent Fuel Pool	Rectangular Parallelepiped	869.00	1113.50	381.00	Water: 70% UO ₂ : 22% Zircaloy: 8%	0.70 1.98 0.56	Not considered	
	Cask Loading Pit	Rectangular Parallelepiped	20.23	20.23	381.00	Water: 58% UO ₂ : 30% Zircaloy: 12%	0.58 2.76 0.79	Not considered	
	GRS Header Drain Tank	Cylinder	45.72	—	172.48	Vapor: 100%	0.001293	Not considered	
Compound Building	Chemical Waste Tank	Cylinder	304.80	—	466.91	Water: 100%	1.00	Not considered	
	Floor Drain Tank	Cylinder	358.14	—	676.38	Water: 100%	1.00	Not considered	
	Equipment Drain Tank	Cylinder	358.14	—	676.38	Water: 100%	1.00	Not considered	
	Low Activity Spent Resin Tank	Cylinder	274.32	—	383.33	Water: 100%	1.00	Not considered	
	Spent Resin Long Term Storage Tank	Cylinder	243.84	—	482.92	Water: 100%	1.00	Not considered	
	LRS IX	Cylinder	120.17	—	124.83	Water: 100%	1.00	Not considered	
	Waste Drum Storage	Rectangular Parallelepiped	601.98	782.57	262.89	Carbon: 100%	2.62	Not considered	

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Table 12.2-26 (1 of 8)

Assumptions and Parameters Used in Airborne Source Term Calculations

Reactor Containment Building

Assumptions/Parameters	Values
Fuel Failure	0.25 % Fuel Defect
RCS Leak rate	1.89 L/min (0.5 gpm) (Design Basis Leak Rate)
RCS Letdown Flow	364 L/min (80 gpm)
Iodine Spike	Not Considered
Low Volume Purge Flow Rate	2.549E+03 m ³ /hr (1,500 cfm) (Assumed that LVPS operates for 100 hours before shutdown)
High Volume Purge Flow Rate	9.14E+04 m ³ /hr (54,000 cfm) (Assumes that HVPS operates at cold shutdown operation mode for 29 hours)
Filter Efficiency for Low Volume Purge Filters	Particulate: 99 % Halogen: 99 %
Gas Stripping	No Gas Stripping
Containment Free Volume	8.858E+04 m ³ (3.128E+06 ft ³)

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Table 12.2-26 (2 of 8)

Fuel Handling Area in Auxiliary Building

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

Table 12.2-26 (3 of 8)

Auxiliary and Compound Building

Cubicle	Volume (m ³)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms ^{(1), (2)}	HVAC Flow (m ³ /hr)
CS Pump and Miniflow HX Rm	461	Pump seal (1) Flange (4) Valve 1" (4) Valve 4" (2) Valve 14" (1)	1.67E-04 2.00E-03 6.66E-04 1.33E-03 2.33E-03	IRWST	170
SI Pump Rm	762	Pump seal (1) Flange (3) Valve 4" (4) Valve 10" (1)	1.67E-04 1.40E-03 2.66E-03 1.67E-03	IRWST	850
Floor Drain Sump Pump Rm	94.9	Evaporation from sump	6.74E-02	0.1 PCA	680
Pipe Chase and Valve Rm (055-A14C)	1,246	Valve 18" (2) Valve 20" (1)	5.98E-03 3.33E-03	IRWST	170
Shutdown Cooling Hx Rm	1,155	Valve 1.5" (1) Valve 10" (2)	2.50E-04 3.33E-03	IRWST	170

Table 12.2-26 (4 of 8)

Cubicle	Volume (m ³)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms ^{(1), (2)}	HVAC Flow (m ³ /hr)
Charging Pump Rm	1,156	Pump Seal (1) Flange (3) Valve 1" (1) Valve 3" (4) Valve 4" (2)	8.33E-04 1.40E-03 1.67E-04 2.00E-03 1.33E-03	VCT – Halogens RCS – NG	850
Charging Pump Miniflow Hx Rm	187	Valve 2.5" (1)	4.16E-04	VCT – Halogens RCS – NG	170
Equipment Drain Tank Rm	501	Flange (2) Valve 1" (1) Valve 6" (1)	9.99E-04 1.67E-03 9.99E-04	EDT	170
Reactor Drain Pump Rm	200	Pump Seal (1) Flange (4)	8.33E-04 2.00E-03	RDT	850
Gas Stripper Rm	697	Flange (1)	5.00E-04	Gas Stripper	1,444
Filter and Demi. Valve Area	174	Valve 3" (1) Valve 3" (27)	5.00E-04 1.40E-02	RDT 1.0 PCA	1,529

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Table 12.2-26 (5 of 8)

Cubicle	Volume (m ³)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms ^{(1), (2)}	HVAC Flow (m ³ /hr)
SFP Clean up Pump Rm	224	Flange (6) Valve 4" (3) Valve 6" (4)	5.00E-03 2.00E-03 4.01E-03	SFP	510
Reactor Makeup Water Pump Rm	168	Flange (8) Pump Seal (2)	1.67E-03 4.01E-03	RMWT	340
Holdup Pump Rm	11,000	Flange (8) Pump Seal (2)	1.67E-03 4.01E-03	Holdup Tank	340
Valve Rm (120-A23A)	331	Valve 3" (1) Valve 3" (1) Valve 3" (1)	5.00E-04 5.00E-04 5.00E-04	1.0 PCA RMWT BAST	510
Valve Rm (063-P07)	762	Flange (6) Valve 2" (9) Valve 2.5" (2)	3.00E-03 3.00E-03 8.33E-04	1.0 PCA	340
Equipment Waste Pump Rm	138	Pump Seal (1) Flange (6) Valve 2" (2) Valve 3" (6)	8.44E-03 3.00E-03 6.66E-04 3.00E-03	0.32 PCA	510
Equipment Waste Tank Rm	221	Flange (2)	9.99E-04	Equipment Waste Tank	510

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Table 12.2-26 (6 of 8)

Cubicle	Volume (m ³)	Leak Sources and Number of Sources	Leak Rate (L/min)	Source Terms ^{(1), (2)}	HVAC Flow (m ³ /hr)
Floor Drain Pump Rm	413	Pump seal (1) Flange (2) Valve 2" (4) Valve 3" (12)	1.67E-02 5.98E-03 1.33E-03 5.98E-03	0.44 PCA	1,359
Normal Sump Pump Rm	68.5	Flange (4) Valve 3" (2) Evaporation from sump	2.00E-03 9.99E-04 4.92E-02	0.01 PCA	1,019
Chemical Waste Pump Rm	215	Pump seal (2) Flange (10) Valve 3" (11) Valve 2" (2)	1.67E-02 5.00E-03 5.49E-03 6.66E-04	0.01 PCA	1,104
Floor Drain Tank Rm	184	Flange (2)	9.99E-04	Floor Drain Tank	170
Chemical Waste Tank Rm	331	Flange (2)	9.99E-04	Chemical Waste Tank	170

Table 12.2-26 (7 of 8)

Cubicle	Volume (m ³)	Leak Sources and Number of Sources	Leak Rate (m ³ /min)	Source Terms ^{(1), (2)}	HVAC Flow (m ³ /hr)
Detergent Waste Tank and Pump Rm	657	Pump seal Flange (4) Valve 2" (2) Valve 3" (3) Valve 4" (7) Evaporation from sump	1.67E-05 2.00E-06 6.66E-07 1.50E-06 4.66E-06 3.51E-05	Detergent Waste Tank	170
Chemical Drain Sump Pump Rm	119	Flange (2) Valve 2" (1) Evaporation from sump	8.93E-07 3.33E-07 4.92E-05	0.01 PCA	934
Monitor Tank Rm	88.6	Flange (4)	2.00E-06	Monitor Tank	170
Monitor Tank Pump Rm	197	Pump seal (2) Flange (10) Valve 2.5" (9) Valve 3" (2) Valve 4" (14)	1.53E-05 5.00E-06 3.75E-06 9.99E-07 9.31E-06	Monitor Tank	170
Valve Rm (085-P06)	180	Valve 4" (1)	6.66E-07	1.0 PCA (Except for NG)	170

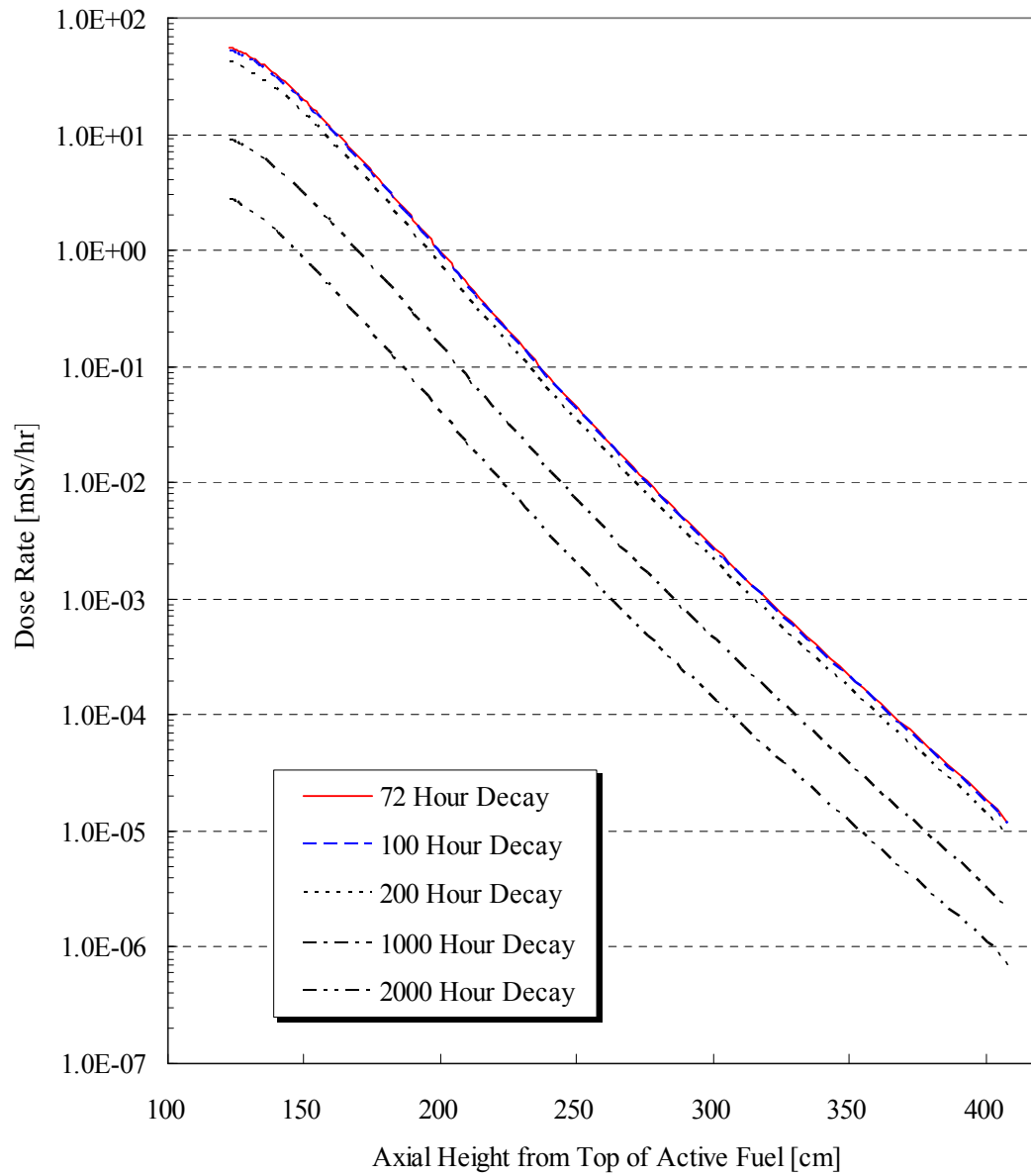
Table 12.2-26 (8 of 8)

Cubicle	Volume (m ³)	Leak Sources and Number of Sources	Leak Rate (m ³ /min)	Source Terms ^{(1), (2)}	HVAC Flow (m ³ /hr)
Valve Rm (085-P15)	263	Valve 3" (2) Valve 6" (4)	5.00E-07 4.01E-06	1 PCA (Except for NG)	510
Valve Rm (085-P16)	269	Valve 4" (1) Valve 6" (2) Valve 6" (3) Valve 6" (1) Valve 4" (2)	6.66E-07 2.00E-06 3.00E-06 9.99E-07 1.33E-06	0.32 PCA 0.32 PCA 0.1 PCA 0.44 PCA 0.01 PCA	1,444

(1) PCA: Fraction of primary coolant activity concentrations

(2) NG: Noble gases

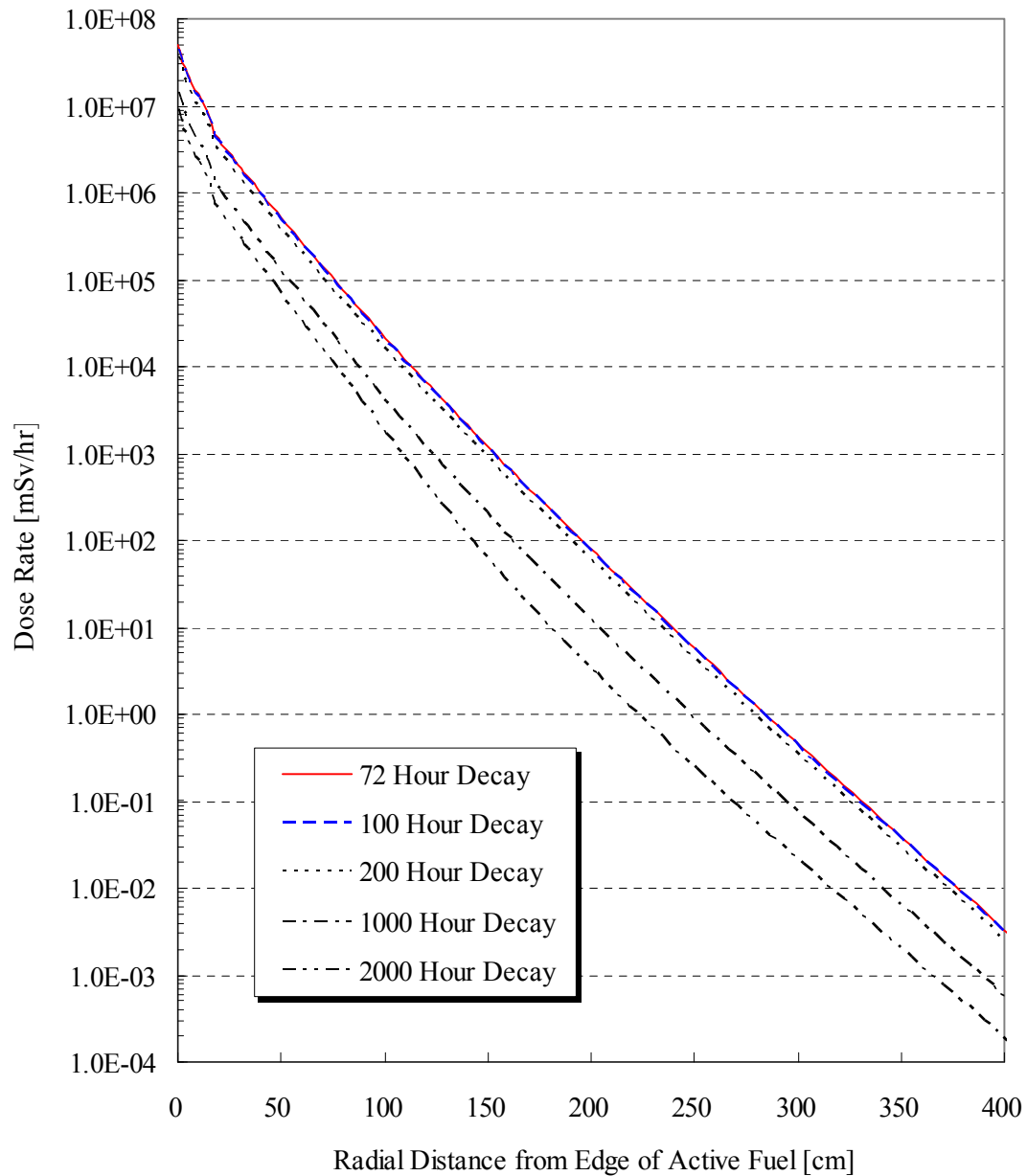
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Note : Shielding effect of the Refueling Machine is included.

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

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Note : Shielding effect of the Refueling Machine is included.

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

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12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

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

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

12.3.1.1 General Arrangement Design Features

a. Locations of radioactive systems and equipment

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

Radioactive components are located in separate compartments wherever possible. The compartment design takes into consideration the frequency of access, operational requirements, and radiation level. For example, ion exchangers containing radioactive resins are located in separate compartments, and although the valves are remotely operated, they are located in a separate shielded gallery in

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the event manual operation or access for maintenance activities is needed. The compartment walls provide shielding, which enables personnel to perform operation and maintenance activities in a lower radiation area with significantly decreased radiation exposure. This approach is consistent with NRC RG 8.8, Position 2.

b. Pipe routing

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

c. Component Access

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

d. Hot tool cribs and hot machine shop

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

The provision of a hot machine shop in the compound building enables personnel to remove equipment and perform maintenance in a lower radiation area. Access

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to the hot machine shop is also provided from the truck bay for ease of equipment movement.

e. Staging areas

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

f. Personnel decontamination and change areas

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

g. Radiation control area

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

h. Accessways and entrances to high radiation areas

Labyrinths are provided at the entrances to high-radiation areas to minimize the exposure due to scattering and streaming of radiation through entrances and piping penetrations. This approach is consistent with NRC RG 8.8, Position 2.b(4).

The plant layout design can accommodate removable shields as necessary to provide shielding during normal operation for adjacent corridors. These shields can provide additional shielding during the removal of radioactive components,

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such as heat exchangers, for maintenance activities. This approach is consistent with NRC RG 8.8 Position 2.b (2).

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

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

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

12.3.1.2 Equipment and System Design Features for Control of Onsite Exposure

The APR1400 design specifies the use of reliable and simplistic equipment with a minimal number of parts and straightforward operation to reduce the frequency of maintenance and radiation exposure to plant personnel. The design characteristics of the equipment used in radioactive systems are as follows:

a. Pumps

- 1) Pumps and associated piping are flanged to facilitate pump removal to a lower radiation area for maintenance or repair. Pump internals are also removable. This design approach is consistent with NRC RG 8.8 Position 2.b (9).
- 2) All pump casings are provided with drain connections to facilitate decontamination. The drain connections are free of internal crevices to minimize accumulation of radioactive corrosion products (crud).
- 3) Pump seals are easily serviceable without the removal of the entire pump or motor. The seals of reactor coolant pump, centrifugal charging pump, shutdown cooling pump and safety injection pump are cartridge type to facilitate their removal for maintenance or repair.

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b. Ion exchangers (demineralizers)

- 1) Ion exchangers are designed for complete drainage.
- 2) Spent resin removal is designed to be done remotely by hydraulic flushing from the vessel to the SWMS.
- 3) Piping, strainers, and resin screens are flushable so that all spent resin is removed from the ion exchange column and connected piping. This minimizes radiation exposure resulting from spent resin transfer operations.
- 4) Fresh resin is added remotely from a low-radiation area above the shielded compartment housing the ion exchanger.
- 5) Ion exchangers are designed with minimal internal crevices to prevent an accumulation of radioactive cruds. This design approach is consistent with NRC RG 8.8 Position 2.f (6).

c. Liquid filters

- 1) Filter housings are provided with vent connections and are designed for complete drainage.
- 2) Filter housings are designed with minimal internal crevices to prevent an accumulation of radioactive cruds.
- 3) Adequate space is provided near cartridge filters to allow for cartridge removal and loading for transport to the solid waste storage area.
- 4) Filter housings and cartridges are designed to permit the remote removal of spent radioactive filter elements. Cartridge filter seals are an integral part of the filter cartridge and can be removed concurrently with the filter elements. This design approach is consistent with NRC RG 8.8, Position 2.i (4).

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- 5) Cartridge filter housing closure heads are designed to swing free for the unobstructed removal of the cartridge.
- 6) Design features to handle spent filters are described in Subsection 11.4.2.

d. Tanks

- 1) Tanks are designed for complete drainage and are therefore free of internal crevices and pockets. The drain line is connected to the bottom of the tank.
- 2) Tanks are provided with at least one of the following means of decontaminating the tank internals based on tank contents and radioactivity levels:
 - a) Ample space to permit decontamination of the tank manway
 - b) Internal spray nozzles on potentially highly contaminated tanks for internal decontamination
 - c) Back flush capability for tank inlet screens
- 3) Tanks are designed with an ellipsoidal or sloped bottoms to facilitate drainage and minimize the accumulation of crud.
- 4) Tanks are provided with vents to facilitate the removal of potentially radioactive gases.
- 5) Non-pressurized tanks are provided with overflow lines routed to a floor drain pump or other suitable collection point to avoid spillage of radioactive fluids onto the floor or ground. The floor drain system (FDS) is connected to the LWMS for additional processing prior to release to the environment.
- 6) All tanks containing suspended solids are provided with mixing capabilities to prevent settling of radioactive solids.

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e. Valves

- 1) The following descriptions summarize the valve design features that minimize valve leakage and extend valve design life.
 - a) A packless valve is used as an isolation valve that is non-modulating, is less than 5.08 cm (2 in) in diameter, and operates on a weekly interval in the radioactive material processing systems to minimize the external leakage from the valves.
 - b) Modulating valves and valves greater than 5.08 cm (2 in) in diameter use live loading of the packing by conical spring washers or equivalent means to maintain a compressive force on the packing, where possible.
 - c) Double stem packing with a leakoff between the packings and a graphite lantern ring is used for valves 10.16 cm (4 in) and larger, as well as normally open valves 5.08 cm (2 in) to 10.16 cm (4 in) in diameter. Stem leakage is piped to an appropriate drain sump or tank.
 - d) Valves using stem packing are provided with a backseat capability.
 - e) Radiation-resistant seals, gaskets, and elastomers are used, when practicable, to extend the design life and reduce maintenance frequency.
 - f) Valves located in high-radiation areas are equipped with reach rods or motor operators to allow operation from lower radiation zones to minimize radiation exposure.
- 2) Fully ported valves are used so that valves are fully open and radioactive fluid freely flows. This minimizes internal accumulation of cruds.
- 3) Valves requiring removal during maintenance and inspection activities are flanged.

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- 4) Internal valve surfaces are designed to be as smooth as possible and free of crevices to minimize the accumulation of crud.
 - 5) Valve wetted parts are made of austenitic stainless steel or a similar corrosion-resistant material.
 - 6) Valves are designed so that they may be repacked without removing the yoke or topworks.
 - 7) Valves for highly radioactive components are located in shielded valve galleries to the extent possible to minimize operator exposure.
- f. Piping and penetrations
- 1) The field run piping is minimized to the extent practicable in order to reduce the potential for unintended leakage to the environment.
 - 2) Resin and concentrate piping is designed as follows:
 - a) The length of pipe runs is minimized by locating the related components as close to each other as possible.
 - b) Piping is routed through shielded pipe chases whenever possible to minimize the radiation exposure rate in personnel access corridors.
 - c) Large diameter piping (greater than 12.7 cm (5 in) in diameter) is used to minimize the potential for clogging during slurry or resin transfer without violating minimum flow requirements.
 - d) The number of pipe fittings, such as elbows and tees, is minimized to reduce the potential for radioactive crud accumulation. Where elbows are needed, large radius bends are used.
 - e) Low points, deadlegs, elbows, and vertical pipe runs that may cause accumulation of radioactive material are minimized.

