SOUTHERN COMPANY

UFSAR Table of Contents

UFSAR Formatting Legend

Color	Description
Black	Original Westinghouse AP1000 DCD Revision 19 content (part of plant-specific DCD)
Purple	Departures from AP1000 DCD Revision 19 content (part of plant-specific DCD)
Blue	Standard FSAR content
Green	Site-specific FSAR content
Red	Linked cross-references (chapters, appendices, sections, subsections, tables, figures, and references)

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Chapter 12 Radiation Protection

12.1 Assuring that Occupational Radiation Exposures Are As-Low-As-Reasonably Achievable (ALARA)

This section incorporates NEI 07-08A, Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA). VEGP 3&4 takes exception to the annual ALARA retraining frequency defined in NEI 07-08A Subsection 12.1.1, as evaluated in a Systematic Approach to Training Difficulty - Importance - Frequency analysis. ALARA retraining frequency is defined in Subsection 12.1.1.2. ALARA practices are developed in a phased milestone approach as part of the procedures necessary to support the Radiation Protection Program. Table 13.4-201 describes the major milestones for ALARA procedures development and implementation.

12.1.1 Policy Considerations

The AP1000 plant is designed with administrative programs and procedures to maximize the incorporation of good engineering practices and lessons learned to accomplish ALARA objectives.

12.1.1.1 Design and Construction Policies

The ALARA policy is applied during the design of AP1000. The design is reviewed for ALARA considerations and updated and modified as experience from operating plants is applied. ALARA reviews include the plant design and integrated layout, considering shielding, ventilation, and monitoring instrument designs as they relate to traffic control, security, access control and health physics.

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

Many of the engineers and supervisors assigned to the AP1000 design have performed similar design work or service work on other nuclear power plants. Through this experience, they have acquired knowledge of the radiation protection aspects which are applied to AP1000. Nuclear plant operating experience is incorporated through Nuclear Regulatory Commission (NRC) inspection and enforcement bulletins, information notices, and other documents. Independent reviews are conducted by the Electric Power Research Institute (EPRI) and Utility Steering Committee and its subcommittees. Knowledge of radiation protection and ALARA is applied to AP1000 design. This allows integration of experience and ALARA considerations from plant operators and plant designers and promotes incorporation of recent operating and service experience and lessons learned.

12.1.1.2 Operation Policies

Company and station policies are to keep all radiation exposure of personnel within limits defined by 10 CFR 20, Standards for Protection Against Radiation (Reference 12.1-1). Administrative procedures and practices related to maintaining radiation exposure of personnel as low as is reasonable achievable (ALARA) are described in this section.

The ALARA policy is consistent with the guidelines of Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable* (Reference 12.1-7), and Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable* (Reference 12.1-9) in establishing, organizing, and operating an effective ALARA program.

To varying extents, all station personnel are responsible for ALARA. Each supervisor is responsible for enforcing the ALARA requirements as described in 10CFR 20.1101. Individual workers are responsible for complying with ALARA requirements, which are presented during initial plant training and reinforced through retraining, at a frequency of every 24 months nominal, not to exceed 27 months, in accordance with the requirements contained in 10 CFR 19.12. The extent of ALARA training provided for each person is at least commensurate with the worker's job responsibilities and plant areas frequented. The radiation protection training program is maintained and implemented by the training department.

To ensure that personnel comply with established radiological policies, procedures and practices, radiation protection management personnel are charged with the responsibility to promptly advise higher management of any radiologically unsafe practices which exceed their authority to correct. They have the authority to halt any operation which, in their judgment, is radiologically unsafe. Radiation protection technicians are responsible for notifying the operations shift supervision or radiation protection management immediately in order to stop work on any operation deemed to be radiologically unsafe.

12.1.1.3 Compliance with 10 CFR 20 and Regulatory Guides 1.8, 8.8, and 8.10

Compliance of the design with 10 CFR 20 is confirmed by compliance of the design and operation of the facility within the guidelines of Regulatory Guides 1.8, 8.8, and 8.10. Compliance with Regulatory Guides 1.8, 8.8, and 8.10 is addressed as discussed in Subsection 12.1.1.

The design of AP1000 meets the guidelines of Regulatory Guide 8.8, Sections C.2 and C.4, which address facility, equipment and instrumentation design features. Features of the plant that are examples of compliance with Regulatory Guide 8.8 are delineated in Section 12.3.