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- f) Pipe runs are sloped and gravitational flow is used where practicable.
- g) Crevices on piping internal surfaces are minimized by the use of butt welds instead of socket welds. Socket welds are known to produce crud traps in radioactive systems. The use of butt welds generally results in smoother internal surfaces reducing crud buildup.
- h) Flushing capability is provided to facilitate the decontamination of piping containing radioactive fluids.
- i) Penetrations are located so that the source and penetration are not in a direct line and are located high above the floor where practicable. This minimizes the potential for personnel exposure due to radiation streaming pathways.

g. Heat exchangers

- 1) Heat exchangers are designed for complete drainage and with vents.
- 2) Internal wetted surfaces are designed to be free of crevices to minimize the potential for the accumulation of radioactive crud on internal surfaces.
- 3) Corrosion-resistant materials, such as stainless steel, are used to minimize the need for part replacement and reduce the frequency of required maintenance.

h. Reactor vessel head vent

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

i. Reactor coolant system leakage control

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

- 1) Pilot-operated safety relief valve (POS RV) leakage is directed to the in-containment refueling water storage tank (IRWST).
- 2) Valves larger than 5.08 cm (2 in) in diameter are provided with a double-packed stem with an intermediate graphite lantern ring and a leakoff connection to the reactor drain tank (RDT) with emphasis on the valve stem finish to reduce wear and leakage.
- 3) Instrumentation is provided to detect abnormal reactor coolant pump (RCP) seal leakage. The RCPs are equipped with two stages of seals, plus a vapor or backup seal, as described in Section 5.4. The vapor or backup seal prevents leakage to the reactor containment building atmosphere and allows sufficient pressure to be maintained to direct the controlled seal leakage to the volume control tank (VCT) and RDT. The vapor seal is designed to withstand full reactor coolant system (RCS) pressure in the event of failure of any or all of the two primary seals.

j. Refueling equipment

- 1) All spent fuel transfer and storage operations are designed to be conducted underwater to provide reasonable assurance of adequate shielding and limit the contribution of radioactivity in working areas.
- 2) Equipment is designed to prevent the fuel from being lifted above the minimum safe water depth, thereby limiting personnel exposure and avoiding fuel damage.
- 3) The equipment design limits the possibility of inadvertent fuel drops, which could cause fuel damage and personnel exposure.

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- 4) The refueling equipment design facilitates the transfer of new and spent fuel at the same time to reduce overall fuel handling time and personnel exposures.
- 5) Underwater cameras are used to facilitate safe handling through visual control, thus minimizing errors and potential exposures.
- 6) Portable hydraulic cutters are provided to cut expended control element assemblies (CEAs) and in-core instrumentation (ICI) leads. The cutters allow underwater handling of these items.
- 7) Equipment is provided to allow for the underwater inspection of fuel elements.

k. Inservice inspection equipment

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

l. Remote instrumentation

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

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

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

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

n. Blanket-type thermal insulation

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

o. Electrical service and lighting

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

p. Spent fuel pool (SFP) decontamination

The APR1400 design provides the capability to use high-pressure demineralized water for the decontamination in the SFP. Alternative methods of decontamination, such as a strippable coating, may be evaluated by the operator for practicality.

q. Snubbers

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

12.3.1.3 Source Term Control

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

a. Fuel performance

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

b. Corrosion product control

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

1) Primary system materials

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

Cobalt content for components with a large wetted surface area and an operating temperature of greater than 93 °C (200 °F) is restricted to a maximum of 0.1 weight percent. For components in or near the core or components that are expected to release significant quantities of corrosion products, cobalt content targets (mean values) lower than 0.1 weight percent are specified.

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

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

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

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

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

2) Primary system chemistry

Increasing the primary system water chemistry pH from 6.9 to 7.4 reduces equilibrium corrosion rates and the buildup of activated corrosion products on primary system surfaces.

12.3.1.4 Airborne Contamination Control

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

a. Leakage inside reactor containment building

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

b. Ventilation

HVAC systems are provided for individual buildings as part of the APR1400 design. Each HVAC system is designed to provide the proper ventilation flow to remove contaminated gases and vapor to levels below the limits of 10 CFR 20, Appendix B, Table 1, and to keep building areas, cubicles, walkways, and control

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areas in the proper environmental conditions (temperatures and air quality). For maximum efficiency, air flow generally is directed from clean/low contamination areas to higher contamination areas at a velocity suitable for minimizing entrainment of moisture and particulate. There is also system redundancy to minimize interruption of continuous operation. Exhausted air flows are processed through high efficiency particulate air (HEPA) filters, charcoal adsorbers, and monitors for radiological contamination before being discharged into the atmosphere. This design approach provides proper airborne contamination control and meets the requirements of NRC RG 8.8, Position 2.d. HVAC systems are addressed in Subsection 9.4.8.

c. Hot machine shop

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

d. Loop seals

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

12.3.1.5 Equipment Improvements

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

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

- 1) Use of automatic or robotic equipment to perform inspection and maintenance activities
 - 2) Adequate pull and laydown areas
 - 3) Platforms to facilitate access to SG parts for maintenance and inspection
 - 4) Handholes to facilitate worker handling of SG parts
 - 5) Adequately sized manways to facilitate easy access for workers performing maintenance and inspection activities
 - 6) Removable insulation to facilitate weld inspection
 - 7) Alloy 690 for tubes to reduce corrosion product production
- c. The APR1400 design includes features that are important to achieving ALARA goals. The features are summarized as follows:
- 1) Equipment selection that is based on reliability, maintainability, and accessibility
 - 2) Component design such as tank design, piping design, and instrument design to minimize particulate deposition
 - 3) System flushing and decontamination capability for systems and components containing radioactive materials
 - 4) Radwaste handling operations are performed remotely, wherever practicable

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- 5) Separation of radiologically contaminated components and the provision of adequate shielding surrounding the rooms/areas containing radioactive components
- 6) Controlled access to high-radiation areas via locked doors and administrative controls
- 7) Piping containing radioactive liquid, resins, or gases routed through shielded pipe chases, wherever possible

12.3.1.6 Radiation Zone Designation

In order to maintain radiation exposures ALARA throughout the layout and shielding design, the plant is divided into several radiation zones. The zones are used in the development of plant procedures to provide reasonable assurance that maintenance, inspection, and testing activities are performed in the lowest radiation zone possible. The zones indicate maximum dose rates based only on design basis activities. The radiation zone designations during normal operation are summarized in Table 12.3-2. The radiation zone maps for the auxiliary/reactor containment building are shown in Figures 12.3-1 through 12.3-9. The radiation zone maps for the compound building are shown in Figures 12.3-10 through 12.3-16.

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

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

The monitoring systems provide continuous monitoring of the SG performance and while the condensate system and blowdown system process secondary water during SG leakage

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events; the activity of the spent resins and filter cartridges are continuously monitored and are not anticipated to be significantly contaminated by PTS leakage. The condensate polisher room and the condenser areas are also provided with area radiation monitors to provide reasonable assurance that personnel radiation exposure is maintained ALARA.

12.3.1.7 General Design Considerations to Keep Post-Accident Exposures ALARA

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

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

12.3.1.8 Post-Accident Radiation Zones

Post-accident radiation zone maps are developed in accordance with NRC RG 1.183 (Reference 9) to assess the access throughout the plant following a DBA. The layout of the plant facilities is designed with the intent of keeping occupational doses ALARA during a DBA as well as during normal operation. Post-accident source terms are described in Subsection 12.2.3 and Table 12.2-24.

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

- a. Main control room
- b. Technical support center

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

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- a. Post-accident sampling system
- b. Instrumentation and control (I&C) systems and equipment room
- c. Electric equipment room
- d. Remote shutdown room

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

- a. Safety injection system
- b. Containment spray system
- c. Shutdown cooling system
- d. Chemical and volume control system

The zone limits are summarized for the design basis loss-of-coolant accident (LOCA) in Table 12.3-3. Post-accident radiation zone maps are shown in Figures 12.3-17 through 12.3-48.

12.3.1.9 Vital Area Access

The following descriptions detail the post-accident access routes to the vital areas listed in Subsection 12.3.1.8. Vital area access routes are illustrated by arrows in Figures 12.3-17 through 12.3-48. During normal operating conditions, the radiation control area has a single point of access in the compound building at elevation 30.48 m (100.0 ft). However, during post-accident conditions, emergency access and egress are possible on all elevations if needed.

- a. Main control room (MCR)

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The MCR is occupied continuously during post-accident situations. The design dose for this area does not exceed 0.15 mSv/hr averaged over 30 days for a TEDE of 50 mSv. The MCR is located in the auxiliary building at an elevation of 47.5 m (156.0 ft). Access to the MCR is via the compound building. It is generally accessible on all elevations from stairwells and elevators located at 14-15, AH-AJ, and AB-AD. The location of the MCR and associated access way are shown in Figure 12.3-37.

b. Technical support center (TSC)

The TSC is continuously occupied during post-accident situations. The design dose for this area does not exceed 0.15 mSv/hr averaged over 30 days for a TEDE of 50 mSv. The TSC is located in the auxiliary building at an elevation of 47.5 m (156.0 ft). The TSC is generally accessible on all elevations from stairwells and elevators located at 14-15, AH-AJ, and AB-AD. The location of the TSC and associated access are shown in Figure 12.3-37.

c. Post-accident sampling system (PASS)

The PASS area is irregularly accessed, not continuously occupied, during post-accident situations. The design dose for these areas may not exceed the TEDE of 50 mSv. The post-accident sampling room is located in the auxiliary building at an elevation of 16.7 m (55.0 ft) and the analysis room is in the compound building is at an elevation of 24.5 m (85.0 ft). The location of the post-accident sampling room and associated access way are shown in Figure 12.3-17.

d. Remote shutdown room (RSR)

The RSR is accessed when there is remote shutdown after post-accident situations. The design dose for this area may not exceed the TEDE of 50 mSv. The RSR is located in the auxiliary building at El. 137.5 ft. The location of the RSR and associated access way are shown in Figure 12.3-33.

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12.3.1.10 Minimization of Contamination and Radioactive Waste Generation

The APR1400 design includes features and operational programs to comply with 10 CFR 20.1406 and NRC RG 4.21 in order to minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste throughout the lifecycle of the facility.

The design includes the control measures that address the requirements in Regulatory Position C.1 through C.4, and Appendix A of NRC RG 4.21. The control measures are designed to work in conjunction with the operational procedures for the collection, processing, sampling, storing, monitoring, and disposal of radioactive waste during normal operation, including AOOs. The COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach (COL 12.3 (1)). The design also takes into account the life-cycle planning of the facility and through COL 12.3(1), identifies the operational programs and procedures to address maintenance of structures, systems, and components, site radiological environmental monitoring, and documentation of operational incidents for decommissioning planning associated with NRC RG 4.21. The APR1400 systems with the potential to contain radiological contamination are listed in Table 12.3-7 and require full NRC RG 4.21 evaluations.

Because NRC RG 4.21 control measures cover different phases of the plant life cycle, the APR1400 design divides the control measures into the following six design and operational objectives that are used to evaluate the compliance of the measures with 10 CFR 20.1406 and NRC RG 4.21:

- a. Objective 1 – Prevention/Minimization of Unintended Contamination
- b. Objective 2 – Provision of Adequate and Early Leak Detection Capability
- c. Objective 3 – Reduction of Cross-Contamination, Decontamination, and Waste Generation
- d. Objective 4 – Decommissioning Planning

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- e. Objective 5 – Operations and Documentation
- f. Objective 6 – Site Radiological Environmental Monitoring

The design features in the systems that are intended to meet the objectives for minimizing contamination and the generation of radioactive waste are described in Table 12.3-8. The table also provides cross references to the DCD sections that provide further information on how the SSCs meet the requirements of 10 CFR 20.1406 and NRC RG 4.21.

12.3.1.10.1 Design Considerations

The control measures delineated in Regulatory Position C.1 through C.4 and Appendix A of NRC RG 4.21 are categorized into four design objectives that address the design features described below.

Objective 1 – Prevention/Minimization of Unintended Contamination

Objective 1 requires control measures to prevent and/or minimize radioactive contamination of the facility and environment. This objective is met by implementing design features such as:

- a. Selection of proper materials for SSCs to withstand process conditions without the generation of excess contamination, such as corrosion-resistant piping
- b. Use of containment walls or dikes around contaminated or potentially contaminated SSCs and other means of prevention/minimization such as leak detection, drain collection, double-walled piping, and piping sleeves with leak detection for piping between buildings
- c. Minimization of buried or embedded piping and drains (alternatively utilizing piping within a concrete structure or double-walled piping, both with leak-detection capabilities)
- d. Provisions for proper operational interlocks and alarms and administrative controls to avoid inadvertent bypasses and releases

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- e. Minimization of the potential for component failures through the use of nuclear-industry-proven technologies, reinforced quality control, and strict adherence to the applicable codes and standards

Objective 2 – Provision of Adequate and Early Leak Detection Capability

Objective 2 requires provisions for leak-detection capabilities, allowing prompt assessment to support a timely and appropriate response in the event of an unintended release of radioactive contamination. Leak-detection systems are included within the facility design to initiate alarm signals in the radwaste control room (where applicable) and the main control room for timely operator actions. This objective is met by the implementing the following design features:

- a. Adequate leak-detection instruments (including individual and separate leak-detection instruments for plant SSCs with a potential for leakage to the extent possible)
- b. Facility design (e.g., floors that slope toward drainage collection piping, level-detection instruments, smooth surfaces for drainage, catch basins)
- c. Drainage collection and pathways and the placement of detection instruments as close as practicable to the sources of leaks
- d. Sensitive (including smaller liquid volume collection) liquid/moisture detection instruments
- e. Facilitation of periodic calibration and replacement of leak detection instruments
- f. Adequate access pathways and spaces for prompt assessment of detected leakage and appropriate operator responses, tank overflow management, and adequate sampling points for operational analysis

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Objective 3 – Reduction of Cross Contamination, Decontamination, and Waste Generation

Objective 3 is met by providing capabilities to reduce cross contamination, the need for decontamination, and the generation of radioactive waste. The control measures that are implemented to meet this objective include:

- a. Segregation of components in accordance with contamination types, levels, and characteristics
- b. Onsite decontamination facility
- c. Flushing capabilities for components and piping that handle or transport radioactive fluid
- d. Smooth and cleanable surfaces for component fabrication and facility structures such as epoxy coating for cubicle floors
- e. Capability to temporarily hold and isolate contamination and quickly initiate operator action to provide timely and appropriate responses
- f. Life-cycle planning of waste processing, packaging, handling, storing, and shipment for disposal

Objective 4 – Decommissioning Planning

Objective 4 is met by providing features that facilitate decommissioning. The design features include:

- a. Modular construction
- b. Minimization of buried and embedded components and piping
- c. Documentation and information in a centralized area for ready recovery
- d. Removable block walls to facilitate the removal of SSCs

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- e. SSCs that minimize the generation of radioactive waste during decommissioning activities

12.3.1.10.2 Operational/Programmatic Considerations

The operational programs and procedures in 10 CFR 20.1406 must be met. The COL applicant is to address Objectives 5 and 6, described below, to meet the operational and programmatic requirements of NRC RG 4.21 Positions C.1 through C.4 COL 12.3(1).

The requirements are related to procedures and programs that support the operation and maintenance of the design features described in Objectives 1 through 4 and documentation of any incidents that may occur during the operating life of the facility. The documentation of these incidents, including spillage, leakage, overflows, and associated cleanup requirements, is an integral part of decommissioning planning.

Objective 5 – Operations and Documentation

Objective 5 is met by developing operational procedures for the following:

- a. Supplemental control of the SSCs to prevent/minimize the spread of contamination.
- b. Periodic review of operational procedures to provide reasonable assurance that the procedures reflect lessons learned and up-to-date content, including personnel qualification and training.
- c. Accurate records and documentation of design, construction, modifications, and operational incidences to facilitate decommissioning.
- d. Development of radwaste processing and management programs.
- e. Development of maintenance and surveillance programs that provide reasonable assurance that SSCs will perform as designed to prevent the spread of contamination.

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Objective 6 – Site Radiological Environmental Monitoring

Objective 6 is met by developing a conceptual site model (COL 12.3 (1)) that:

- a. Includes site-specific hydrogeology, potential migration, and groundwater transport pathways
- b. Assesses the effects of construction on hydrogeological characteristics of the site
- c. Establishes a site-specific contamination monitoring program for potential groundwater pathways from release source to receptor point

12.3.1.10.3 Summary of Design Features

The APR1400 design features that minimize facility and environmental contamination are described in the relevant sections of this DCD. Table 12.3-8 summarizes how these features meet the design objectives in accordance with the requirements of 10 CFR 20.1206 and NRC RG 4.21 and provides cross references to the relevant DCD subsections.

Many of the features are designed to work together to support the defense-in-depth strategy for the prevention and minimization of the contamination of the facility and the environment.

12.3.1.10.4 Design Details to Address Minimization of Contamination

12.3.1.10.4.1 Early Leak Detection Drain Pipe

The early leak detection design for plant areas containing radioactive components consists of a drain pipe with a level switch and a manual valve installed on the exposed pipe portion in the valve pit or corresponding location of the trench. The drain pipe collects drainage from cubicles that may contain radioactive liquid from leakage or overflow. The drain pipe is a 5.08 cm (2 in) stainless steel pipe embedded in the basemat and is sloped to facilitate drainage. The embedded portion of the drain pipe is encased in an outer pipe. The inner pipe is open to drainage into the trench, which directs the collected liquid to the normal sump for collection. Because the drain pipe is relatively small, a small leak can be

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detected, thus achieving early detection. The level switch is designed to initiate an alarm in the MCR for operator action. Early leak detection capability is built into each cubicle that may contain radioactive or potentially radioactive fluid for quick assessment. The typical design of the leak detection configuration is presented in Figure 12.3-50. The leak detection instrumentation is shown in Figure 12.3-51.

12.3.1.10.4.2 Leak Collection Trench

Concrete trenches are provided within the basemat of the auxiliary and compound buildings to facilitate leak drainage to liquid collection sumps. The trench has an epoxy coating and is significantly sloped to facilitate flow and prevent the accumulation of fluid along the flow path. The trench has removable steel grated covers to facilitate access for inspection and calibration of leak detection instruments; trench cleaning; and inspection, repair, and maintenance of the epoxy coating. Where area drainage is expected to contain significant debris, the trench is equipped with screens or a drain pipe to facilitate liquid flow without entraining solids or debris in the sumps. The typical design of the compound building trench design is presented in Figure 12.3-52.

12.3.1.10.4.3 Penetration Design between Buildings

Piping that extends from one building to another with a separation gap between the buildings is equipped with sleeves or double-walled concentric piping to prevent direct, unintended leakage to the environment. The use of either piping sleeves or double-walled concentric piping depends on the separation distances, with piping sleeves generally used for smaller separation gaps and double-walled piping used for larger separation gaps.

Piping sleeves and the outer wall piping for the double-walled piping are equipped with seals to maintain building pressure differentials and to prevent the infiltration of outside water and air into the buildings. Piping sleeves and outer piping are also equipped with sleeve joints to accommodate some degree of building movement. Piping sleeves and outer wall piping are installed with a mild slope to facilitate drainage of liquid in the event of inner pipe leakage. The leakage is drained directly into a nearby sump or onto the floor for collection in a nearby sump. The typical designs for piping penetrations between buildings and building sleeves are presented in Figure 12.3-53.

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12.3.1.10.4.4 Minimization of Embedded and/or Buried Piping

Embedded or buried piping is minimized using the following approach:

- a. Piping is routed inside pipe chases to the extent possible to minimize embedded piping. Drain pipes on the floors above the basemat level are allowed to penetrate the floor and are routed inside pipe chases below the floor level as required.
- b. To the extent practicable, double-walled piping is used when embedded piping segments cannot be avoided. The outside piping of the double wall pipe is designed to drain to a local sump or onto the floor for drainage to a nearby sump for collection.
- c. For the basemat level, drain pipes are routed in concrete trenches that are sloped towards the local sump. The trenches are coated with epoxy to facilitate drainage and cleaning.
- d. For piping that carries contaminated or potentially contaminated fluid and is located outside the plant structures, the piping is routed in underground concrete tunnels to the maximum extent practicable. The tunnels are coated with epoxy and are equipped with sumps with liquid detection level switches. If liquid is accumulated to the detectable level, an alarm is initiated in the MCR for operator actions.

12.3.2 Shielding

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

12.3.2.1 General Shielding Design Criteria

Shield walls are provided around components that contain and handle radioactive materials for worker safety and to maintain radiation doses ALARA. The thicknesses of the walls

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listed in Table 12.3-4 are based on the shielding basis source terms of the component, the design dose rate, and the shielding material. This approach is consistent with NRC RG 8.8.

The shielding design for the MCR and the primary shielding in the reactor containment building are safety related. The shielding for the MCR meets the requirements of 10 CFR 50, Appendix A, GDC 19.

Radiation protection of personnel, equipment, and materials is dependent primarily on the adequacy of the design of the plant shielding. Radiation shielding has the passive protection function of radiation attenuation and consists of materials placed between radiation sources and plant personnel as well as the public. The shielding system is designed and constructed to provide reasonable assurance that the station can be operated and maintained so that the resultant radiation levels and doses are below the limits of 10 CFR 20 and are ALARA. Design dose rate limits to achieve this objective are addressed in Subsection 12.3.1 and listed in Table 12.3-2.

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

- a. Plant conditions

- 1) Normal operating condition

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

- a) Normal power operation of the reactor
 - b) AOOs (startup, shutdown, and refueling of the reactor)

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Shielding is designed to provide a protective function under such conditions. Normal operation zone designations are provided in Table 12.3-2.

2) Accident conditions

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

a) MCR habitability

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

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

b) Direct offsite doses

Adequate shielding is provided to limit accident exclusion area boundary doses due to direct and scattered radiation from contained sources within the plant to the limits specified in 10 CFR 50.34 (Reference 11) and Chapter 15 of the Standard Review Plan (SRP) (Reference 12).

b. Seismic and safety classification

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

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

c. Protection of equipment and structures

Adequate shielding is provided for the following purposes:

- 1) To limit radiation heating of structural concrete
- 2) To reduce neutron activation of equipment
- 3) To limit the radiation dose to equipment and materials

d. Maintenance, inspection, and testing considerations

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

e. Additional requirements

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

12.3.2.2 Shielding Analysis

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

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The only major source generating radionuclides is the reactor core at full power. The codes ANISN (Reference 13) and MCNP (Reference 14) are used to verify the effectiveness of the primary shield. Sources of gamma radiation are distributed throughout the reactor containment building and nuclear island. The codes MICROSHIELD (Reference 15) and RUNT-G (Reference 16) are used to verify gamma source shielding. The following sequence typifies a gamma source shielding analysis:

- a. Determine the concentration of each principal nuclide in the source medium.
- b. Adjust the concentration to account for issues such as accumulation, dilution, decay, and removal.
- c. Convert the resulting concentrations into gamma source strength.
- d. Select a model or combination of models to represent the physical geometry of the source container(s) and all shields present.
- e. Assemble the necessary data on attenuation properties of the source and the shield materials.
- f. Perform the calculation for the desired dose point location and tabulate the results for comparison with design objective dose rates.

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

- a. ANISN

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

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b. MCNP

MCNP is a continuous energy, generalized geometry, time-dependent, coupled neutron-photon-electron Monte Carlo transport code system. MCNP is widely used for verification of shielding design adequacy in dose rate reduction because it provides more realistic modeling capabilities and more reliable calculation results than other programs. MCNP is used in shielding calculations as described in Section C.1.a of NRC RG 1.69.

c. MICROSHIELD

MICROSHIELD is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used for designing shields, estimating source strength from radiation measurements, and teaching shielding principles. MICROSHIELD has 16 geometries that accommodate offset dose points and as many as 10 standard shields, plus source self-shielding and cylinder cladding. This code uses library data for radionuclides, attenuation, and buildup and the dose conversion reflects standard data from ICRP 38 and 107, as well as ANSI/ANS standards and RSICC publications.

d. RUNT-G

RUNT-G uses a modified version of the multiple compartment model from the RACER-II program to calculate time-dependent radionuclide activities, the source geometry models from the Gaussian quadrature version of the ISOSHLD (Reference 17) program to calculate the resulting time dependent dose rates, and a simple trapezoidal integration scheme to determine cumulative doses. All of the containment model parameters used in RACER-II and shielding geometries available in ISOSHLD are included in RUNT-G.

12.3.2.3 Shielding Design

The plant shielding is designed to achieve the radiation zones designated in Tables 12.3-2 and 12.3-3 for normal operation and post-accident conditions, respectively.

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In accordance with Standard Review Plan 12.3-4, Sections II.1 and III.3 and Appendix B, and RG 8.38 (Reference 18) C.1.6(3), adequate shielding or operational precautions are provided for all accessible plant areas capable of radiation levels greater than 1 Gy/hr. Areas with the potential for radiation greater than 1 Gy/hr are listed in Table 12.3-5.

Transient sources greater than 1 Gy/hr are considered in the shielding design to provide reasonable assurance that adequate shielding is provided. One such source is a spent fuel assembly. During transfer of a spent fuel assembly through the fuel transfer tube, adjacent areas may experience elevated radiation levels. Streaming from this source up through the joint between the reactor containment building and the auxiliary building has been a concern for the current generation of nuclear plants. The APR1400 design uses connected building structures to reduce the potential for streaming. In addition, sufficient concrete shielding is provided to maintain radiation levels in adjacent areas ALARA during spent fuel transfer. This permits personnel to perform maintenance and inspection activities in a lower radiation area and reduces the potential for high radiation levels adversely affecting refueling outage schedules. An inspection area is provided for the fuel transfer tube. Access control is provided by the personnel airlock through the reactor containment building.

Sufficient shielding provides reasonable assurance that the areas adjacent to the spent fuel transfer tube are accessible and expected radiation zones are consistent with those in Figure 12.3-49 during transfer of a spent fuel assembly.

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

The ICI chase is potentially a high-radiation area (greater than 1 Gy/hr) during ICI withdrawal. Stringent access control is provided to this area during movement of the ICI.

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A lockable access door is provided with a warning light. During withdrawal of the ICI, the warning light illuminates providing indication that the ICI is being moved. An area radiation monitor is located in the ICI chase to provide indication of radiation levels and to alarm the personnel when the ICI is being withdrawn. Emergency egress from the area is also provided from the ICI chase.

Components that handle a significant amount of radioactive materials, such as LWMS floor drain tanks and equipment drain tanks, are located in shielded cubicles provided with pump and valve galleries and labyrinths for access to the galleries. This design approach minimizes radiation streaming and scattering but permits inspection and maintenance access and removal of smaller items such as pumps, valves, and instruments for repair in lower radiation areas. This design approach meets the requirements of NRC RG 8.8 2.b(4). The plant shielding is designed not only to maintain personnel occupational exposure ALARA, but also to maintain exposure to the general public ALARA.

The APR1400 shielding design has target dose rates that are below the limits for radiation zone designations provided in Table 12.3-2 in order to provide a sufficient margin in maintaining radiation exposure to plant personnel and the public ALARA.

12.3.3 Ventilation

The spread of airborne contamination within the plant is minimized by the design of the plant HVAC systems to provide airflow from areas of lower potential for airborne contamination to areas of greater potential for airborne contamination. For building compartments with the potential for contamination, the exhaust from the areas is designed with pressure and flow balances to minimize the amount of uncontrolled exfiltration from these areas. These design features provide reasonable assurance that the average concentration of radioactive material in the air in the areas that are normally occupied is less than the small fraction of DAC prescribed in 10 CFR 20 Appendix B for areas that are normally accessible such as radiation zone 2. Therefore, personnel exposure due to inhalation of and contact with airborne contamination is maintained ALARA.

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

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

HVAC systems are described in Section 9.4.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation monitoring system (ARMS) supplements the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and provides reasonable assurance of compliance with the personnel radiation protection requirements of 10 CFR 20, 10 CFR 50, 10 CFR 70 (Reference 19); the guidelines of NRC RGs 1.21 (Reference 20), 1.97, 8.2 (Reference 21), 8.25, and 8.8 (Reference 1); and ANSI N13.1-1999 (Reference 22) and IEEE 497-2002 (Reference 23). The ARMS is in conformance with ANSI/ANS HPSSC-6.8.1 (Reference 24).

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

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

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

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12.3.4.1 Area Radiation Monitoring System

12.3.4.1.1 Design Objective

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

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

12.3.4.1.2 Location of Area Radiation Monitors

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

12.3.4.1.3 General System Description

Area radiation monitors consist of microprocessors and Geiger-Mueller (G-M) tubes or ionization chambers for detecting gamma radiation. Each monitor using microprocessors may be configured with one or more detectors and may cover multiple areas. Selection of detectors is based on the range needed for the particular monitoring application. Some

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areas may require extended or high-range detector configurations to cover special operational or post-accident monitoring functions.

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

Table 12.3-6 lists the area radiation monitors and specifies the range, electrical class, and seismic category of each monitor.

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

- a. Areas that are typically high-radiation areas but require little or no access (e.g., pipe chases) are not provided with an area radiation monitor.
- b. Areas that typically have a high frequency of access and are normally low-radiation areas, but are potentially high-radiation areas, are provided with an area radiation monitor. In accident conditions, areas such as corridors outside personnel and equipment hatches may become very-high-radiation areas and are therefore provided with an area radiation monitor to provide reasonable assurance of worker safety.

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

12.3.4.1.4 Redundancy, Diversity, and Independence

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

12.3.4.1.5 ARMS

The ARMS monitors presented in Table 12.3-6 are categorized into subsystems used for process monitoring functions or other monitoring applications. The location and flow diagrams of the monitors are shown in Figure 11.5-1.

The ARMS monitors are as follows:

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

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

The high-range containment area monitors (RE-233A, 234B) consist of physically independent and electrically separated detectors located inside the containment away from the influence of the reactor coolant system to measure high range gamma radiation. This monitor gives operators a seismically and environmentally qualified indication of containment airborne activity. These

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monitors comply with the requirements of 10 CFR 50.34(f)(2)(xvii) and the criteria in Attachment 3 to Item II.F.1 of NUREG-0737 and NRC RG 1.97.

Two fuel handling accident monitors (RE-241A, 242B) are provided in the SFP area. One is located on a wall and the others near the SFP bridge area.

- b. Non-safety-related area monitors (RE-205, 235, 237, 238, 245, 257, 275, 279, 284, 285, 286, 289, 292, 293, and 242B)

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

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

An area monitor is provided for the main control room (RE-275) and Technical Support Center (PR-RE-279) to measure gross gamma dose rates.

Other areas where an area monitor is provided are the in-core instrument area (RE-235), hot machine shop area (RE-293), new fuel storage area (RE-245), low level lab area (RE-257), calibration facility area (RE-286), waste drum storage area (RE-292), compound building truck bay area (RE-289), and dry active waste storage area (RE-284) in compound building.

12.3.4.1.6 Range and Alarm Setpoints

The ranges of the ARMS are shown on Table 12.3-6. Alarm setpoints for safety-related monitors are determined by plant procedures and the Offsite Dose Calculation Manual (ODCM). The setpoint methodology includes the relationship between the analytical limit, setpoint, and channel uncertainty. The setpoint methodology also provides and channel uncertainty calculations associated with the setpoints used for ESF actuation functions. The setpoint methodology follows the methodology in ANSI/ISA-67.04-1994 (Reference 27).

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12.3.4.1.7 Calibration Methods and Frequency

The methodology to determine the calibration methods and frequency of the ARMS is provided by the Offsite Dose Calculation Manual based on plant procedures.

12.3.4.1.8 Power Supplies

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

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

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

- a. High-energy line break area HVAC exhaust monitor
- b. Auxiliary building controlled area common HVAC exhaust monitor
- c. Containment purge exhaust monitor
- d. Containment air monitor
- e. Fuel handling area HVAC exhaust monitor
- f. Condenser vacuum vent exhaust monitor

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- g. Steam packing exhaust monitor
- h. Main control room air intake monitor
- i. Compound building HVAC exhaust monitor
- j. Miscellaneous process monitors

12.3.4.2.1 Design Objectives

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

12.3.4.2.2 Location of Airborne Radioactivity Monitors

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

12.3.4.2.3 System Description

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

12.3.5 Dose Assessment

The dose assessment is described in Section 12.4.

12.3.6 Combined License Information

COL 12.3(1) The COL applicant is to provide operational procedures and programs, including the development of a site radiological environmental monitoring program, to implement the minimization of contamination approach.

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

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12.3.7 References

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3. 10 CFR 20, "Standards for Protection Against Radiation."
4. NRC RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," Nuclear Regulatory Commission, June 2008.
5. NRC RG 8.25, "Air Sampling in the Workplace," Nuclear Regulatory Commission, June 1992.
6. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."
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11. 10 CFR 50.34, "Contents of Applications, Technical Information."
12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Nuclear Regulatory Commission.
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Table 12.3-1

Cobalt Contents of Finished Surface for Primary System Materials

Component	Cobalt Content
Reactor Vessel Assembly Cladding	0.05 w/o (max)
Reactor Coolant Pipe	0.05 w/o (max)
Pressurizer Assembly Cladding	0.05 w/o (max)
Steam Generator Cladding	0.05 w/o (max)
Reactor Coolant Pump	0.1 w/o (max)
Core Support Barrel Assembly	0.05 w/o (max)
Lower Support Structure Assembly	0.05 w/o (max)
Instrument Nozzle Assembly	0.05 w/o (max)
Upper Guide Structure Assembly	0.05 w/o (max)
Inner Barrel Assembly	0.05 w/o (max)
Core Shroud Assembly	0.05 w/o (max)

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Table 12.3-2

Normal Operation Radiation Zone Designations

Zone Designation ⁽¹⁾	Design Dose Rate (mSv/hr) ⁽²⁾	Zone Description
1	$DR \leq 0.001$	Uncontrolled, unlimited access
2	$0.001 < DR \leq 0.025$	Controlled, limited access, 40 hr/wk
3	$0.025 < DR \leq 0.05$	Controlled, limited access, 20 ~ 40 hr/wk
4	$0.05 < DR \leq 0.2$	Controlled, limited access, 5 ~ 20 hr/wk
5	$0.2 < DR \leq 1$	Controlled, limited access, 1 ~ 5 hr/wk
6	$1 < DR \leq 10$	Controlled access under supervision of radiation protection personnel
7	$10 \leq DR \leq 5,000$	Controlled access under supervision of radiation protection personnel
8	$DR > 5,000$	Controlled access under supervision of radiation protection personnel

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

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

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Table 12.3-3

Post-Accident Radiation Zone Designations and Design Dose Rates

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

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Table 12.3-4 (1 of 3)

Design Basis Radiation Shield Thicknesses

Room number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Reactor Containment Building							
069-C01	UGS Laydown Area and ICI Cavity	60	60	79	60	-	-
078-C01	Reactor Cavity	80	80	79	79	-	-
080-C01	Holdup Volume Tank	36	36	-	79	-	-
100-C03	Reactor Drain Tank	24	24	48	24	-	-
100-C04	Letdown Hx Room	24	24	48	24	-	-
128-C01	Regenerative Hx Room	60	60	48	48	-	-
Auxiliary Building							
050-A04A	SC Pump and Miniflow Hx. Room	8	20	20	22	-	30
050-A04B	SC Pump and Miniflow Hx. Room	20	21	22	25	-	30
055-A14C	Pipe Chase and Valve Room	16	12	38	16	-	16
055-A14D	Pipe Chase and & Valve Room	21	21	6	21	-	21
055-A18A	Pipe Chase and Valve Room	7	9	7	38	-	16
055-A18B	Pipe Chase and Valve Room	15	48	11	6	-	21
055-A21A	Pipe Chase and Valve Room	9	23	25	7		29
055-A22A	Pipe Chase	9	9	15	48	-	9
055-A22B	Pipe Chase	9	9	24	48	-	9
055-A30A	SC Hx. Room	22	35	35	20	-	23
055-A30B	SC Hx. Room	35	27	35	22	-	29
055-A34A	Floor Drain Sump Pump Room	7	22	6	22	-	18
055-A43A	Charging pump Mini Flow HX Room	15	18	6	15	-	19
055-A51B	Equipment Drain Tank Room	12	28	37	26	-	24
055-A52B	Reactor Drain Pump Room	13	12	6	13	-	6

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Table 12.3-4 (2 of 3)

Room number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Auxiliary Building (continue)							
055-A53B	Reactor Drain Pump Room	13	12	23	6	-	6
055-A54B	Aux. Charging Pump Room	6	14	14	12	-	13
055-A55B	Charging Pump Room	14	6	14	40	-	13
077-A01A	Reactor Drain Filter Pit	27	6	6	14	24	23
077-A09A	SGBD Filter Pit	6	6	7	8	27	23
077-A10A	Seal Injection Filter Pit	9	9	9	8	27	23
077-A11A	SGBD Filter Pit	6	6	7	6	15	23
077-A12A	Seall Injection Filter Pit	9	3	6	9	18	24
077-A13A	SGBD Filter Pit	27	6	7	6	15	23
077-A14A	Boric Acid Filter Pit	13	16	6	29	17	17
077-A15A	Filter Cartridge Storage	32	39	39	39	18	39
078-A33A	SGBD Demin. Room	25	7	25	7	7	23
078-A34A	Pre-Holdup Ion Exch. Room	33	15	15	33	21	24
078-A37A	Deborating Ion Exch. Room	15	12	8	26	10	24
100-A35B	Loading and Unloading Area	11	11	11	1	-	-
119-A01B	Refueling Canal	-	62	62	62	-	-
120-A16A	Mechanical Penetration Room	60	3	22	48	24	30
120-A14A	SGBD Regen. Hx Room	3	12	9	19	19	24
137-A19A	SG Blowdown Flash Tank Room	21	21	11	21	21	21
174-A15B	Containment High/Low Volume Purge ACU Room	25	14	14	25	18	23

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Table 12.3-4 (3 of 3)

Room number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Compound Building							
063-P02	GRS Header Drain Tank Room	19	35	47	31	-	18
063-P13	Hot Pipe Chase	28	14	28	14	-	24
063-P21	Equipment Waste Pump Room	8	19	8	21	-	8
063-P25	Floor Drain Pump Room	5	8	7	12	-	10
063-P30	Chemical Waste Tank Room	-	12	10	7	-	18
063-P39	Spent Resin Long Term Storage Tank Sump Pump Room	34	34	12	34	-	-
063-P41	Concentrate Holding Tank Room	28	14	37	28	-	1
063-P44	IX Feed Tank Room	14	14	12	8	-	16
077-P01	Hot Pipe Way	27	35	35	21	33	-
085-P04	Charcoal Guard Bed Room	10	12	12	4	4	8
085-P21	Charcoal Guard Bed Room	10	12	4	24	4	8
085-P06	Valve Room	26	10	19	27	10	35
085-P08	Valve Room	14	29	22	13	20	24
085-P44	RO Feed Tank Room	20	6	32	20	11	19
085-P47	MF Membrane Module Room	24	14	-	12	7	21
085-P48	RO Membrane Module & Valve Skid Room	48	24	48	34	35	20
096-P01	Charcoal Delay Bed Room	33	22	46	7	-	37
096-P02	Charcoal Delay Bed Room	33	19	7	10	-	31
100-P02	GRS Equip. Removal Area	13	13	10	8	7	8

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Table 12.3-5

Areas Potentially Greater than 1 Gy/hr⁽¹⁾

Auxiliary and Reactor Containment Building
Elevation of 55.0 ft; see Figure 12.3-1

Area	Coordinates
ICI Cavity	F-G, 18-19

Auxiliary and Reactor Containment Building
Elevation of 78.0 ft; see Figure 12.3-3

Area	Coordinates
Pre-holdup ion exchanger pit	C-D, 23-24
Purification ion exchanger pit	B-C, 23-24
Purification filter pit	C-D, 24-25
Core debris chamber	F-G, 17-18

Auxiliary and Reactor Containment Building
Elevation of 100.0 ft; see Figure 12.3-4

Area	Coordinates
Reactor cavity	E-G, 18-20
Steam generator cavity area	D-E and G-H, 19
Reactor drain tank room	E-F, 16-17
Area	Coordinates

Auxiliary and Reactor Containment Building
Elevation of 120.0 ft; see Figure 12.3-5

Refueling pool area	E-G, 17-21
Fuel transfer tube ⁽²⁾	F-G, 21-23
Spent fuel pool ⁽²⁾	G-H, 23-25
Cask loading pit ⁽²⁾	H-I, 23-24
Refueling canal ⁽²⁾	F-G, 23-25

(1) During normal operating conditions and AOOs

(2) Only when fuel is in the area

Table 12.3-6 (1 of 3)

Area Radiation Monitors

Description	Tag No.	Class ⁽¹⁾				Range					Function and Remarks
		S	SE	Q	E	Airborne Particulate	Iodine	Gas	Liquid	Area ⁽²⁾ (mSv/hr)	
Post-accident primary sample room	RE-205	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Normal primary sample room	RE-285	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Main steam and FW containment piping penetration area	RE-237 RE-238	N	II	A	N	N/A	N/A	N/A	N/A	$10^0 \sim 10^5$	Alarm (MCR, local)
Fuel handling ACC and POST-ACC high-range monitor in containment	RE-231A RE-232B	3	I	Q	A B	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	<ul style="list-style-type: none"> • Alarm (MCR, local) • CPIAS
	RE-233A RE-234B	3	I	Q	A B	N/A	N/A	N/A	N/A	$10^1 \sim 10^8$	<ul style="list-style-type: none"> • Alarm (MCR, local) • CPIAS
In-core instrument	RE-235	N	II	A	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)

Table 12.3-6 (2 of 3)

Description	Tag No.	Class ⁽¹⁾				Range					Function and Remarks
		S	SE	Q	E	Airborne Particulate	Iodine	Gas	Liquid	Area ⁽²⁾ (mSv/hr)	
Containment personnel access hatch area	RE-236	N	II	A	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Spent fuel pool area	RE-241A RE-242B	3	I	Q	A B	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	<ul style="list-style-type: none"> • Alarm (MCR, local) • FHEVAS
New fuel storage area	RE-245	N	II	A	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Hot machine shop	C-RE-293	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Low level lab	RE-257	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Instrument calibration facility	C-RE-286	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Main control room area	RE-275	N	II	A	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
TSC area	RE-279	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Truck bay	C-RE-289	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Waste drum storage area	C-RE-292	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
Compound building dry active waste storage area	C-RE-284	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)

Table 12.3-6 (3 of 3)

Description	Tag No.	Class ⁽¹⁾				Range					Function and Remarks
		S	SE	Q	E	Airborne Particulate	Iodine	Gas	Liquid	Area ⁽²⁾ (mSv/hr)	
[Low radioactive waste storage area]	RE-321	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^5$	Alarm (MCR, local)
[Medium radioactive waste storage area]	RE-322	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^2$	Alarm (MCR, local)
[Medium radioactive waste storage area]	RE-323	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^3$	Alarm (MCR, local)
[Truck bay area]	RE-324	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^5$	Alarm (MCR, local)
[Crane control room]	RE-325	N	III	S	N	N/A	N/A	N/A	N/A	$10^{-3} \sim 10^5$	Alarm (MCR, local)

(1) S: safety Class per ANSI/ANS-51.1 ; 1=SC-1, 2=SC-2, 3=SC-3, N=NNS

SE: seismic Category I, II, III

E: Electrical Class A, B, C, D=Class 1E Separation Division, N=Non-Class 1E

Q: Quality Class Q, A, S

Refer to Section 3.2 for the definition.

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

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Table 12.3-7

APR1400 Plant Systems Requiring Full NRC RG 4.21 Evaluations

System Code	System Title	System Code	System Title
AF	Auxiliary Feedwater	MS	Main Steam
AT	Auxiliary Feedwater Pump Turbine	PR	Radiation Monitoring
CA	Condenser Vacuum	PS	Process Sampling
CC	Component Cooling Water	PX	Primary Sampling
CD	Condensate	RC	Reactor Coolant
CM	Containment Monitoring	RG	Reactor Coolant Gas Vent
CP	Condensate Polishing	SC	Shutdown Cooling
CS	Containment Spray	SD	Steam Generator Blowdown
CV	Chemical and Volume Control	SI	Safety Injection
CW	Circulating Water	SX	Essential Service Water
DE	Radioactive Drain	TA	Turbine Generator System
DM	Miscellaneous Building Drain	VB	Compound Building HVAC
DT	Turbine Generator Building Drain	VF	Fuel Handling Area HVAC
FC	Spent Fuel Pool Cooling and Cleanup	VK	Auxiliary Building Controlled Area HVAC
FT	Feedwater Pump Turbine	VP	Reactor Containment Building HVAC
FW	Feedwater	VQ	Reactor Containment Building Purge
GW	Gaseous Waste Management	WV	Liquid Waste Management
HD	Heater Drain	WX	Solid Waste Management
HG	Containment Hydrogen Control		
IW	In-Containment Water Storage		

Table 12.3-8 (1 of 82)

NRC RG 4.21 Design Objectives and Applicable DCD Subsection Information for
Minimizing Contamination and Generation of Radioactive Waste

System: Reactor Coolant System

Objective		SSC Control Measures ⁽¹⁾ /Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> • The RCS components are located inside the reactor containment building. The bottom floor inside the containment is lined with steel and the floors of the cubicles inside containment are sloped, coated with epoxy, and provided with drains that are routed to sumps. This design approach prevents unintended contamination of the facility and the environment. • The reactor coolant pumps (RCPs) are provided with machined flow passages in seal housing assemblies designed to limit seal leakage. • The wetted portions of the system components are fabricated from stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. 	5.4.3
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> • Subsection 5.2.6 describes the method of detecting any RCS leakage in detail. Specific items are listed below. • Indications of RCS leakage are provided by the containment sump level and flow monitor, an airborne radioactivity monitor, and an atmosphere humidity monitoring system (see Subsection 5.2.6.1). • The RCS is equipped with an Acoustic Leak Monitoring System (ALMS), described in Subsection 7.7.1. This system alarms in the MCR for operator actions to investigate the source of leakage. • As stated in Subsection 5.4.2.2.2.5, the primary-to-secondary leakage monitoring is implemented in accordance with EPRI guidelines. 	5.4.3

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Table 12.3-8 (2 of 82)

System: Reactor Coolant System

Objective		SSC Control Measures ⁽¹⁾ /Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection (Cont'd)	<ul style="list-style-type: none"> • A steam generator program is in place (Subsection 5.4.2.2) to monitor integrity of the steam generator tubes for both preventive and corrective actions, as necessary. • All of these methods provide the ability to quickly detect any RCS leakage and take operator action as necessary. In addition, the CVCS also monitors coolant inventory for leak identification. 	5.4.3
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> • The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The primary components use steel with low cobalt content as much as practicable. This approach minimizes contamination and waste generation. • The steam generators are designed in accordance with ASME code and operating experience to minimize steam generator tube degradation and rupture, thus reducing cross contamination and associated waste generation. • Process sampling connections are provided to determine levels of contamination and other water chemistry requirements. In addition, process radiation monitoring is provided within the containment vessel to detect high levels of airborne radioactivity and initiate an alarm in the MCR [also meets objective 2]. 	5.4.3

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Table 12.3-8 (3 of 82)

System: Reactor Coolant System

Objective		SSC Control Measures ⁽¹⁾ /Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal during decommissioning. The SSCs are designed with decontamination capabilities. Design features, such as the utilized welding technique, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objective 3]. The RCS is designed without any embedded or buried piping for contaminated or potentially contaminated fluid which minimizes the potential for unintended contamination of the environment [also meets Objectives 1 and 3]. 	5.4.3
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. 	5.4.3
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The RCS is located within the reactor containment building. Because of its location and associated safety design features, the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Therefore, RCS inclusion in the Site Radiological Environmental Monitoring Program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection of radiological contamination. 	5.4.3

(1) The “SSC Control Measures” consists of excerpts/summary from the referenced DCD sections.