As further discussed in Section 12.5, the station's ALARA policies and practices are consistent with the applicable regulations in 10 CFR 20 (Reference 201). Compliance with the guidance found in Regulatory Guides 1.8, 1.206, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, 8.38, and the applicable portions of NUREG-1736 (References 202 and 204 through 219) is discussed in Section 12.5, Radiation Protection. Compliance with Regulatory Guide 1.8 is further discussed in Section 13.2, Training. ALARA procedures are established, implemented, maintained and reviewed consistent with 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description, which is discussed in Section 17.5.

12.1.2 Design Considerations

Provisions and designs for maintaining personnel exposures ALARA are presented in the following paragraphs. The basic management philosophy guiding the AP1000 design effort so that radiation exposures are ALARA can be expressed as:

- Design structures, systems and components for reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components.
- Design structures, systems and components to reduce the radiation fields, thereby allowing operation, maintenance and inspection activities to be performed in the minimum design radiation field.
- Design structures, systems and components to reduce access, repair and removal times, thereby effectively reducing the time spent in radiation fields during operation, maintenance, and inspection.

- Design structures, systems and components to accommodate remote and semi-remote operation, maintenance and inspection, thereby effectively reducing the time spent in radiation fields.

12.1.2.1 General Design Considerations for ALARA Exposures

General design considerations and methods to maintain in-plant radiation exposures ALARA consistent with the recommendations of Regulatory Guide 8.8 have two objectives:

- Minimizing the necessity for access to and personnel time spent in radiation areas

- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention

Equipment and facility layouts and designs are considered for maintaining exposures ALARA during plant operations, including:

- Normal operation
- Maintenance and repairs
- Refueling operations and fuel storage
- Inservice inspection and calibrations
- Radioactive waste handling and disposal
- Other anticipated operational occurrences
- Decommissioning

The actual design features are described in Section 12.3. Examples of features that assist in maintaining exposures ALARA include:

- Provision of features to allow maintenance of state-of-the-art reactor coolant chemistry conditions, such that corrosion and consequential source terms are minimized: these include pH control capability sufficient to meet current and evolving industry standards and the ability to add zinc to the primary coolant

- Provision of features to allow draining, flushing, and decontaminating equipment and piping

- Design of equipment to minimize the creation and buildup of radioactive material and to ease flushing of crud traps

- Provision of shielding for personnel protection during maintenance or repairs and during decommissioning

- Provision of means and adequate space for the use of movable shielding

- Separation of more highly radioactive equipment from less radioactive equipment and provision of separate shielded compartments for adjacent items of radioactive equipment

- Provision of shielded access hatches for installation and removal of plant components

- Provision of design features, such as the chemical and volume control system, to minimize crud buildup

- Provision for means and adequate space for the use of remote and robotic maintenance and inspection equipment

- Simplifying the plant design compared to previous pressurized water reactors with design approaches such as:

  – Elimination of boron recycle
  – Elimination of evaporators
  – Use of an extended fuel cycle
  – Reduction in components containing radioactive fluids
  – Clearly and deliberately separating clean areas from potentially radioactive ones

12.1.2.2 Equipment General Design Considerations for ALARA

Equipment design considerations to minimize the necessity for, and amount of, time spent in a radiation area generally include:

- Reliability, durability, constructibility, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance.

- Servicing convenience for anticipated maintenance or potential repair, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair (For example, the passive residual heat removal heat exchanger is designed with extra tubes to allow for plugging of some tubes. Heat exchangers have drains to allow draining of the shell side water.)

- Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment.

- Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when there is no feasible method available to reduce radiation levels.

- Provisions for equipment to be operated from, and have its instrumentation and control in, accessible areas both during normal and abnormal operating conditions.

- Provisions for remote operation, draining and flushing of systems such as the chemical and volume control system.

- Past experience and lessons learned from servicing currently operating nuclear power plants.

Equipment design considerations directed toward minimizing radiation levels near equipment or components requiring personnel attention include:

- Selection of materials that minimize the creation of radioactive contamination.

- Provision of equipment and piping designs that minimize the accumulation of radioactive materials (for example, the use of seamless piping and minimizing the number of fittings reduces radiation accumulation at the seams and welds).

- Provisions for draining, flushing, or if necessary, remote cleaning or decontamination of equipment containing radioactive materials.

- Provision in the design for limiting leaks or controlling the fluid that does leak. This includes the use of high quality valves and valve packings, and the direction of leakage via drip pans and piping to sumps and floor drains.