Table 12.3-8 (4 of 82)

System: Reactor Coolant Gas Vent System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> As the system is located entirely within containment, any leakage from the system components will be collected in the RDT or IRWST inside containment. Hence, the reactor coolant gas vent system has low potential to contaminate other areas of the plant or the environment. The piping directs the vented gases to the IRWST or the reactor drain tank (RDT) and is sloped to facilitate the drainage of condensation, thus minimizing contamination. 	5.4.12
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The RDT and IRWST are equipped with level instrumentation to alert operators if a high level due to inleakage is reached. 	5.4.12
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. 	5.4.12
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. 	5.4.12
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	5.4.12

Table 12.3-8 (5 of 82)

System: Reactor Coolant Gas Vent System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none">The containment monitoring system is located within the reactor containment building, and therefore the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Therefore, the system inclusion in the Site Radiological Environmental Monitoring Program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection of radiological contamination.	5.4.12

Table 12.3-8 (6 of 82)

System: Safety Injection System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The SIS components are located inside the reactor containment building and the auxiliary building. The floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs and sumps. This design approach prevents unintended contamination of the facility and the environment. The system tanks and pumps are fabricated from stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. 	6.3.6
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The SIS is not used during normal power operation. When in use, any leakage is drained to the floor and is collected in the local sump which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator actions to investigate the source of leakage. 	6.3.6
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The pumps are equipped with drain directly routed to radioactive drain system. This design approach minimizes contamination to the facility and the environment [also meets Objective 1]. The SIS is designed to be isolated from the RCS during normal power operation to minimize cross contamination of systems. 	6.3.6

Table 12.3-8 (7 of 82)

System: Safety Injection System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for extended service life and are fabricated as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities. Design features, such as the utilized welding technique, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objective 3]. The SIS is designed with minimal embedded piping for contaminated or potentially contaminated fluid which minimizes the potential for unintended contamination of the environment [also meets Objectives 1 and 3]. 	6.3.6
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	6.3.6
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The SIS is on standby mode during normal power operation and is designed to have low levels of contamination. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the SIS is well maintained, resulting in a very low level of contamination of the facility. Hence, the SIS is not required to be part of the Site Radiological Environmental Monitoring Program. 	6.3.6

Table 12.3-8 (8 of 82)

System: Shutdown Cooling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The SCS components are located in individual cubicles inside the Auxiliary Building. The floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs and sumps. This design approach prevents unintended contamination of the facility and the environment. The SCS components (pumps, heat exchangers, piping) are fabricated from stainless steel material and are of welded construction, thus minimizing leakage and unintended contamination of the facility and the environment. 	5.4.7
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The SCS is used only in post shutdown periods. Any leakage is drained to the floor and is collected in the local sump which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator actions to investigate the source of leakage. 	5.4.7
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The pumps are equipped with drain directly routed to radioactive drain system. This design approach minimizes contamination to the facility and the environment. The SCS is provided with isolation from the RCS when the RCS is at high pressure (during normal power operation). The interlocks associated with six valves on the two SCS suction lines are provided to prevent the valves from opening in the event RCS pressure exceeds SCS operating pressure, thus minimizing cross-contamination between the systems. 	5.4.7

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System: Shutdown Cooling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for extended service life and are fabricated as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities. Design features, such as the utilized welding technique, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The SCS is designed with minimal embedded piping for contaminated or potentially contaminated fluid which minimizes the potential for unintended contamination of the environment [also meets Objectives 1 and 3]. 	5.4.7
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	5.4.7
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The SCS is on standby mode during normal power operation and is designed to have low levels of contamination. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the SCS is well maintained, resulting in a very low level of contamination of the facility. Hence, the SCS is not required to be part of the Site Radiological Environmental Monitoring Program. 	5.4.7

Table 12.3-8 (10 of 82)

System: Containment Spray System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The Containment Spray System components are located in individual cubicles inside the Auxiliary Building (AB). The cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents unintended contamination of the facility and the environment. The system heat exchangers and pumps are fabricated from stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment [also meets Objective 3]. The pump shaft seal is the mechanical type and is constructed of materials compatible with the spray fluids. A flow restrictor is provided to minimize the loss of fluid in the event of a seal failure. This design approach minimizes contamination to the facility and the environment [also meets Objective 3]. 	6.5.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The Containment Spray System is not used during normal power operation. Any leakage is drained to the floor and is collected in the local sump which is equipped with a liquid level switch. If leakage exceeds a predetermined liquid level within the sump, the level switch initiates an alarm in the MCR for operator actions to investigate the source of leakage. 	6.5.2

Table 12.3-8 (11 of 82)

System: Containment Spray System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. Shell and tube type heat exchangers are used to transfer heat from the Containment Spray System side to the Component Cooling Water System side. The heat exchangers are designed with stainless steel tubes to minimize the potential for cross-contamination between containment spray and component cooling water. The leakage from the heat exchangers is collected in the local floor drain sumps. Process sampling connections are provided to determine levels of contamination and determine treatment requirements. 	6.5.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities. Design features, such as the utilized welding technique, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The Containment Spray System is designed without any embedded or buried piping for contaminated or potentially contaminated fluid which minimizes the potential for unintended contamination of the environment [also meets Objective 1]. 	6.5.2

Table 12.3-8 (12 of 82)

System: Containment Spray System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	6.5.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The Containment Spray System is on standby mode during normal power operation and is designed to have only low levels of contamination. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the Containment Spray System is well maintained, resulting in a very low level of contamination of the facility. Because of its location, the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Hence the Containment Spray System is not required to be part of the Site Radiological Environmental Monitoring Program. 	6.5.2

Table 12.3-8 (13 of 82)

System: Containment Hydrogen Control System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The containment hydrogen control system is designed to provide control of the combustible gases, mainly hydrogen, inside the containment and the In-containment refueling water storage tank (IRWST), to within the acceptable limits through the use of the passive autocatalytic recombiners (PAR) and/or the hydrogen igniters (HI) located at various locations inside the containment and the IRWST. The PARs and HIs are designed to combust hydrogen and release the combustion products, mainly water vapor, within the containment and the IRWST. 	6.2.5
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> Radiation monitoring within containment provides indication of leakage from radiologically contaminated system. 	6.2.5
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> The Containment Hydrogen Control System is designed to be on standby mode during normal operation. Hence, the Containment Hydrogen Control System has a very low potential to contaminate other areas and systems of the facility or the environment during normal operation, including AOOs [also meets Objective 1]. There is no piping associated with the PARs and the HIs and the system has no interfaces with non-contaminated systems, except electrical power. 	6.2.5
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. 	6.2.5

Table 12.3-8 (14 of 82)

System: Containment Hydrogen Control System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	6.2.5
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The containment hydrogen control system is located within the reactor containment building, and therefore the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Therefore, the system inclusion in the Site Radiological Environmental Monitoring Program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection of radiological contamination. 	6.2.5

Table 12.3-8 (15 of 82)

System: In-Containment Water Storage System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The In-Containment Water Storage System components, primarily the in-containment refueling water storage tank (IRWST) and the holdup volume tank (HVT) located inside containment, are constructed of reinforced concrete lined with austenitic stainless steel. The IRWST and the HVT are interconnected by two spillways. The bottom of the spillway is submerged below the surface of the water in the IRWST and the top of the spillway is below the IRWST ceiling during normal operation. This limits the amount of water vapor and possible gaseous radioactivity that can enter the containment atmosphere. This design approach prevents unintended contamination of the facility and the environment. The top of the basemat inside the containment is provided with a steel liner. This design approach prevents unintended contamination of the basemat and the environment [also meets Objective 3]. The In-Containment Water Storage System uses the Spent Fuel Pool Cooling and Cleanup System (FC) components to maintain and purify the water contents, thus minimizing the spread of contamination and unintended contamination of the facility and the environment [also meets Objective 3]. The system piping is fabricated from stainless steel to minimize corrosion and erosion. Granulated tri-sodium phosphate is provided in a stainless steel basket to minimize the corrosion of the stainless steel piping during LOCA situations [also meets objectives 3 and 4]. 	6.8.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The IRWST is designed to prevent the spread of contamination due to overflow. Four wide range (0 to 100%) level indicating channels are provided locally and in the MCR. Adequate instrumentation, including temperature and pressure elements, is provided to monitor and control the operations to prevent overpressurization or vacuum, thus minimizing the spread of contamination and waste generation [also meets objectives 3 and 4]. 	6.8.2

Table 12.3-8 (16 of 82)

System: In-Containment Water Storage System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection (cont'd)	<ul style="list-style-type: none"> The HVT has one narrow range and four wide range level indicating channels. The narrow range level channel is used to detect the presence of fluid in the HVT. Upon detection of water, the narrow range channel actuates an alarm in the MCR to alert for operator actions. The reactor cavity has four wide range level indicating channels. The four wide range channels provide level indication in the MCR to facilitate monitoring fluid level after an accident. 	6.8.2
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The In-Containment Water Storage System provides water for safety injection and containment spray functions, and is also a source of water for refueling operations. During normal operation, the IRWST water can be cooled by the shutdown cooling heat exchangers and purified by the Spent Fuel Cooling and Cleanup System. The Chemical and Volume Control System provides a means to add makeup and adjust the boron concentration in the IRWST. This design approach minimizes the spread of contamination to the facility and the environment and waste generation [also meets Objective 1]. 	6.8.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life. All structures inside the IRWST are simple and smooth in contour to facilitate its lining with stainless steel. The bottom of the tank is flat to avoid having sharp corners. This design approach minimizes potential for damages to the liner which may result in unintended leakage and contamination [also meets Objective 1]. The SSCs are designed with smooth surface finishes to minimize the need for decontamination and hence waste generation during decommissioning [also meets Objective 3]. 	6.8.2

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System: In-Containment Water Storage System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The In-Containment Water Storage System is designed with adequate instrumentation for remote operation and monitoring. A maintenance access hatch is provided to facilitate inspections and the maintenance of spargers and other hardware inside the tank. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	6.8.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The In-Containment Water Storage System is designed to provide safety features for uninterrupted operation of the plant. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the In-Containment Water Storage System is well maintained, and contributes a very low level of contamination to the facility. Because of its location inside containment, the potential for environmental contamination of soil and groundwater from liquid leakages is minimized. Hence, this system is not required to be included in the site Radiological Environmental Monitoring Program. 	6.8.2

Table 12.3-8 (18 of 82)

System: Chemical and Volume Control System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system design, including the volume control tank (VCT), the equipment drain tank (EDT), and the letdown cooling and purification equipment inside the auxiliary building (AB); the reactor drain tank (RDT) located inside the reactor containment building; and the holdup tank (HT), boric acid storage tank (BAST), and reactor makeup water tank located outside the AB; are designed with stainless steel and carbon steel materials that are compatible with the chemical and radiological environment. All components are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. The CVCS is also designed to collect drainage from system components, including relief valve leakages, for purification and recycle. This design approach minimizes unintended contamination of the facility and the environment. The heat exchangers inside the containment are designed to minimize the potential for leakage by using shell and tube type heat exchangers designed for compliance with ASME III [also meets Objective 3]. Butt weld joints are used for all piping 1 inch and larger in the letdown line to the letdown strainer, the charging line from the charging pump to the RCS, the pressurizer auxiliary spray line, the reactor coolant pump (RCP) seal injection line, the RCP bleedoff line to the VCT, the boric acid makeup line from the BAST to the VCT, resin sluice lines to the high activity spent resin tank, and the boric acid concentrate line from the HT through the holdup pumps to the BAST or solid waste management system. This design minimizes leakage from the CVCS piping carrying radioactive material. 	9.3.4

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System: Chemical and Volume Control System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination (Cont'd)	<ul style="list-style-type: none"> The piping tunnel connection from the yard tanks to the AB is designed and construction to have above ground building entrance and penetrations, therefore minimizing underground piping penetrations. This design approach minimizes potential for the infiltration of water and the spread of contamination to the environment. 	9.3.4
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The CVCS is equipped with instrumentation to assist in the detection of: Reactor Coolant System (RCS) leakage, pressurizer level, VCT level, RDT level, EDT level, letdown flowrate, and charging flowrate. The VCT, RDT, EDT and CVCS processing equipment are designed with sufficient capacity to provide temporary holding of primary coolant for normal operation, including anticipated operational occurrences. The tanks are equipped with level-indicating and control instruments to prevent overflow and provide reasonable assurance of timely processing, thus minimizing interruption of normal processing operation, the spread of contamination, and waste generation [also meets Objectives 1 and 3]. The cubicles in which the CVCS tanks are located are designed with leak detection instruments to initiate alarms for operator actions in the event of leakage or overflow. The leak detection design has the capability to detect a small quantity of leakage to provide early warning to operators. The piping connecting the outside tanks to the SSCs that are inside the AB is routed through a pipe tunnel. The piping tunnel is epoxy coated and has a sump with a level switch to detect piping leakage and infiltrated water. In the event that liquid is detected, the level switch sends a signal to the MCR for operator action. This design approach minimizes contamination of the environment [also meets Objectives 1 and 3]. 	9.3.4

Table 12.3-8 (20 of 82)

System: Chemical and Volume Control System

	Objective	SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> • The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. • The primary coolant purity is maintained with chemical injection for optimum reactor operation. Boric acid is added to compensate for reactivity changes, fuel burnup, and xenon variations, and to provide shutdown margin. Lithium hydroxide is added for pH control, and hydrogen is added in order to minimize the occurrence of radiolysis. The addition of chemicals helps to minimize the generation of contaminated waste. • Boric acid in the letdown flow is recovered for reuse to the maximum extent possible. In the event that the boric acid concentrate contains radioactivity exceeding a predetermined setpoint, the concentrate is sent to the liquid waste management system (LWMS) for neutralization, treatment, and release. The boric acid concentrator operates automatically to the desired boron concentration and sends the concentrate to the BAST for reuse. This design approach minimizes waste generation. • The HT, BAST, and the RMWT are located outside in a tank house to prevent the infiltration of rainwater and the spread of contamination. The tank house is designed with a sloped floor which is coated with epoxy to facilitate draining and cleaning and is equipped with a sump that has level switch instrumentation to detect leakage and overflows. In the event that leakage is detected, the level switch sends a signal to the MCR for operator action. This design approach minimizes the spread of contamination and waste generation. 	9.3.4

Table 12.3-8 (21 of 82)

System: Chemical and Volume Control System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> The process piping containing contaminated fluids is properly sized to facilitate flow with sufficient velocities to prevent the settling of solids. The piping is designed to reduce fluid traps, thus reducing the decontamination needs and waste generation. Decontamination fluid is collected and routed to the LWMS for processing and release. Utility connections are designed with a minimum of two barriers to prevent the spread of contamination to clean systems. 	9.3.4
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed with decontamination capabilities. Design features, such as integrated component packages, utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and the resulting waste generation [also meets Objectives 1 and 3]. The SSCs are designed for the full service life of the plant and are fabricated as individual assemblies for easy removal to the maximum extent practicable. The CVCS is designed with minimum embedded and buried piping. Piping between buildings is equipped with piping sleeves such that leakage is directed back to the building for collection, thus preventing unintended contamination. 	9.3.4

Table 12.3-8 (22 of 82)

System: Chemical and Volume Control System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The CVCS is designed for automated operation with manual initiation. Boron injection is controlled by the Pressurizer Level Control System to maintain the desired boron concentration. Adequate instrumentation, including level, flow, and pressure elements, as well as process sampling, is provided to monitor and control the CVCS operations to prevent undue interruption, thus minimizing the spread of contamination and waste generation. Leak detection instruments are provided to detect individual tank leakage. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control are to be prepared by the COL Applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	9.3.4
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The CVCS is part of the overall plant and is included in the Site Radiological Environmental Monitoring Program for monitoring of facility and environmental contamination. The program includes sampling and analysis of boric acid and contaminated fluid leakage from the BAST, RMWT, HT, and piping tunnel, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. The program is included as a COL Information item. 	9.3.4

Table 12.3-8 (23 of 82)

System: Steam Generator Blowdown System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The SGBS components are located in elevated cubicles inside the Auxiliary Building. The cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination through the facility and to the environment. The SGBS is designed with sufficient capacity for different modes of operation, including continuous blow down (BD), abnormal BD, high capacity BD, and emergency BD. The system piping is adequately sized to prevent blockage and is sloped to facilitate drainage and prevent crud buildup. The flash tank is constructed of carbon steel material; other components, including the heat exchangers, filters, demineralizers, and the wetted parts of the WLS recirculation pumps are fabricated from stainless steel material and utilize welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. 	10.4.8
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The SGBS is designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level, flow rate, temperature, and pressure elements, and a process radiation monitor, are provided to monitor the system operation in order to prevent undue interruption. 	10.4.8
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The SGBS components are provided with demineralized water for decontamination purposes. Nitrogen and other utilities are provided to facilitate operations. The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems. 	10.4.8

Table 12.3-8 (24 of 82)

System: Steam Generator Blowdown System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> Process sampling connections are provided to determine the levels of contamination, treatment requirements, and confirmation of the continual radiation monitoring output. Continuous process radiation monitoring is provided on the outlet line of the treated blowdown water. The detection of high radiation levels initiates automatic valve closure for isolation and operator actions, thus minimizing cross-contamination. 	10.4.8
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal to the maximum extent possible. The SSCs are designed in order to facilitate decontamination. Design features, such as the utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The SGBS is designed without any embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the AB for collection, thus preventing contamination of the environment [also meets Objectives 1 and 3]. 	10.4.8
5	Operations and Documentation	<ul style="list-style-type: none"> The removal and packaging of spent filter elements and spent resin is designed for remote manual operation. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.4.8

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System: Steam Generator Blowdown System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The SGBS is part of the overall plant and is included in the Site Radiological Environmental Monitoring Program for monitoring the potential for environmental contamination. The Program includes sampling and analysis of waste samples, meteorological conditions, hydrogeological parameters, and potential migration pathways of the radioactive contaminants. The Program is included as a COL Information item. 	10.4.8

Table 12.3-8 (26 of 82)

System: Component Cooling Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system components, including the CCWS surge tanks, pumps, and associated piping, are fabricated from carbon steel material and are of welded construction for life-cycle planning. The CCWS is injected with a corrosion inhibitor, and the surge tanks are covered with nitrogen gas to minimize the oxidation of steel and formation of corrosion. The CCWS heat exchangers are made of titanium, and pump impellers are made of stainless steel. The system is also designed with periodic sampling and analysis to maintain water quality and provisions for in-service testing and inspection to maintain system integrity. This design approach minimizes leakage and unintended contamination of the facility and the environment. The CCWS surge tanks, heat exchangers, and pumps are designed with multiple divisions and sufficient capacity to accommodate different modes of operation, including normal and anticipated operational occurrences. The tanks are designed as ASME III, Division I and are equipped with dual level instruments to facilitate control of the content liquid level, thus minimizing the spread of contamination and waste generation [also meets Objective 3]. 	9.2.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The CCWS surge tanks are designed with dual level instruments to provide reasonable assurance of safe operation and to provide alarms to the operating personnel in the event of overflow or leakage. 	9.2.2

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System: Component Cooling Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection (Cont'd)	<ul style="list-style-type: none"> Two radiation monitors, one for each division, are provided to continuously monitor contamination levels at the discharge of the CCWS pumps and indicate radiation activity in the MCR. In the event that radiation is detected above the pre-determined limit, an alarm is initiated by one of the monitors for operator action. Since the CCWS is segregated into two independent and parallel divisions, each division can be isolated for inspection, mitigation, and maintenance. This design approach provides early leak detection and minimizes the spread of contamination to other components, the facility, and the environment [also meets Objectives 1 and 3]. 	9.2.2
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The CCWS surge tanks are located at a high elevation to maintain liquid pressure in order to prevent infiltration and cross-contamination of the CCWS. This location also helps to prevent unintended contamination of the environment as any leakages from the tanks are drained to the local sumps for collection and forwarded to the LWMS for treatment and disposal [also meets Objective 1]. The cubicles, in which the CCWS SSCs are housed, are designed with sloped floors, epoxy coating to provide drainage and cleanable surfaces, and local sumps to collect leakages and overflows. Cubicle curbs are also provided to reduce cross-contamination and the spread of contamination to other areas. The CCWS heat exchangers are located in a separate structure close to the essential service water building to minimize radiation exposure to the essential service water piping. 	9.2.2

Table 12.3-8 (28 of 82)

System: Component Cooling Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> • Plate type heat exchangers are used to transfer heat from the component cooling water side to the essential service water side. The heat exchangers are designed with titanium plates to minimize the potential for pinhole leaks between component cooling water and essential service water. The gaskets between the plates are also designed to direct leakages to the outside of the heat exchangers. The leakages are then collected in the floor drain sumps and are routed for treatment and release. This design approach minimizes contamination of clean systems as well as to the facility and the environment [also meets Objective 1]. • Sampling points are provided to facilitate sampling and analyses to provide reasonable assurance that the water quality is maintained and also to determine the need for the addition of corrosion inhibitor. This design approach minimizes waste generation. • Utility connections (demineralized water and nitrogen) are designed with a minimum of two barriers to prevent the contamination of clean systems. 	9.2.2
4	Decommissioning Planning	<ul style="list-style-type: none"> • The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal to the maximum extent possible. • The CCWS is designed with minimum embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the Auxiliary Building for collection. Piping to the component cooling water heat exchanger structures is routed through Seismic Category I reinforced concrete pipe tunnels (one per division) under the yard. The tunnel is coated with epoxy and is designed with a collection sump with a level switch to initiate an alarm signal in the MCR for operator actions in the event liquid is detected. This design approach thus minimizes unintended contamination of the facility and the environment [also meets Objectives 1 and 3]. 	9.2.2

Table 12.3-8 (29 of 82)

System: Component Cooling Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> • The CCWS is designed for automated operation with manual initiation for the different modes of operation. The component cooling water surge tanks are designed with dual level instruments to provide reasonable assurance of safe operation. A high level signal, indicating in-leakage, or a low level signal, indicating out-leakage, is transmitted to the MCR for operator actions. A low-low level signal isolates the non-essential headers and the RCP headers from the remaining portion of the system, thus minimizing the spread of contamination and waste generation. • Adequate space is provided around the equipment to enable prompt assessment and responses when required. • Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. • Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	9.2.2

Table 12.3-8 (30 of 82)

System: Component Cooling Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The CCWS is considered to have a potential low level of contamination through leakage from contaminated systems through heat exchangers. Through monitoring, in-service inspection, and lessons learned from industry experiences, the integrity of the CCWS is expected to be well maintained, resulting in a minimal level of contamination. Leakage from the system to the facility and the environment is captured by the drainage collection provisions. Any residual contamination of the hydrogeology is not likely to be distinguishable from other contamination sources. Hence contamination characteristics of the CCWS are not specifically monitored in the site wide program. However, a site-wide Radiological Environmental Monitoring Program is to be included to monitor environmental contamination by the COL applicant. 	9.2.2

Table 12.3-8 (31 of 82)

System: Essential Service Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The component cooling water heat exchangers are plate type and constructed of titanium material, which minimizes pinhole leaks. 	9.2.1
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The heat exchanger seals are designed to leak towards the outside of the heat exchangers where leakage is collected in the building sump. A radiation monitor is provided at the outlet of the CCW heat exchangers to detect any radioactive leakage from the CCWS to the ESWS. This monitor is indicated and alarmed in the MCR. 	9.2.1
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> With the exception of the component cooling water system heat exchangers, the ESWS is designed not to be in contact with potentially radiologically-contaminated components. 	9.2.1
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal to the maximum extent possible. The ESWS is designed with minimum embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the Auxiliary Building for collection. This design approach thus minimizes unintended contamination of the facility and the environment [also meets Objective 1]. 	9.2.1

Table 12.3-8 (32 of 82)

System: Essential Service Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL Applicant. Documentation requirements are included as a COL Information item. 	9.2.1
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The ESWS is considered to have a potential low level of contamination through leakage from contaminated systems through heat exchangers. Hence contamination characteristics of the ESWS are not specifically monitored in the site wide program. However, a site-wide Radiological Environmental Monitoring Program is to be included to monitor environmental contamination by the COL applicant. 	9.2.1

Table 12.3-8 (33 of 82)

System: Spent Fuel Pool Cooling and Cleanup System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The SFPCCS components are located in elevated cubicles inside the Auxiliary Building. The cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination within the facility and to the environment [also meets Objective 3]. The spent fuel pool, refueling pool, refueling canal, and cask loading pit are installed with stainless steel liner plates and weld seam drain channels. Other components, including the heat exchangers, filters, demineralizers, and pumps are fabricated from stainless steel material and utilize welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment [also meets Objective 3]. 	9.1.3
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The SFPCCS is designed to include sight glasses for visual inspections. Early leak detection can be achieved through the use of sight glasses and operating procedures. Adequate instrumentation, including level, flow rate, temperature, and pressure elements, is provided to monitor the system operation to prevent undue interruption. 	9.1.3
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. 	9.1.3

Table 12.3-8 (34 of 82)

System: Spent Fuel Pool Cooling and Cleanup System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> • Plate type heat exchangers are used for thermal transfer from the spent fuel pool cooling side to the component cooling water side. The heat exchangers are designed with stainless steel plates to minimize the potential for pinhole leaks between the potentially contaminated system and component cooling water. The heat exchangers are designed for the component cooling water to operate at a higher pressure than the process fluid side, which prevents the spread of contamination to the clean system side through leakage. The gaskets between the plates are also designed to direct leakage to the outside of the heat exchangers. The leakage is then collected in the local floor drain sump. This design approach minimizes the spread of contamination to the facility and the environment [also meets Objective 1]. • The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems. 	9.1.3
4	Decommissioning Planning	<ul style="list-style-type: none"> • The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal, with the exception of the liner plates. • The SSCs are designed to facilitate decontamination. Design features, such as the utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. • The SFPCCS is designed with minimum embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the Auxiliary Building for collection, thus preventing unintended contamination to the environment [also meets Objectives 1 and 3]. 	9.1.3

Table 12.3-8 (35 of 82)

System: Spent Fuel Pool Cooling and Cleanup System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The removal and packaging of spent filter elements and spent resin is designed for remote manual operation. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	9.1.3
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The SFPCCS is designed to manage radioactive contamination through the storage of spent fuel. Through monitoring, in-service inspection, and lessons learned from industry experiences the integrity of the SFPCCS is well maintained, resulting in a consequent low level of contamination to the facility. Because of its location at higher plant elevations, the potential for environmental contamination of soil and groundwater resulting from pool liquid leakages is minimized. However, because the pool is open, contamination resulting from the evaporation of water from this and other systems is included in the Site Radiological Environmental Monitoring Program. The Program is included as a COL Information item. 	9.1.3

Table 12.3-8 (36 of 82)

System: Gaseous Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system contains sufficient charcoal material to hold the noble gas nuclides for a period of decay to reduce the release of radioactivity, thus minimizing contamination of the facility and the environment. The system design, including the waste gas dryers and guard beds, are configured into two parallel trains, one operating and one standing by, each with sufficient capacity to remove the moisture to protect the charcoal beds. A HEPA filter is provided downstream of the charcoal beds to prevent the distribution of contaminated charcoal fines. Cubicles, in which contaminated materials are stored and processed, are epoxy-coated to facilitate cleaning. The GWMS Header Drain Tank is equipped with level instrumentation to detect fluid accumulation and drain the fluid to the Radioactive Drain System. The system is designed with above-ground piping to the extent practical. Buried and/or embedded piping is minimized. Piping is sloped to facilitate drainage of condensate to the Header Drain Tank [also meets Objective 3]. The system uses valves with leak-tight characteristics, such as the bellows or metal diaphragm types, to minimize leakage. The system utilizes welded construction to the maximum practical extent to minimize leakage [also meets Objective 3]. 	11.3.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The system is designed with gas analyzers and a radiation monitor to provide reasonable assurance of the integrity of the SSCs including piping, and to provide alarms to warn the operating personnel of explosive gas concentrations. The system is designed with adequate space around all components to enable prompt evaluation and response in the event of leakage detection 	11.3.2

Table 12.3-8 (37 of 82)

System: Gaseous Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry proven equipment and materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation. The process piping containing contaminated fluid is sloped to facilitate easier flow and to reduce fluid traps, thus reducing the decontamination needs and waste generation. Decontamination fluid is collected and routed to the LWMS for processing and release. Utility connections are designed with a minimum of two barriers to prevent contamination of non-radioactive systems from radioactive systems. 	11.3.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities using low pressure nitrogen. Design features, such as welding techniques, surface finishes, etc., are included to minimize the need for decontamination and hence minimize waste generation [also meets Objectives 1 and 3]. The GWMS is designed with minimum embedded or buried piping. The drain gas header between buildings is equipped with piping sleeves with leakage directed back to the Compound Building for collection, thus preventing the spread of contamination [also meets Objectives 1 and 3]. 	11.3.2

Table 12.3-8 (38 of 82)

System: Gaseous Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The GWMS is designed for remote and automated operations. The system is equipped with instruments to actuate the drain valve from the header drain tank. Operational procedures and maintenance programs related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	11.3.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The GWMS is part of the overall plant and is included in the site Process Control Program and the Site Radiological Environmental Monitoring Program, for monitoring of facility and environmental contamination. The Site Radiological Environmental Monitoring Program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. Both Programs are included as COL Information items. 	11.3.2

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System: Liquid Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system components, including the collection tanks and the monitor tanks, are fabricated of stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment [also meets objectives 3 and 4]. The LWMS tanks are designed with sufficient capacity to provide temporary storage of the liquid waste generated from normal operation including anticipated operational occurrences. The tanks are equipped with cross-connected inlet headers for the control of overflow and to provide reasonable assurance of timely processing. This design minimizes the interruption of normal processing operation, the spread of contamination, and waste generation [also meets Objective 3]. The LWMS tanks are designed with ellipsoidal bottoms and mixing educators to minimize settling of suspended solids. The tanks have polished internal surfaces to minimize crud traps. The LWMS is designed with minimum embedded or buried piping. Piping between buildings is designed to be equipped with piping sleeves with leakage directed back to the Compound Building for collection, thus preventing the spread of contamination [also meets Objective 3]. 	11.2.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> All LWMS tanks are designed with level instruments to provide reasonable assurance of safe operation of the SSCs. The instruments provide alarms to the main control room and the radwaste control room for operator action in the event of high liquid levels. The cubicles in which the LWMS tanks are located are designed to include leak detection instrumentation to initiate alarms for operator actions in the event of leakage. The leak detection design has the capability to detect a small quantity of leakage for early detection. 	11.2.2

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System: Liquid Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The floor drains, equipment drains, chemical drains, and the laundry liquid wastes are collected in segregated tanks in separate cubicles. Since the LWMS is designed to operate in batches, treatment requirements for these collected wastes are determined through sampling and analyses. This design approach minimizes cross-contamination and waste generation. The process piping containing contaminated solids is properly sized to facilitate easier flow and with sufficient velocities to prevent the settling of solids. The piping is designed to reduce crud traps, thus reducing the decontamination requirements and waste generation. Decontamination fluid is collected and processed. Utility connections are designed with a minimum of two barriers to prevent contamination of non-radioactive systems from potentially radioactive systems. 	11.2.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed with decontamination capabilities. Design features such as spargers, welding techniques, surface finishes, etc., are included to minimize the need for decontamination and minimize waste generation [also meets Objective 3]. The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal. 	11.2.2

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System: Liquid Waste Management System

	Objective	SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> • The LWMS is designed for automated operation with manual initiation when sufficient liquid volumes are accumulated in each category of collection tanks to warrant processing. Hence, the LWMS is designed to operate in batch modes. Adequate instrumentation including level, flow rate, and pressure elements, and a process radiation monitor, is provided to monitor and control the operations to prevent undue interruption and the spread of radiologically contaminated material. • Piping is to be flushed clean after each processing to prevent solids from settling and accumulating. The flushing procedure also minimizes and/or prevents unintended leakage when the piping is not in use. • Leak detection instruments are provided to detect individual tank leakage. Adequate clearance around each tank is provided to enable prompt assessment and response when required. • A leak identification program is to be developed by the COL applicant to identify site-specific components that contain radioactive materials, buried piping, embedded piping, leak detection methods and capabilities, and the methods utilized for the prevention of unnecessary contamination of clean components, facility areas, and the environment. The leak identification program, to be a part of the Process Control Program, is designed to facilitate timely identification of leaks, prompt assessment, and appropriate responses to isolate and mitigate leakage [also meets Objectives 1 and 2]. • Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. 	11.2.2

Table 12.3-8 (42 of 82)

System: Liquid Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation (Cont'd)	<ul style="list-style-type: none"> Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	11.2.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The LWMS is part of the overall plant and is included in the site Process Control Program and the Site Radiological Environmental Monitoring Program, for monitoring of facility and environmental contamination. The Site Radiological Environmental Monitoring Program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. Both Programs are included as COL Information items. 	11.2.2

Table 12.3-8 (43 of 82)

System: Solid Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> • The system components, including the Low Activity Spent Resin Tank and the Spent Resin Long Term Storage Tank, are designed with stainless steel material and welded construction for life-cycle planning. The tanks are designed to have sufficient capacity to contain the spent resins for a decay period to reduce the radioactivity. • The Concentrate Treatment Subsystem and the Spent Resin Polymer Solidification Subsystem are designed as skid packages with self-containing drip pans to contain leakage. The drains connected to the drip pan are routed to a floor drain sump for collection and then pumped to the LWMS for treatment and release. • The High Activity Waste Storage Area and the Low Activity Waste Storage Area are designed with epoxy-coated floors and drain pipes to direct drainage to a floor drain sump for collection and subsequent pumping to the LWMS for treatment and release. • Cubicles, in which contaminated materials are handled and stored, have floors which are sloped and epoxy-coated to facilitate cleaning. Sump tanks are equipped with level switches to detect liquid accumulation and pumps are provided to transfer the fluid for proper treatment. • The SWMS is designed with above-ground piping to the extent practical. Buried and/or embedded piping is minimized. A covered concrete trench is provided for drainage piping as much as practicable. In the event that buried and/or embedded piping cannot be avoided, considerations to use double-wall piping and leak detection instruments are to be evaluated based on the risks and the radiological consequences associated with the contamination of the facility and the environment [also meets Objective 3]. 	11.4.2

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System: Solid Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The low activity spent resin tank and the spent resin long-term storage tank are designed with level instrumentation to provide reasonable assurance of the integrity of the SSCs including the associated piping and to provide alarms to warn the operators of leakages. The high activity wastes storage area and the low activity waste storage area are designed with remote cameras for waste handling operation as well as for periodic surveillance. Other components are designed for batch operation and are provided with adequate space to enable prompt assessment and response when required. 	11.4.2
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation. The process piping containing contaminated slurry is sized properly to facilitate easier flow and with sufficient velocity to prevent settling. The piping is designed to reduce fluid traps, thus reducing the decontamination needs and waste generation. Decontamination fluid is collected and routed to the LWMS for processing and release. Utility connections are designed with a minimum of two barriers to prevent contamination of non-radioactive systems from potentially radioactive systems. 	11.4.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities. Design features, such as welding techniques, surface finishes, etc., are included to minimize the need for decontamination and minimize waste generation [also meets Objectives 1 and 3]. 	11.4.2

Table 12.3-8 (45 of 82)

System: Solid Waste Management System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The spent resin removal and packaging is designed for remote and automated operations. The system is equipped with instruments to provide alarms for operator actions in the event leakage is detected. Operational procedures and maintenance programs related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. A leak identification program is to be developed by the COL applicant to identify site-specific components that contain radioactive materials, buried piping, embedded piping, leak detection methods and capabilities, and the methods utilized for the prevention of unnecessary contamination of clean components, facility areas, and the environment. The leak identification program, to be a part of the Process Control Program, is designed to facilitate timely identification of leaks, prompt assessment, and appropriate responses to isolate and mitigate leakage [also meets Objectives 1 and 2]. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	11.4.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The SWMS is part of the overall plant and is included in the site Process Control Program and the Site Radiological Environmental Monitoring Program, for monitoring of facility and environmental contamination. The Site Radiological Environmental Monitoring Program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. Both Programs are included as COL Information items. 	11.4.2

Table 12.3-8 (46 of 82)

System: Radioactive Drain System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system components, including all of the EFDS sumps that handle radioactive or potentially radioactive liquid, are designed with stainless steel liners encased in concrete cavities. The concrete cavities are epoxy-coated to provide smooth surfaces for cleaning and decontamination for life-cycle planning. The EFDS sump liners are designed to be sealed at the rim to prevent infiltration of contaminated drainage. The sump liners are also designed to be removable to facilitate cleaning, inspection, and maintenance activities on a predetermined frequency based on the service life of the epoxy coating. This design approach minimizes leakage and unintended contamination of the facility and the environment. The EFDS sumps are equipped with duplex pumps and level instruments to facilitate automated pump starting and stopping operation. Normally one pump in the sump is used. In the event of excessive collected drainage, the second pump in the sump is automatically started on the high-high level signal. This design approach minimizes sump overflow and thus minimizes the spread of contamination and waste generation [also meets Objective 3]. 	9.3.3
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> In areas that are not normally accessible during power operation, such as the ICI Cavity Sump, the RCB Drain Sump, the AB Floor Drain Sumps, etc., continual level indication is provided in the MCR and the RCR. In the event the sump liquid level reaches a preset limit, the level instrument initiates a signal to alarm in the MCR and the RCR for operator actions. This design approach, supplemented with operational procedures, provides adequate leak detection capability and minimizes the potential for the spread of contamination [also meets Objective 1]. 	9.3.3

Table 12.3-8 (47 of 82)

System: Radioactive Drain System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The sumps that handle contaminated or potentially contaminated fluids are specifically designed to minimize the spread of contamination through the use of liners, epoxy coating, and seals. This design approach is nuclear industry proven and is compatible with the chemical, physical, and radiological environment, thus minimizing waste generation The EFDS sumps are strategically located to collect floor and equipment drainage. Drains are segregated for different handling and processing requirements to minimize cross-contamination. The pumps are adequately sized for maximum input to prevent overflow; one pump is normally used, but when the sump liquid content level continues to rise, the second pump is initiated at the high-high level. This design approach minimizes the spread of contamination [also meets Objective 1]. The sumps are housed in individual cubicles. Cubicle curbs and/or doors are provided to reduce the spread of contamination to other areas and additional needs for decontamination. Sampling points are provided to facilitate sampling and analyses to properly route drainage for treatment and disposal. This design approach minimizes waste generation. Utility connections (e.g. demineralized water) are designed with a minimum of two barriers to prevent cross-contamination. 	9.3.3
4	Decommissioning Planning	<ul style="list-style-type: none"> The sump liners are designed for the full service life and are fabricated as individual assemblies for easy removal. The drain pipe flange connections to the sumps are also designed to permit disconnect to facilitate removal of the liners for inspection, cleaning, and maintenance of the epoxy coating of the concrete cavity. This design approach minimizes contamination and facilitates decommissioning [also meets Objective 3]. 	9.3.3

Table 12.3-8 (48 of 82)

System: Radioactive Drain System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning (Cont'd)	<ul style="list-style-type: none"> The EFDS is designed with minimum embedded or buried piping through the use of pump transfer. Drain pipes are routed through pipe chases as much as practicable. Piping between buildings is equipped with piping sleeves with leakage directed back to the originating facility for collection. This design approach thus minimizes the spread of contamination and facilitates decommissioning [also meets Objectives 1 and 3]. 	9.3.3
5	Operations and Documentation	<ul style="list-style-type: none"> The EFDS is designed for automated operation. The sumps are designed with dual level instruments to provide reasonable assurance of safe operation. A high level signal initiates liquid transfer with one pump. If the liquid level continues to rise, a high-high signal initiates additional liquid transfer with the secondary pump. Level signals are transmitted to the MCR for monitoring and operator response. Adequate space is provided in the vicinity of the sumps to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	9.3.3

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System: Radioactive Drain System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The EFDS is designed to handle radioactive fluids and the sumps are located at lower elevations to facilitate drainage collection. The EFDS is part of the overall plant and is included in the Site Radiological Environmental Monitoring Program, for monitoring of facility and environmental contamination. The Site Radiological Environmental Monitoring Program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. The Site Radiological Environmental Monitoring Program is included as a COL Information item. 	9.3.3

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System: Primary Sampling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The primary sampling system components for normal operation, including anticipated operational occurrences, are located in elevated cubicles inside the compound building. The primary sampling cooler rack and the associated sink are equipped with drain lines which are connected to the local drain piping. A hood is provided to remove radioactive gases from the collected samples during analysis to minimize the spread of contamination. In addition, the cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents unintended contamination of the facility and the environment. The post-accident sampling cooler rack and sink are located at the basemat level in the auxiliary building. The post-accident sampling rack and the sink are equipped with drain lines routed to the local drain piping. A hood is provided to remove radioactive gases from the collected samples during analysis to minimize the spread of contamination. In addition, the cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents unintended contamination of the facility and the environment. The primary sampling system is designed to be segregated with individual coolers and sampling points. Sampling piping sizes and components are designed to provide desired flow and sampling conditions in order to obtain representative samples. The normal primary sample sink, normal primary sample cooler rack, post-accident primary sample sink, and post-accident primary sample cooler rack are constructed of stainless steel material. The primary off-gas sample pump is fabricated from stainless steel material and is of welded construction for life-cycle planning, thus facilitating decontamination and minimizing the spread of contamination through leakage [also meets Objective 3]. 	9.3.2

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System: Primary Sampling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The primary sampling system is designed with automated operation with manual initiation for the two modes of operation. Adequate instrumentation is provided to control the sampling operations, thus minimizing waste generation. 	9.3.2
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation. The primary sampling system components are provided with demineralized water and nitrogen gas for decontamination and purging. The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems. 	9.3.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The components are located in accessible areas for operation, maintenance, and decommissioning purposes. The SSCs are designed with decontamination capabilities. Other design features, such as the utilized welding techniques, surface finishes, etc., are implemented in order to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The primary sampling system is designed without any embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the compound building or auxiliary building for collection, thus preventing unintended contamination of the environment [also meets Objectives 1 and 3]. 	9.3.2

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System: Primary Sampling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The primary sampling system is designed for automated and remote operations with manual initiation for the collection of samples. Adequate space is provided around the components to enable prompt assessment and response when required. The primary sampling system is a packaged vendor design. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	9.3.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The process and post-accident sampling system is designed to be used only for the purpose of collecting and analyzing samples. Because of the small volumes involved in the sampling operation and the procedures for proper handling and treatment of sampled materials, the process and post-accident sampling system is not required to be included in the Site Radiological Environmental Monitoring Program. 	9.3.2

Table 12.3-8 (53 of 82)

System: Turbine Generator System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The components of the turbine generator system are located in an open area at the 136' 6" elevation inside the turbine generator building. The HP turbine shell drain is discharged to the HP extraction line, which directs the drains to the condenser, and the LP turbine casing drains are discharged to the LP extraction line. This design approach prevents unintended contamination of the facility and the environment. The turbine generator system is designed with sufficient capacities and redundancy to support normal operation, including AOOs. The HP outer casing is fabricated from copper bearing carbon steel, the LP outer casing is fabricated from low carbon steel, and the other components and piping are fabricated from carbon steel and are of welded construction. This material selection and fabrication technique minimizes leakage and unintended contamination of the facility and the environment. 	10.2.3
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The turbine generator system is designed with automated operation for the different modes of operation through the TGCS. Adequate instrumentation, including pressure elements, is provided to monitor and control the system operation. 	10.2.3
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems. 	10.2.3

Table 12.3-8 (54 of 82)

System: Turbine Generator System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as integrated units for decommissioning planning. The SSCs are designed with decontamination capabilities. Design features, such as the utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The turbine generator system is designed without any embedded or buried piping, thus preventing unintended contamination to the environment [also meets Objectives 1 and 3]. 	10.2.3
5	Operations and Documentation	<ul style="list-style-type: none"> The turbine generator system is located in an open area inside the turbine generator building. Adequate space is provided around the components to enable prompt assessment and response when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.2.3
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The turbine generator system is expected to contain low levels of contamination due to steam generator leakage. From industry experience, the concentration is expected to be low. Any leakage from the system components is collected and directed to a contained location to prevent the spread of contamination within the plant. There is no embedded or buried piping for this system. Hence, inclusion in the Site Radiological Environmental Monitoring Program for environmental contamination is not specifically required for this system. 	10.2.3

Table 12.3-8 (55 of 82)

System: Main Steam System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The main steam lines are equipped with radiation monitors to detect radiological contamination and to provide alarms for operator notification. Following a steam generator tube rupture, the MSIVs for the affected steam generator are automatically closed via a MSIS in order to isolate the affected steam generator. This design minimizes radiological cross-contamination and unintended contamination of the facility [also meets Objective 3]. The main steam piping is sloped in the direction of steam flow to avoid water entrenchment and the collection of condensate drainage. To minimize the possibility of water impingement into the main turbine, drain traps are provided at low points in the main steam piping where water may collect. Condensate from these drain traps is continuously removed with direct piping to the main condenser during normal plant power operation. This design approach prevents the spread of contamination within the facility [also meets Objective 3]. 	10.3.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> Radiological monitoring of the MSS is provided continuously through N-16 gamma detection with a scintillation detector and microprocessor on each main steam line from the steam generator. In addition, radiation monitors are provided on the condenser vacuum pump exhaust line and the steam generator blowdown line to monitor contamination levels associated with the condensate and steam generator blowdown systems. These radiation monitors are designed to provide indication of steam generator leakage and alert the operator of a steam generator tube leak condition during power operation, including the identification of the affected steam generator. 	10.3.2

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Table 12.3-8 (56 of 82)

System: Main Steam System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> The safety-related portion of the main steam piping is designed to comply with the ASME Code Section III. The non-safety related portion of the main steam piping is designed to ASME B31.1 for safe operation and the minimization of waste generation. Main steam piping is designed to minimize the effects of erosion/corrosion, is adequately sized to limit velocities, and is routed with long radius elbows to minimize potential erosion and waste generation. The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. Process sampling connections are provided downstream of the MSIVs for monitoring of steam chemistry. 	10.3.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The main steam piping is designed for the full service life and is fabricated as individual segments for easy assembly and removal. The main steam piping is designed with clean out capabilities. Design features, such as the utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and resultant waste generation [also meets Objectives 1 and 3]. The MSS is designed without any embedded or buried piping, thus preventing contamination to the environment and facilitating decommissioning [also meets Objectives 1 and 3]. 	10.3.2

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Table 12.3-8 (57 of 82)

System: Main Steam System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The MSS piping and components are located in the Reactor Containment Building, Auxiliary Building, and Turbine Generator Building. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.3.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The MSS is part of the overall plant but does not have direct release points or contamination migration pathways to the environment during normal operation. Therefore, the MSS is not required to be included in the Site Radiological Environmental Monitoring Program. 	10.3.2

Table 12.3-8 (58 of 82)

System: Condensate System and Feedwater System (includes Feedwater Pump Turbine System and Heater Drain System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> • The condensate and feedwater system components are located in elevated cubicles inside the turbine generator building. The cubicle floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination through the facility and to the environment. • The condensate and feedwater system is designed with sufficient capacities to accommodate different modes of operation. The piping is adequately sized to prevent blockage and primary process piping is sloped to facilitate drainage and prevent fluid accumulation and crud buildup. • The condenser hotwell in each shell contains conductivity cells under each tube bundle to provide for the detection and location identification of condenser tube leaks. • The condenser is designed to the HEI Standard and manufactured using high grade austenitic stainless steel. • The low and high pressure feedwater heaters are constructed of carbon steel and are welded for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment [also meets Objective 3]. • The heat exchangers are designed such that tube side pressure is higher than shell side pressure to protect against leakage from the potentially contaminated steam to the clean circulating water system [also meets Objective 3]. • The feedwater pump turbine system is designed with casing leaking (exhausting) from the turbine directly to the condenser pit sump through the equipment and floor drains. 	10.4.7

Table 12.3-8 (59 of 82)

System: Condensate System and Feedwater System (includes Feedwater Pump Turbine System and Heater Drain System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination (Cont'd)	<ul style="list-style-type: none"> The heater drain system moisture separator drain tanks and the reheater drain tanks are located in an enclosed area at the top level inside the turbine generator building. The components are designed to cascade drains to the proper heaters. The high pressure heater drains are directed to the deaerator and the low pressure drains are directed to the condensers. Any leakage from the shell side or the tube side or the system piping, is dripped to the floor below. The floor is sloped and is epoxy coated to facilitate drainage to local sumps. This design approach minimizes the spread of contamination within the facility [also meets Objective 3]. Unintended contamination to the facility is minimized by the floor drainage design, the elevated location, and regular inspection. 	10.4.7
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The condensate and feedwater system is designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level, flow rate, temperature, and pressure elements, is provided to monitor and control the operations to prevent undue interruption. Leak detection trays are included at all tube-to-tube sheet interfaces. Provisions for early leak detection are provided for the tube sheet trays and in each hotwell section. The heater drain system drain tanks and heaters are equipped with level controls and in the drain lines which are controlled by air-operated control valves. The normal drain valve is an air-to-open, fail-closed valve. At abnormal levels, alarms are actuated in the MCR and RSR for operator action. If the emergency level is reached, the emergency drain valve is modulated which allows the drains to be directed to the condensers. This design approach thus provides early detection and minimizes the spread of contamination [also meets Objectives 1 and 3]. 	10.4.7

Table 12.3-8 (60 of 82)

System: Condensate System and Feedwater System (includes Feedwater Pump Turbine System and Heater Drain System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection (Cont'd)	<ul style="list-style-type: none"> The heater drain system is provided with sampling connections for determination of the water chemistry and contamination levels. 	10.4.7
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing the spread of contamination and waste generation. Normal and emergency drains are routed to the condenser and are forwarded to the condensate polishers for treatment, thus minimizing cross-contamination and waste generation. If leakage occurs from the condensate and feedwater system equipment, the water is drained to local drain hubs by gravity and transferred to the turbine generator building drain system sump for collection. For the heater drain system, the material used for piping and valves downstream of the drain control valves is chrome-moly steel. This material is used to minimize erosion and corrosion. The material for the first stage reheater scavenging lines and valves is also chrome-moly steel. 	10.4.7

Table 12.3-8 (61 of 82)

System: Condensate System and Feedwater System (includes Feedwater Pump Turbine System and Heater Drain System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities. Design features, such as the utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The condensate and feedwater system is designed without any embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the turbine generator building for collection, thus preventing unintended contamination [also meets Objectives 1 and 3]. 	10.4.7
5	Operations and Documentation	<ul style="list-style-type: none"> The condensate and feedwater system is located in an open area inside the turbine generator building. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.4.7
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The condensate and feedwater system is located inside the turbine generator building and drainage is collected and treated before release. Hence, the condensate and feedwater system is not required to be directly monitored for environmental contamination as part of the Site Radiological Environmental Monitoring Program. 	10.4.7

Table 12.3-8 (62 of 82)

System: Auxiliary Feedwater System and Auxiliary Feedwater Pump Turbine System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The Auxiliary Feedwater Pump Turbine System consists of two non-condensing steam turbines for the auxiliary feedwater pumps and their associated piping and valves. The Auxiliary Feedwater Pump Turbine System is not used during normal power operation, startup, or shutdown. The turbines are either manually or automatically actuated by an auxiliary feedwater actuation signal from the engineered safety system or the diverse protection system, which is initiated when the steam generator water level is low. The valve stem leak offs and drains are collected and directed to the Liquid Radwaste System for treatment and release. The Auxiliary Feedwater Pump Turbine System piping uses stainless steel material and is fabricated to ASME Section III, Class 3 requirements. 	10.4.9
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The valve leakoffs and drains are collected by the LWMS which includes level instrumentation to indicate abnormally high input from leakage. 	10.4.9
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> Normally, the Auxiliary Feedwater System is not in use and is not radioactively contaminated. However, there is a segment of pipe, connected to the high pressure feedwater lines after the containment penetration to the steam generators, which may become contaminated by the feedwater (recycle condensate) up to the containment isolation check valves. The feedwater piping is fabricated of stainless steel material, is of welded construction, and is designed to safety class 3 and seismic category I requirements. This design approach minimizes leakage and unintended contamination of the facility and the environment [also meets Objective 1]. 	10.4.9
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. 	10.4.9

Table 12.3-8 (63 of 82)

System: Auxiliary Feedwater System and Auxiliary Feedwater Pump Turbine System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.4.9
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> Because of its normal status in standby mode and low possibility for contamination, the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Therefore, the auxiliary feedwater system inclusion in the Site Radiological Environmental Monitoring Program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection of radiological contamination. 	10.4.9

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System: Condensate Polishing System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system components, including the polishing cation and mixed bed demineralizer columns, are designed to be constructed of carbon steel material with rubber linings and welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. The condensate polishing tanks are designed with ellipsoidal bottoms. The tanks have polished internal surfaces to minimize crud traps. 	10.4.6
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The condensate polishing tanks are designed to include pressure and level instrumentation to provide reasonable assurance of safe operation of the SSCs including the associated piping and provide alarms for operator notification in the event of high content levels. The area in which the condensate polishing tanks are located is designed with trenches to drain leakage to the nearest sump for collection and containment. The sump is equipped with a level switch which provides an alarm signal to the MCR in the event of high liquid level detection. This design prevents the spread of contamination to the facility and the environment and allows for prompt leak detection [also meets Objective 1]. 	10.4.6
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The condensate is collected in the provided sump with leak collection trenches. The trenches are sloped and epoxy coated to enhance drainage and facilitate cleaning. This design approach minimizes the spread of contamination and waste generation. 	10.4.6

Table 12.3-8 (65 of 82)

System: Condensate Polishing System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> The condensate polishing system is designed to operate in total, partial, and bypass flow modes to meet secondary side chemistry requirements. The system is designed with on-line specific conductivity instruments, pH indicators, and other process instrumentation for automated operation with manual initiation. Sampling and analyses are implemented to determine optimum operational requirements. The process piping containing contaminated solids is properly sized to facilitate easier flow with sufficient velocities to prevent the settling of solids. The piping is designed to reduce fluid traps, thus reducing the need for decontamination and the resultant waste generation. Utility connections (sluice water and service air) are designed with a minimum of two barriers to prevent the contamination of clean systems. 	10.4.6
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The condensate polishing system is designed with minimum embedded or buried piping. Piping between buildings is equipped with piping sleeves with leakage directed back to the compound building for collection, thus preventing unintended contamination [also meets Objectives 1 and 3]. 	10.4.6

Table 12.3-8 (66 of 82)

System: Condensate Polishing System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> The condensate polishing operation is designed for automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including conductivity, level, flow rate, and pressure elements, is provided to monitor the process to prevent undue interruption. Adequate space is provided in the vicinity of the sumps to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.4.6
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The condensate polishing system is expected to have low levels of contamination as a result of leakage in the steam generator. Due to the containment of leakage from the system within the Turbine Generator Building sumps, inclusion in the Site Radiological Environmental Monitoring Program is not required for this system. 	10.4.6

Table 12.3-8 (67 of 82)

System: Condenser Vacuum System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The condenser vacuum system components are located in an open area at the foundation level inside the turbine generator building. The components are designed to be skid mounted. The floors are sloped, coated with epoxy, and provided with drains that are routed to the local drain hubs. This design approach prevents the spread of contamination to the facility and the environment. The condenser vacuum system is designed with sufficient capacity and redundancy to support normal operation, including anticipated operational occurrences. The components and piping are fabricated from carbon steel with welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. 	10.4.2
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The condenser vacuum system is designed with automated operation with manual initiation for the different modes of operation. Adequate instrumentation, including level and pressure elements and a radiation monitor, is provided to monitor the operations. Upon receipt of a high radiation signal, the vacuum discharge is diverted to the reactor containment building drain sump, and the vent from that sump is processed through the containment ventilation purge system. This design approach minimizes the spread of contamination and provides early detection [also meets Objectives 1 and 3]. 	10.4.2
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. The condenser vacuum system components are provided with seal water (demineralized water) for decontamination. Chilled water and other utilities are provided to facilitate operations. The utility connections are designed with a minimum of two barriers to prevent the contamination of clean systems. 	10.4.2

Table 12.3-8 (68 of 82)

System: Condenser Vacuum System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The SSCs are designed to facilitate decontamination. Design features, such as the utilized welding techniques, surface finishes, etc., are included to minimize the need for decontamination and therefore minimize waste generation [also meets Objectives 1 and 3]. The condenser vacuum system is designed without any embedded or buried piping, thus preventing unintended contamination to the environment and facilitating eventual decommissioning [also meets Objectives 1 and 3]. 	10.4.2
5	Operations and Documentation	<ul style="list-style-type: none"> The condenser vacuum system is located in an open area inside the turbine generator building. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.4.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The condenser vacuum system is part of the overall plant and is included in the Site Radiological Environmental Monitoring Program, for monitoring of the release of non-condensable gases and the potential for environmental contamination. The Program includes air sampling and analysis and monitoring of meteorological conditions, hydro-geological parameters, and potential migration pathways of the radioactive contaminants. The Site Environmental Monitoring Program is included as a COL Information item. 	10.4.2

Table 12.3-8 (69 of 82)

System: Circulating Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system components, including the piping and heat exchangers, are designed with corrosion and erosion resistant materials for life-cycle planning. The piping and tubes are designed with hydraulic features to maintain heat transfer performance and tube integrity, thus minimizes fouling that may result in corrosion and the spread of contaminated material. The Circulating Water System is sampled and analyzed periodically to assess the water quality including the level of radiological contamination. In the event that contamination is detected, inspection and repair of the heat exchanger tubes are performed as corrective actions. This design approach minimizes leakage and unintended contamination of the facility and the environment [also meets Objective 3]. 	10.4.5
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The Circulating Water System is designed in conjunction with the Turbine Generator Building Drain System. The sumps in the Turbine Generator Building Drain System are used to collect above ground piping leakages and are equipped with dual level instruments to provide reasonable assurance of the detection of liquid level, thus minimizing the spread of contamination and waste generation [also meets Objectives 1 and 3]. Circulating Water System leakage is detected by sampling and analysis of the circulating water in the piping and through measured fluid levels in the Turbine Generator Building Drain System sumps followed by sample analysis and piping inspection. This approach is consistent with the operating requirements of NRC RG 4.21 and nuclear industry practices. 	10.4.5
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. 	10.4.5

Table 12.3-8 (70 of 82)

System: Circulating Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, Decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> The Circulating Water System is designed to operate at a higher pressure than the condenser. This design approach minimizes infiltration of condensate into the circulating water which reduces the potential for contamination. The areas in which the Circulating Water System SSCs are housed are designed with sloped floors, epoxy coating to provide drainage and cleanable surfaces, and local sumps to collect leakage and overflows. Cubicle curbs are also provided to reduce the spread of contamination to other plant areas. 	10.4.5
4	Decommissioning Planning	<ul style="list-style-type: none"> The Circulating Water System may contain some buried piping in the yard related to the site-specific cooling tower design. Circulating water is sampled and analyzed periodically to assess water quality and contamination levels. When contamination is detected, the piping segment will be isolated for inspection and repair. This design approach minimizes unintended contamination of the facility and the environment and facilitates decommissioning [also meets Objective 1]. 	10.4.5
5	Operations and Documentation	<ul style="list-style-type: none"> The Circulating Water System is designed for automated operations with manual initiation for the different modes of operation. Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	10.4.5

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System: Circulating Water System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The Circulating Water System is designed to be a generally nonradioactive system with the potential to have low levels of contamination through leakage in the condenser tubes. Through monitoring, inspection, and lessons learned from industry experiences the integrity of the circulating water is well maintained, resulting in minimal levels of contamination of the facility. Continuous cooling tower blowdown is used to provide reasonable assurance that the water quality is maintained. Hence, inclusion in the Site Radiological Environmental Monitoring Program is not required for this system. 	10.4.5

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System: HVAC Systems for Potentially-Contaminated Building Areas (Fueling Handling Area HVAC System, Auxiliary Building Controlled Area HVAC System, Reactor Containment Building HVAC System, Reactor Containment Building Purge System, Compound Building HVAC System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The system components, including the air cleaning units (ACUs), air handling units (AHUs), cubicle coolers, and recirculating fans, are fabricated from materials, such as carbon and stainless steels that are compatible with the chemical, physical, and radiological environment and are of welded construction for life-cycle planning. The design thus minimizes unintended leakage and unintended contamination of the facility and the environment. For the reactor containment building HVAC system, the chilled water side of the AHUs and fan coolers is operated at a higher pressure than the containment environment to prevent the contamination of the chilled water. Corrosion inhibitor chemicals are also added to the chilled water system to minimize the potential of leakage due to corrosion. The mounting frames of the HEPA filter and carbon adsorber are stainless steel and of welded construction for life-cycle planning and to prevent unfiltered inleakage [also meets Objective 3]. Doors and door frames are the marine bulkhead type or an equivalent airtight construction, thus minimizing leakage and unintended contamination of the facility and the environment. All ACU housing penetrations are sealed by welds or adjustable compression-gland type seals to prevent unfiltered inleakage [also meets Objective 3]. All drain lines from ACUs, AHUs, and fan coolers are sloped and routed to the protective equipment drain. 	9.4.2/9.4.5 9.4.6/9.4.7

Table 12.3-8 (73 of 82)

System: HVAC Systems for Potentially-Contaminated Building Areas (Fueling Handling Area HVAC System, Auxiliary Building Controlled Area HVAC System, Reactor Containment Building HVAC System, Reactor Containment Building Purge System, Compound Building HVAC System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The ACU housing leak test is performed in the manufacturer's shop prior to shipment and in the field following the recommended test frequency described in ASME N510 to prevent unfiltered leakage [also meets Objectives 1 and 3]. The mounting frame leak tests of HEPA filters and carbon adsorbers are performed in the manufacturer's shop prior to shipment and performed in accordance with ASME N510 recommended test frequencies at the site to prevent unfiltered leakage [also meets Objectives 1 and 3]. Radiation monitors are provided at the outlet of ACUs to monitor the radiological contamination levels to provide reasonable assurance that the effluent release limit will not be exceeded. 	9.4.2/9.4.5 9.4.6/9.4.7
3	Reduction of Cross Contamination, decontamination, and Waste Generation	<ul style="list-style-type: none"> The ACU is designed to operate slightly below atmospheric pressure through the use of an induced fan. This design approach minimizes air leakage from the ACU to its surrounding. All ducts are of welded construction to minimize cross-contamination. HVAC component housing surfaces are painted smooth for corrosion protection, ease of decontamination, and cleaning. HEPA filters capture radioactive particles and carbon adsorbers remove organic and inorganic forms of iodine from the air stream, thus minimizing the spread of contamination and the resultant waste generation. 	9.4.2/9.4.5 9.4.6/9.4.7

Table 12.3-8 (74 of 82)

System: HVAC Systems for Potentially-Contaminated Building Areas (Fueling Handling Area HVAC System, Auxiliary Building Controlled Area HVAC System, Reactor Containment Building HVAC System, Reactor Containment Building Purge System, Compound Building HVAC System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
3	Reduction of Cross Contamination, decontamination, and Waste Generation (Cont'd)	<ul style="list-style-type: none"> The low volume purge exhaust ACUs are equipped with medium and high efficiency particulate filters to capture radioactive particles and carbon adsorbers to remove organic and inorganic forms of iodine from the air stream. This design minimizes cross contamination within the building. Because of the large air flow, the high volume purge exhaust ACUs are designed to have medium and high efficiency particulate filters to remove particulates from the air stream. Adequate space is provided around the HVAC equipment to enable prompt assessment and responses when required. 	9.4.2/9.4.5 9.4.6/9.4.7
4	Decommissioning Planning	<ul style="list-style-type: none"> The ACUs, AHUs, fan coolers, and recirculating fans are designed and fabricated as modular units and compartments for easy decommissioning. 	9.4.2/9.4.5 9.4.6/9.4.7
5	Operations and Documentation	<ul style="list-style-type: none"> Adequate work space between mounting frames for the ACUs is provided to facilitate maintenance and minimize the potential for the spread of contamination. Carbon removal from the carbon adsorber is facilitated by the use of portable pneumatic carbon removal equipment to minimize spillage. ACUs, AHUs, fan coolers, and recirculating fans are designed with adequate instrumentation to be remotely operated with manual initiation and stop from the main control room and the remote shutdown room. The ACUs and AHUs are packaged units. Instruction manuals including assembly/disassembly procedures and maintenance/surveillance programs are provided from the equipment vendor. 	9.4.2/9.4.5 9.4.6/9.4.7

Table 12.3-8 (75 of 82)

System: HVAC Systems for Potentially-Contaminated Building Areas (Fueling Handling Area HVAC System, Auxiliary Building Controlled Area HVAC System, Reactor Containment Building HVAC System, Reactor Containment Building Purge System, Compound Building HVAC System)

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The air quality around the plant is routinely sampled and analyzed for contamination levels and migration pathways as part of the Site Radiological Environmental Monitoring Program. The potentially contaminated building HVAC systems, as part of the overall plant, are expected to be included in this program. 	9.4.2/9.4.5 9.4.6/9.4.7

Table 12.3-8 (76 of 82)

System: Miscellaneous Building Drain System and Turbine Generator Building Drain System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The turbine generator building drain system and the miscellaneous building drain system are part of the non-radioactive drain system. These systems consist of sumps and the associated discharge piping for pumping non-radioactive floor and equipment drainage from the turbine generator building sumps and the condensate overflow of the condensate system to the wastewater treatment system. These sumps include the condenser pit sumps, condensate polishing area sump, and the condensate overflow storage sump. The inclusion of these drains and sumps prevents overflow from spreading to other plant areas and to the environment. 	9.3.3
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The discharge from the condensate polishing area sump is monitored for radiation. The system controls are designed to stop the condensate polishing sump pump when radioactivity is detected. Also, the discharge from the condenser pit sump pumps is monitored for radiological contamination due to primary-to-secondary side leakage. The system controls are designed to stop the operating sump pumps when there is radiation detected in the discharge line. The drains are then redirected to the LWMS for treatment and release. 	9.3.3
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The drainage piping for these building drain systems is constructed of carbon steel material and is fabricated to ASME B31.1 requirements. The majority of the drainage in the turbine generator building drain system is non-radioactive and is directed to the non-radioactive equipment vent and drain system from which it is sent to the wastewater treatment system for treatment and release [also meets Objective 1]. 	9.3.3

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Table 12.3-8 (77 of 82)

System: Miscellaneous Building Drain System and Turbine Generator Building Drain System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
4	Decommissioning Planning	<ul style="list-style-type: none"> The sump liners are designed for the full service life and are fabricated as individual assemblies for easy removal. The drain pipe connections to the sumps are also designed to permit disconnect to facilitate removal of the liners for inspection, cleaning, and maintenance of the epoxy coating of the concrete cavity. This design approach minimizes unintended contamination and facilitates more easily decommissioning [also meets Objective 1]. 	9.3.3
5	Operations and Documentation	<ul style="list-style-type: none"> Adequate space is provided around the equipment to enable prompt assessment and responses when required. Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	9.3.3
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The non-radioactive drain system is designed to be a generally nonradioactive system with the potential to have low levels of contamination. Hence, inclusion in the Site Radiological Environmental Monitoring Program is not required for this system. 	9.3.3

Table 12.3-8 (78 of 82)

System: Radiation Monitoring System and Process Sampling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The process sampling piping is designed to return unused portions of the sampling fluid to the process lines. The piping is fabricated from stainless steel material and is of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. The process sampling system components for normal operation, including anticipated operational occurrences are located inside the auxiliary building and compound building. The secondary cooler rack, grab sample sink, continuous sample sink, chiller, analyzer rack, condenser hotwell sample sink and rack, secondary local grab sample cooler racks, feedwater corrosion sample panel, liquid radwaste sample sink, and gaseous radwaste sample panel are equipped with drain lines piped to the local drain piping. A hood is provided to remove radioactive gases from samples during analysis in order to minimize the spread of contamination. Sampling piping sizes and components are designed to provide the desired flow in order to obtain representative samples. All sampling piping is sloped to facilitate drainage and to prevent fluid accumulation and crud buildup, thus reducing the level of contamination. The sinks, racks, and panels listed above are constructed of stainless steel material. The sample pumps are fabricated from stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment. The RMS piping and components are located in the reactor containment building, auxiliary building, and compound building. Adequate space is provided around the components to enable prompt assessment and response when required. 	11.5.2

Table 12.3-8 (79 of 82)

System: Radiation Monitoring System and Process Sampling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The process and effluent radiation monitoring and sampling system is designed for automated operation. Adequate instrumentation is provided to enable effective control of sampling operations. 	11.5.2
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation. Demineralized water is provided for decontamination of the system piping. Sample sinks are provided to collect all spillage. The sink is designed to include a drain line directly connected to the local floor drain subsystem, from which the drains are sent to the LWMS for processing and release. This design reduces the potential for the spread of contamination in the plant [also meets Objective 1]. 	11.5.2
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. The SSCs are designed with decontamination capabilities. Design features, such as utilized welding techniques, surface finishes, etc., are implemented in order to minimize the need for decontamination and the resultant waste generation [also meets Objectives 1 and 3]. The process and effluent radiation monitoring and sampling system is designed without any embedded or buried piping, thus preventing unintended contamination to the environment [also meets Objectives 1 and 3]. The system components are provided with polished surfaces for ease of decontamination. 	11.5.2

Table 12.3-8 (80 of 82)

System: Radiation Monitoring System and Process Sampling System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
5	Operations and Documentation	<ul style="list-style-type: none"> Operational procedures and maintenance programs as related to leak detection and contamination control will be prepared by the COL applicant. Procedures and maintenance programs are to be completed before fuel is loaded for commissioning. Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	11.5.2
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The process and effluent radiation monitoring and sampling system is part of the overall plant and the data collected through the use of this system can be used in the Site Radiological Environmental Monitoring Program for monitoring the level of potential environmental contamination. The Site Radiological Environmental Monitoring Program is included as a COL Information item. 	11.5.2

Table 12.3-8 (81 of 82)

System: Containment Monitoring System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
1	Prevention/Minimization of Unintended Contamination	<ul style="list-style-type: none"> The hydrogen analyzers, which are part of the hydrogen monitoring system, are a subsystem of the containment monitoring system. During normal operation, the hydrogen analyzers are on standby mode and are at a low pressure condition. The hydrogen analyzers and the associated piping and valves, therefore have very low potential to contaminate other areas of the facility or the environment during normal operation. This design satisfies the requirements of NRC RG 4.21. 	6.2.4
2	Adequate and Early Leak Detection	<ul style="list-style-type: none"> The containment monitoring system is designed to provide continuous indication of containment conditions including pressure, water level, and hydrogen concentration in the containment atmosphere and in the in-containment refueling water storage tank. 	6.2.4
3	Reduction of Cross Contamination, Decontamination, and Waste Generation	<ul style="list-style-type: none"> In addition to this instrumentation, the containment monitoring system consists of piping and valves for the hydrogen analyzers that are used to extract air samples for analysis of hydrogen concentrations in the containment atmosphere and the in-containment refueling water storage tank. After analysis, the air samples are sent back inside containment thereby minimizing the spread of contamination to other plant areas [also meets Objective 1]. 	6.2.4
4	Decommissioning Planning	<ul style="list-style-type: none"> The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal. 	6.2.4
5	Operations and Documentation	<ul style="list-style-type: none"> Complete documentation of system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL Information item. 	6.2.4

Table 12.3-8 (82 of 82)

System: Containment Monitoring System

Objective		SSC Control Measures/Design Features in DCD to Meet Objective	DCD Tier 2 Reference
6	Site Radiological Environmental Monitoring	<ul style="list-style-type: none"> The containment monitoring system is located within the reactor containment building. Because of its location and normal status in standby mode, the potential for environmental contamination of soil and groundwater from liquid leakage is minimal. Therefore, the system inclusion in the Site Radiological Environmental Monitoring Program is not required. However, a site radiological environmental monitoring program is included for the whole plant for detection of radiological contamination. 	6.2.4

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-11 Radiation Zones (Normal) Compound Building Partial Plan

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-17 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 55'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-18 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 55'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-19 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 55'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-21 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 78'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-22 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 78'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-23 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 78'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-25 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 100'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-26 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 100'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-27 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 100'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-29 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 120'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-30 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 120'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-31 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 120'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

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

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-33 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 137'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-34 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 137'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-35 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 137'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-36 Radiation Zones (1 Month after Accident)-Auxiliary/Reactor Containment Building El. 137'-6"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-37 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 156'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-38 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 156'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-39 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 156'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-40 Radiation Zones (1 Month after Accident)-Auxiliary/Reactor Containment Building El. 156'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-41 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 174'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-42 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 174'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-43 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 174'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-44 Radiation Zones (1 Month after Accident)-Auxiliary/Reactor Containment Building El. 174'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-45 Radiation Zones (1 Hour after Accident)-Auxiliary/Reactor Containment Building El. 195'-0" & Roof Plan

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-46 Radiation Zones (1 Day after Accident)-Auxiliary/Reactor Containment Building El. 195'-0" & Roof Plan

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-47 Radiation Zones (1 Week after Accident)-Auxiliary/Reactor Containment Building El. 195'-0" & Roof Plan

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-48 Radiation Zones (1 Month after Accident)-Auxiliary/Reactor Containment Building El. 195'-0" & Roof Plan

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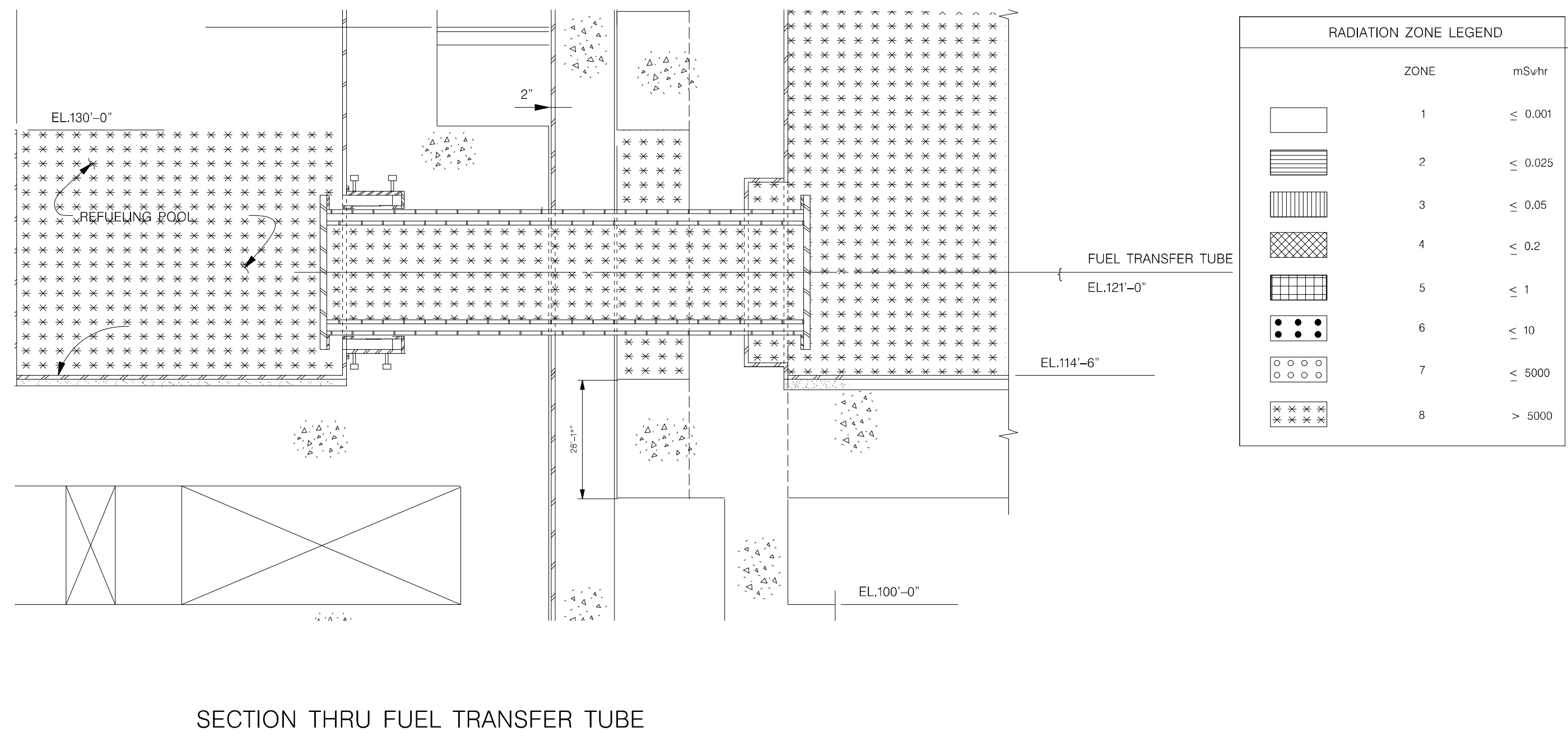


Figure 12.3-49 Sectional View of Fuel Transfer Tube

APR1400 DCD TIER 2

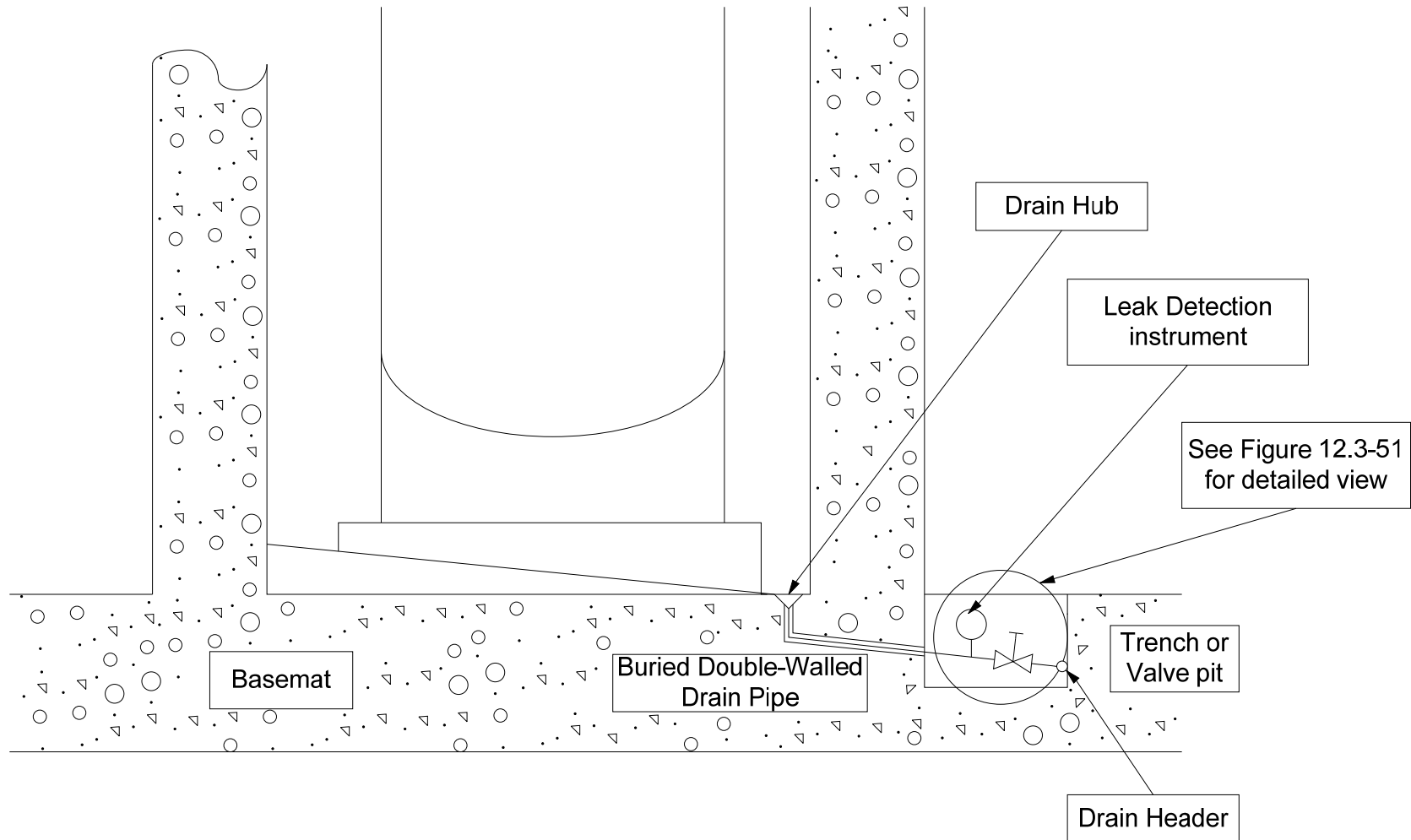


Figure 12.3-50 Typical Design Details of Drainage and Leak Detection Configuration

APR1400 DCD TIER 2

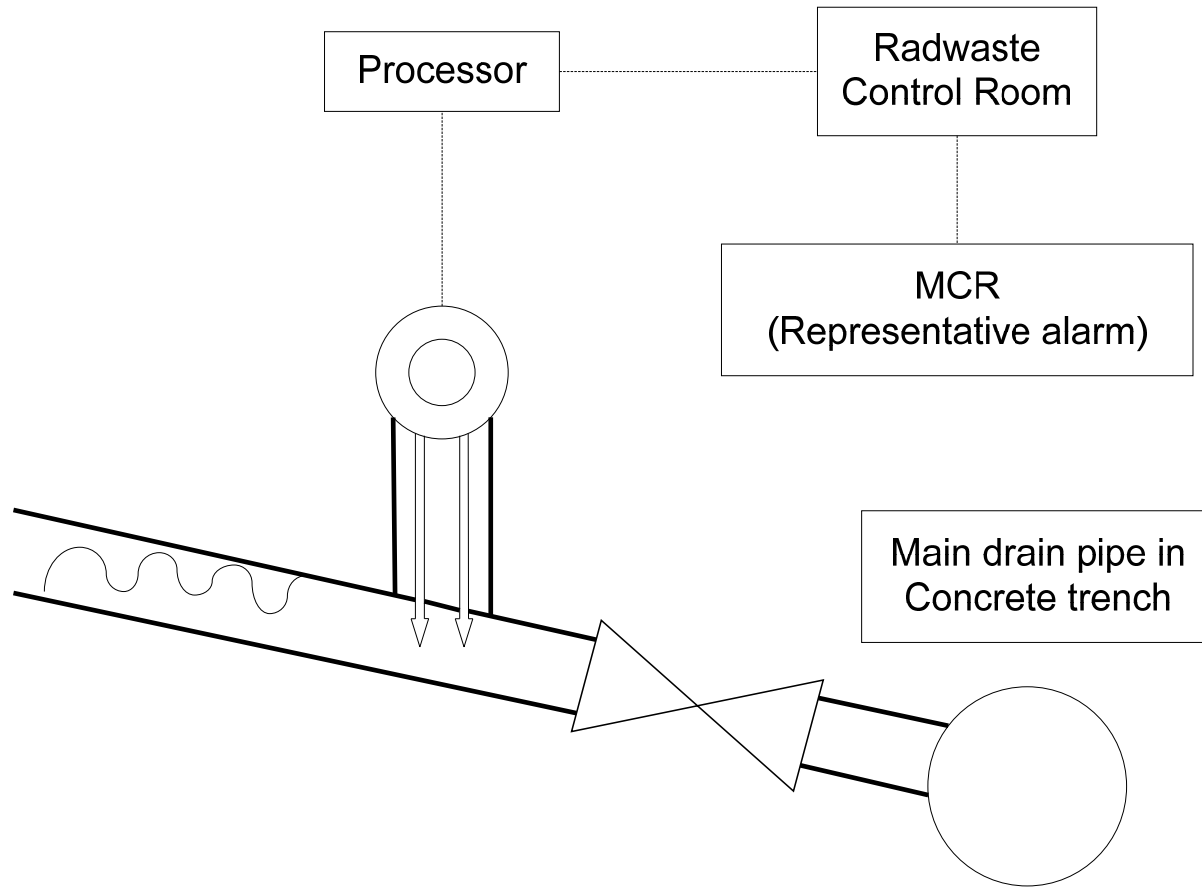


Figure 12.3-51 Typical Design Details of Leak Detection Instrumentation

APR1400 DCD TIER 2

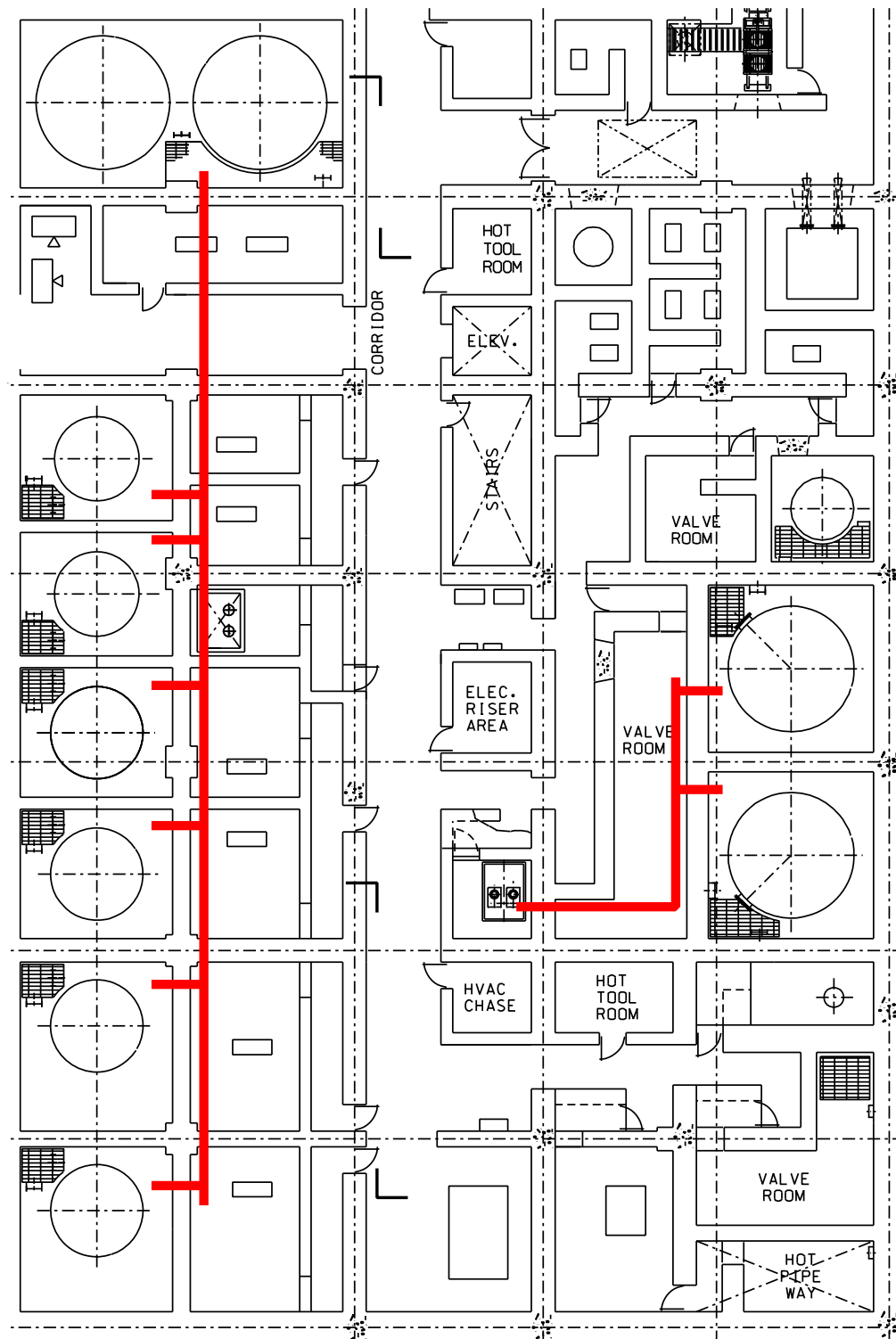


Figure 12.3-52 Typical Compound Building Trench Location Schematic

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- Install piping sleeve with flexible joint between building walls
- Slope sleeve slightly to facilitate drainage
- Provide caulking on sleeve to seal and flexibility to sleeve expansion
- Provide sealing material between sleeve and inside pipe to maintain pressure boundary and piping movement
- Provide drainage

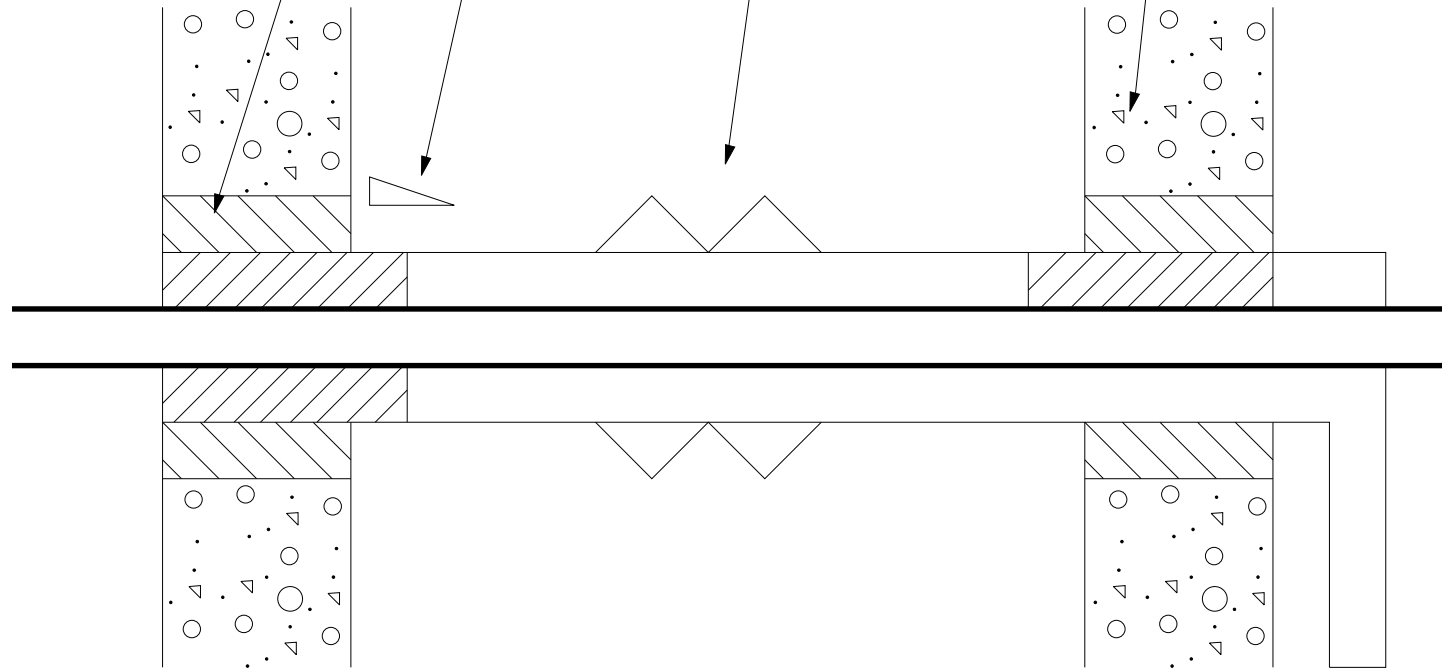


Figure 12.3-53 Typical Design for Building Sleeves Schematic

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12.4 Dose Assessment

12.4.1 Estimated Annual Occupational Exposures

12.4.1.1 ALARA Design Features for Occupational Dose Reduction

A review of the operating data indicates that a significant portion of the occupational exposures (from 50 to 75 percent) are related to the exposure to activated corrosion products. Minimization of primary system corrosion and resultant dose rates is the most effective approach to reduce total personnel occupational exposures.

Leakage of fuel rod cladding accounts for the remaining 25 to 50 percent of pressurized water reactor (PWR) occupational exposure. The fuel rod performance of the APR1400 design is expected to reduce fuel leakage to less than 0.1 percent. This feature also is expected to reduce effluent releases and radwaste activity.

It is expected that proper material selection, chemistry control, and improved fuel cladding leakage reduces the annual occupational exposure.

The design features described in Subsection 12.3.1.5 for SGs greatly reduce the time spent in performing maintenance and inspection. This task represents approximately 25 percent of the total dose for the average PWRs.

The dose received during an outage for SG maintenance is dependent on the number of tubes requiring inspection and repair. The APR1400 SG design provides thermal treatment of the SG tubes and the manufacturing technique found to have avoided the known SG tube cracking. SG tubes are fabricated of Alloy 690 instead of Alloy 600. Alloy 690 has been proven to be less susceptible to SG tube cracking.

The Palo Verde SG tubes are fabricated of Alloy 600 and processed with thermal treatment. The Palo Verde plants are known to have no tube cracking problems. This has resulted in a significant reduction in the annual dose received during SG maintenance. Palo Verde's average occupational exposure due to SG maintenance during a refueling outage has been 0.40 person-Sv, in contrast to the Duke Power Company's average of 0.62 person-Sv. Therefore, a factor of 1.6 reduction in annual occupational exposure is expected for SG

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maintenance based on improved manufacturing techniques and material selection. The advanced PWRs like the APR1400 have several additional features that are not provided at Palo Verde or the Duke Power Company's units as described in Subsection 12.3.1.5. These features serve to reduce the time spent maintaining the SG and are therefore expected to reduce the annual occupational exposure due to steam generator maintenance by a factor of 1.5. In addition, a factor of 2.5 reduction is expected from material selection, chemistry control, and improved fuel performance as described above. Therefore, all of the above advanced PWR design features result in an overall reduction of the annual occupational exposure due to steam generator maintenance by a factor of 6.

The occupational exposure due to RCP maintenance and inspection represents approximately 4 percent of total station exposures. The APR1400 design minimizes the need for seal replacement and reduces the time for performing this task.

Many other features to reduce occupational exposures are applied to the APR1400 design. Some examples are:

- a. Extended fuel cycle
- b. Identification of RCS leakage
- c. Integrated head assembly
- d. Single stud tensioner
- e. Permanent refueling pool seal
- f. RCS equipment and piping materials

These materials in direct contact with the RCS have a low cobalt content.

- g. Steam generator tube material

This material has an average cobalt content of less than 0.015 percent by weight.

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h. Equipment reliability, maintainability, and accessibility

The APR1400 is designed so that adequate spacing is provided around components. This enhances accessibility, maintainability, and inspectability. In addition, reliable equipment is used so that the frequency of maintenance, and thus, dose, is reduced.

i. Components, such as tank, piping, and instruments, are designed to minimize particulate deposition in accordance with NRC RG 8.8 (Reference 1) recommendations. For example, tanks with sloped bottoms are used in the APR1400 design.

j. System flushing and decontamination capability

The APR1400 design provides flushing and decontamination capability to lines where particulate deposition is possible, such as resin transfer lines. This enables the operators to flush the lines to prevent buildup of radioactive particulates in the lines, which could result in high personnel exposure.

k. Radwaste handling operations

Radwaste handling operations are described in detail in Sections 11.2, 11.3, and 11.4. These systems are designed to operate remotely to the extent practicable, and waste streams are segregated to facilitate processing of waste effectively and efficiently and minimize radiation exposure.

l. Isolation of contaminated components and proper shielding

The APR1400 design segregates the radioactive systems from the non-radioactive systems. In addition, components in the radioactive systems are separately located in cubicles based on the source strength, the nature of operation, and the frequency of access. The design also provides adequate shielding between components, valves, and pump galleries to enable personnel to perform operational activities and maintenance in areas of lower radiation.

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m. Design for recovery from abnormal occurrences

1) Divisional separation

The APR1400 design provides separation of SSCs, which meets the single-failure criterion. Fire and flood barriers are provided to prevent the spread of such events and the loss of equipment in adjacent areas. This design approach also facilitates speedier recovery from fire or flood events.

2) RCS depressurization

Use of the IRWST enables the operators to depressurize the RCS by discharging the coolant to the IRWST through the POSRVs. This tank allows the dissipation of a significant portion of energy to the IRWST, thus facilitating the recovery from loss of heat sink events.

3) Safety injection

The provision of a direct header from safety injection, which takes suction from the IRWST to the RV, reduces the likelihood of a failure to provide safety injection water to the reactor.

4) Safety-related equipment power supply

The APR1400 design provides Class 1E and redundant electric power to safety-related equipment, thus reducing the probability of safety-related equipment failure due to a loss of power.

These features are further expected to reduce occupational radiation exposures to ALARA levels.

12.4.1.2 Occupational Dose Assessment

The expected annual cumulative occupational radiation exposure (ORE) from the APR1400 operation is estimated according to the guidance in NRC RG 8.19 (Reference 2). The

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APR1400 ORE estimation is based on operational experience data from the OPR1000 plant, which is the Korean PWR reactor rated at 1,000 MWe. The design features of this Combustion Engineering type OPR1000 are similar to those of the APR1400.

The dose rates for the APR1400 are conservatively assumed to increase in proportion to the thermal power increase from the OPR1000. During a refueling outage, the ORE is strongly dependent on the activated corrosion products. Although the deposition of the activated corrosion products in the APR1400 is expected to be less because of design improvements, the reduction is not taken into account in the ORE estimation.

According to NRC RG 8.19, the ORE assessment should be based on anticipated radiation conditions after at least 5 years of plant operation. The measured ORE data for a rolling 3-year period from 2009 to 2011 for the OPR1000 (Ulchin Unit 3), which started commercial operation in August 1998, is used in the ORE assessment. The internal and external doses due to airborne radionuclides are negligible because the measurement data for the OPR1000 and NUREG-0713 (Reference 3) demonstrate that the ORE contribution from airborne radioactivity is minimal.

The estimated annual occupational radiation exposures for the APR1400 are determined within the following work categories:

- a. Reactor operations and surveillance
- b. Routine maintenance
- c. Inservice inspection
- d. Special maintenance
- e. Waste processing
- f. Refueling

The estimated result of occupational exposure for the APR1400 is presented in Table 12.4-7.

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12.4.1.2.1 Reactor Operations and Surveillance

The performance of various systems and components is monitored by operators. The inspection includes routine inspections and performance tests such as system leakage monitoring, control of instrumentation, electrical and mechanical equipments, and preventive maintenance. Health physics surveys are also included. In some cases, operation of some manual valves requires personnel to enter radiologically controlled areas (RCA) for short periods.

Examples of routine operation and surveillance activities are as follows:

- a. Routine inspections of the plant components and systems
- b. Unidentified leak checks
- c. Operation of the manual valves
- d. Reading of the instruments
- e. Routine health physics patrols and surveys
- f. Decontamination of the equipment or plant work areas
- g. Calibration of the electrical and mechanical equipment
- h. Chemistry sampling and analysis

These activities may be conducted in the reactor containment, auxiliary building, compound building and turbine building. The frequency and duration of activities, which are the major factors for ORE dose, are dependent on the operational programs and procedures.

Improvement of the plant layout, access provisions, and operational procedures can reduce exposure time in RCAs for surveillance, inspection, and testing works, thereby minimizing occupational radiation exposure.

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Table 12.4-1 provides a breakdown of the collective doses for reactor operations and surveillance activities.

12.4.1.2.2 Routine Maintenance

Routine maintenance is required for mechanical and electrical components throughout the plant. Maintenance includes routine scheduled maintenance and inspection activities. Occupational exposure can be reduced by reducing the number of components/equipment that require maintenance and providing ease of maintenance, accessibility of equipment, and ample workspace.

Examples of routine maintenance activities are as follows:

- a. Valves inspection and maintenance
- b. RCP oil changeout
- c. Equipment inspection and maintenance
- d. Radiation safety management and laundry
- e. Leak test
- f. Filter replacement
- f. Relamping works
- g. Removal and replacement of insulation
- h. Snubber inspection and repair

Table 12.4-2 provides a breakdown of the collective doses for routine maintenance activities.

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12.4.1.2.3 Inservice Inspection

ASME Code Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components” (Reference 4) requires extensive periodic inservice inspection of various safety-related systems/components and associated welds. These inservice inspections are generally performed during refueling outage.

Examples of inservice inspection activities are as follows:

- a. Pressure test
- b. Visual inspections
- c. Non-destructive and ultrasonic examinations
- d. Welds examinations
- e. Installation and removal of thermal insulation, platform, etc., and cleanup

Table 12.4-3 provides a breakdown of the collective doses for inservice inspection activities.

12.4.1.2.4 Special Maintenance

Maintenance beyond routine scheduled maintenance is considered special maintenance and can be performed only during refueling outages. This category includes the modification of equipment to upgrade the plant and repairs to any failed components. For special maintenance activities that lead to significant occupational exposure, ORE reduction can be achieved through improved reliability of components and equipments, ease of maintenance or replacement, use of temporary shielding, improvement of maintenance procedures, and use of remotely operated equipment and robotics.

Examples of special maintenance activities are as follows:

- a. Steam generator maintenance

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- b. Repairs on valves, pumps, heat exchanger, tanks, and other components
- c. Pressurizer maintenance

Table 12.4-4 provides a breakdown of the collective doses for special maintenance activities.

12.4.1.2.5 Waste Processing

Radioactive wastes, except spent fuels, are processed by the LWMS, SWMS, and GWMS in the compound building. Radioactive waste processing activities include processing, collecting, storage, and handling of radioactive liquids, solid, and gaseous wastes generated during plant operation. The systems are operated remotely to minimize the expected occupational exposure.

Table 12.4-5 provides a breakdown of the collective doses for waste processing activities.

12.4.1.2.6 Refueling

Refueling replaces depleted spent fuels with new fuels on an 18-month fuel cycle. This includes receipt and inspection of new fuel assemblies, transfer of fuel assemblies, and replacement of control rods or in-core components. Reactor pressure vessel (RPV) opening/closure activities are also included because the RPV head must be disassembled and removed before commencing refueling operations. Most of the refueling activities, including cask loading operations, are conducted underwater to provide shielding and maintain occupation dose ALARA. Occupational exposure is also reduced by using automatic cleaning equipment, remote fuel handling, and an integrated head assembly (IHA).

Examples of refueling activities are as follows:

- a. RPV opening/closure
- b. Refueling

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c. Stud bolt hole cleaning

Table 12.4-6 provides a breakdown of the collective doses for refueling activities.

12.4.1.2.7 Post-Accident Actions

Following a LOCA, the vital functions can be performed with an operator radiation exposure of no more than 50 mSv total effective dose equivalent (TEDE) in accordance with the alternative source term (AST), 10 CFR 50.34(f)(2)(vii), GDC 19, NUREG-0737 (Reference 5), Item II.B.2, and plant operating and emergency procedures.

The areas requiring continuous occupancy or infrequent access to perform post-accident vital functions are described in Subsection 12.3.1.8.

All vital areas requiring infrequent access are located in various rooms at different floor elevations in the auxiliary building (AB). The only significant submergence radiation dose that an individual would receive in accessing (transit time) and occupying (stay time) areas that are infrequently accessed is that from the airborne source. Because transit and stay time are not determined for the vital functions of interest, only airborne TEDE dose rates are considered. The plant emergency procedure mandates the use of respiratory protection to limit intake of airborne radioactive material when the potential for a radiation hazard of 1 DAC exists for the airborne activity. The minimum assigned protection factor (APF) of 10 for a half-mask respirator is used to calculate the inhaled dose rates for the areas requiring infrequent access. The respirator protections are not provided for areas requiring continuous occupancy, such as the main control room (MCR) and technical support center (TSC).

The dose contribution from the post-LOCA containment low-volume purge system release is negligible. Therefore, the whole body (WB), which is the deep dose equivalent (DDE), dose rates and the TEDE dose rates due to the post-LOCA containment and ESF leakages are calculated for radiation exposures beginning at 1 hour, 2 hours, 4 hours, 8 hours, and 24 hours (1 day) using AST methodology.

The post-accident sampling system (PASS) room and the remote shutdown panel room (RSPR) have ample concrete wall shielding (at least 2.7 m [9 ft]) to reduce the containment

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shine dose to a negligible level. The electric equipment rooms (EER) and the instrumentation and control (I&C) rooms have a minimum concrete shielding of just the containment cylindrical wall of 1.2 m (4 ft). Therefore, the post-LOCA containment shine dose rates are only considered in the electric equipment room and the I&C room locations.

The main control room and technical support center are identified as the areas requiring continuous occupancies to perform the post-accident vital functions. These areas are within the CR envelope (CRE) boundary. The post-LOCA doses in the CRE are listed in Table 6.4-1. To calculate the CR average dose rate, the CR operator occupancy time based on a 30-day accident duration and CR occupancy factors contained in NRC RG 1.183 (Reference 6) are considered.

The mission dose rates and allowable occupancy times for areas requiring infrequent access to perform post-accident vital functions are listed in Table 12.4-8, which indicates that there is sufficient stay time and transit times available to perform the required vital functions without exceeding the 50 mSv TEDE dose criterion.

The mission doses and dose rates for areas requiring continuous occupancy to perform post-accident vital functions are listed in Table 12.4-8, which indicates that required vital function from various vital areas can be safely performed without exceeding regulatory allowable 50 mSv TEDE dose limit. Per NUREG-0737, Item II.B.2, when applied to areas requiring continuous occupancy, the 50 mSv dose limit is met when the area dose rate is less than 0.15 mSv/hr averaged over 30 days.

12.4.2 Estimated Annual Dose at the Exclusion Area Boundary

The direct radiation from the reactor containment, auxiliary, compound, and turbine generator buildings is negligible compared with that from outside storage tanks. These tanks are expected to contain radioactivity that yields a dose rate of 0.001 mSv/hr or less at the tank shielded surface. Therefore, the estimated annual direct dose at the site boundary is expected to be small.

The estimated doses at the site boundary due to released activity are given in Subsections 11.2.6 and 11.3.6. The resultant annual doses for the general public are within the limits of the applicable regulations.

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12.4.3 Estimated Dose to Construction Workers

The construction worker doses are the sum of the exposures due to direct radiation and airborne radiation from the adjacent operating unit(s). The dose criterion to the construction workers is 1 mSv/yr. The construction worker doses are site-specific and based on the number of operating units, distances, meteorological conditions, and construction schedule. The COL applicant is to estimate construction worker doses based on site-specific number of operating units, distances, meteorological conditions, and construction schedule (COL 12.4(1)).

12.4.4 Combined License Information

COL 12.4(1) The COL applicant is to estimate construction worker doses based on the site-specific number of operating units, distances, meteorological conditions, and construction schedule.

12.4.5 References

1. NRC RG 8.8, "Information Relevant to Ensuring the Occupational Radiation Exposures at Nuclear Power Stations will be ALARA," Rev. 3, Nuclear Regulatory Commission, June 1978.
2. NRC RG 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Power Plants Design Stage Man-Rem Estimates," Rev. 1, Nuclear Regulatory Commission, June 1979.
3. NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," Nuclear Regulatory Commission.
4. ASME Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," The American Society of Mechanical Engineers, 2007.
5. NUREG-0737, "Clarification of TMI Action Plan Requirements," Nuclear Regulatory Commission, November 1980.
6. NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Nuclear Regulatory Commission, July 2000.

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Table 12.4-1

Occupational Dose Estimates During Reactor Operations and Surveillance

Activity	Average Dose Rate	Working Time	Annual ORE Dose
	[mSv/hr]	[man-hrs]	[man-Sv]
Routine inspection and patrol	0.003	4,301	0.014
System operation	0.001	11,697	0.015
Total			0.029

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Table 12.4-2

Occupational Dose Estimates During Routine Maintenance

Activity	Average Dose Rate	Working Time	Annual ORE Dose
	[mSv/hr]	[man-hrs]	[man-Sv]
CVCS (e.g., filter replacement, valve/pipe inspection, maintenance,) and primary circuit works (e.g., snubber inspection, repair)	0.007	484	0.003
MOV or other valve inspection and maintenance	0.008	195	0.002
Work on RCP (e.g., oil changeout)	0.025	1,340	0.033
Equipment inspection and maintenance	0.0003	1,523	0.001
Radiation safety management and laundry	0.002	18,503	0.046
Leak test (e.g., personal air lock)	0.0003	98	0.00003
Miscellaneous work (e.g., filter replacement, relamping)	0.002	4,639	0.009
Removal and replacement of insulation	0.001	3	0.000005
Total			0.093

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Table 12.4-3

Occupational Dose Estimates During Inservice Inspection

Activity	Average Dose Rate	Working Time	Annual ORE Dose
	[mSv/hr]	[man-hrs]	[man-Sv]
Inspection of welds, installation and removal of thermal insulation, platform, and similar activities. Non-destructive and ultrasonic examination, pressure test, cleanup, and similar activities..	0.033	898	0.030
Total			0.030

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Table 12.4-4

Occupational Dose Estimates During Special Maintenance

Activity	Average Dose Rate	Working Time	Annual ORE Dose
	[mSv/hr]	[man-hrs]	[man-Sv]
Servicing of control rod drives	0.058	59	0.003
SG primary side: man-way opening/closure	0.046	94	0.004
SG nozzle dam installation and removal	0.044	239	0.011
SG primary side: SG tube inspection	0.045	677	0.030
SG tube repairs	0.040	824	0.033
SG Secondary: man-way opening/closure	0.021	75	0.002
SG Secondary: inspection and cleaning	0.020	335	0.007
Other works on SG	0.009	6,764	0.062
Work on RHR and SI System	0.041	2,190	0.089
Works on RCP pump	0.008	2,210	0.020
Works on valves	0.013	384	0.005
Work on CVCS and RCS valve	0.011	107	0.005
Work on tank, sump, pump, heat exchanger, filter, evaporator, and so on	0.004	3,259	0.024
Maintenance of the pressurizer	0.026	128	0.003
Leak test on reactor containment building (ILRT and LLRT)	0.005	351	0.002
Total			0.300

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Table 12.4-5

Occupational Dose Estimates During Waste Processing

Activity	Average Dose Rate	Working Time	Annual ORE Dose
	[mSv/hr]	[man-hrs]	[man-Sv]
Radioactive waste processing	0.006	2,074	0.012
Total			0.012

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Table 12.4-6

Occupational Dose Estimates During Refueling

Activity	Average Dose Rate	Working Time	Annual ORE Dose
	[mSv/hr]	[man-hrs]	[man-Sv]
RPV opening and closure	0.049	1,027	0.050
RPV and internals inspection	0.017	412	0.007
In-core Detectors maintenance	0.058	663	0.038
Work on RPV Head and Flange and Stud hole Inspection and Cleaning	0.025	426	0.034
Work on Fuel and Refueling	0.005	2,211	0.011
Total			0.140

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Table 12.4-7

Annual Personnel Doses per Activity Categories

Category of Activity	Fraction [%]	Estimated Annual ORE Dose [man-Sv]
Reactor operations and surveillance	5 %	0.029
Routine maintenance	15 %	0.093
Inservice inspection	5 %	0.030
Special maintenance	50 %	0.300
Waste processing	2 %	0.012
Refueling	23 %	0.140
Total		0.603

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Table 12.4-8 (1 of 2)

Estimated Accident Mission Dose

- a. Vital area mission dose rates and allowable occupancy time for areas requiring infrequent access to perform post-accident vital functions

Post-LOCA (hr)	Whole Body DDE(mSv/hr)	Total TEDE (mSv/hr)	Allowable Occupancy (hr) ⁽¹⁾
Dose Rate Inside PASS			
1	2.4360E-01	1.0916E+01	4.58
2	7.2096E-01	2.7519E+01	1.82
4	3.9487E-01	6.0725E+00	8.23
8	2.0702E-01	2.6586E+00	18.81
24	2.8247E-02	5.0421E-01	99.16
Dose Rate Inside ERR			
1	7.1841E+00	1.7856E+01	2.8
2	1.0436E+01	3.7234E+01	1.34
4	5.3173E+00	1.0995E+01	4.55
8	1.8002E+00	4.2517E+00	11.76
24	1.5660E+00	2.0420E+00	24.49
Dose Rate Inside RSPR			
1	2.3505E-01	1.0907E+01	4.58
2	6.9567E-01	2.7493E+01	1.82
4	3.8102E-01	6.0586E+00	8.25
8	1.9976E-01	2.6513E+00	18.86
24	2.7256E-02	5.0322E-01	99.36
Dose Rate Inside I&C Room			
1	7.1186E+00	1.7791E+01	2.81
2	1.0242E+01	3.7040E+01	1.35
4	5.2110E+00	1.0889E+01	4.59
8	1.7445E+00	4.1960E+00	11.92
24	1.5584E+00	2.0344E+00	24.58

(1) Stay time to reach the 50 mSv TEDE dose criterion

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Table 12.4-8 (2 of 2)

- b. The vital area mission doses and dose rates for areas requiring continuous occupancy to perform post-accident vital functions

Area Requiring Continuous Occupancy	Dose (mSv TEDE)	Dose Rate (mSv/hr)	Regulatory Limit	
			Dose (mSv TEDE)	Dose Rate (mSv/hr)
MCR	40.00	0.13	50.00	0.15
TSC	40.00	0.13	50.00	0.15

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12.5 Operational Radiation Protection Program

The COL applicant is to provide the operational radiation protection program, which includes (Reference 1) (1) a documented management commitment to keep exposures ALARA, (2) a trained and qualified organization with sufficient authority and well-defined responsibilities, and (3) adequate facilities, equipment, and procedures to effectively implement the program (COL 12.5 (1)).

12.5.1 Combined License Information

COL 12.5(1) The COL applicant is to provide the operational radiation protection program, which includes (1) a documented management commitment to keep exposures ALARA, (2) a trained and qualified organization with sufficient authority and well-defined responsibilities, and (3) adequate facilities, equipment, and procedures to effectively implement the program.

12.5.2 References

1. NRC RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Nuclear Regulatory Commission, June 2007.