- Provisions for isolating equipment from radioactive process fluids.
- Provisions for the chemical and volume control system; the spent fuel pit cleanup system; and the liquid radwaste cleanup system to limit radioactive isotope levels in the process water.

12.1.2.3 Facility Layout General Design Considerations for ALARA

Facility design considerations to minimize the amount of personnel time spent in a radiation area include the following:

- Locating equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection, in a manner that promotes ease of access and minimum of required occupancy time in radiation areas
- Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment
- Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area

Facility design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention generally include the following:

- Separating radiation sources and occupied areas, where practicable (for example, pipes or ducts containing potentially highly radioactive fluids do not pass through occupied areas). Redundant components requiring periodic maintenance that are a source of radiation are located in separate compartments to allow maintenance of one component while the other component is in operation.
- Providing shielding to separate equipment such as demineralizers and filters from nonradioactive equipment to provide unrestricted maintenance on the nonradioactive equipment.
- Providing shielding between radiation sources and access and service areas.
- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms. Adequate space is provided in labyrinth entrances for easy access. Highly radioactive passive components with minimal maintenance requirements are located in completely enclosed compartments and are provided with access via a shielded hatch or removable blocks.
- Separating equipment or components in service areas with permanent shielding, where appropriate.
- Providing means and adequate space for using movable shielding for sources within the service area, when required.
- Incorporating, within the plant layout, restrictions and control of access to the various radiation zones. Access to a given radiation zone generally does not require passing through a higher radiation zone. In the case of an abnormal occurrence or accident, the zone restrictions may change due to increased dose rates. Special access controls would be implemented at that time as discussed in Appendix 12AA.
- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.

- Providing control panels to permit remote operation of essential instrumentation and controls from the lowest radiation zone practicable.
- Providing means to control contamination or facilitate decontamination of potentially contaminated areas.
- Providing means for decontamination of service areas.
- Maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.
- Provide adequate lighting and support services (electrical power, compressed air, demineralized water, ventilation, and communications) at workstations.

12.1.2.4 Equipment and Facility Layout General Design Considerations for 10 CFR 20.1406

General equipment and facility layout design considerations to prevent the spread of contamination and to facilitate eventual decommissioning in accordance with 10 CFR 20.1406 include the features discussed in the following subsections.

12.1.2.4.1 Piping

The use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA.

To the extent possible, radioactive piping is located inside the auxiliary building and the containment vessel. This minimizes the potential for leakage to the groundwater from piping or fittings. The few exceptions are as follows:

- Process piping to and from the radwaste building (which can be fully visually inspected from the radwaste building pipe trench to the auxiliary building wall).
- Drain lines from the radwaste building and annex building back to the auxiliary building. These lines are not normally water filled, and can also be fully visually inspected from the annex or radwaste building pipe trench to the auxiliary building wall.
- Piping associated with the waste monitor tanks in the radwaste building. These tanks contain processed water, and they are located within the curbed radwaste building, which drains to the liquid radwaste system.
- Monitored radwaste discharge pipeline as discussed below.

The monitored radwaste discharge pipeline is engineered to preclude leakage to the environment. This pipe is routed from the auxiliary building to the radwaste building (the short section of pipe between the two buildings is fully available for visual inspection as noted above) and then out of the radwaste building to the appropriate point for dilution and discharge. The exterior piping either incorporates a guard pipe, or is available for visual inspection. No valves, vacuum breakers, or other fittings are incorporated outside of buildings.

As discussed in Subsections 11.1.2 and 11.1.3, operation with primary to secondary leakage would lead to limited radiological contamination of some secondary side systems. Liquid leakage from piping or components located in the turbine building will be collected by the waste water system and routed to a turbine building sump.

Outdoor piping between the condenser hotwell and the condensate storage tank is located above ground where liquid leakage can be detected.

12.1.2.4.2 Fuel Pool Design

The spent fuel pool and connected pools are designed to eliminate undetected leakage to the groundwater as follows:

- The walls of these pools are constructed using modular construction techniques. This allows higher quality than traditional construction. The advanced welding techniques used minimize the potential for weld failures during operation, and allow for inspection to verify weld quality.

- The walls are made of 1/2-inch stainless steel plates, joined to one another with full penetration welds.

- The thickness of the wall plate and the use of full penetration welds prevents wall or weld damage from fuel handling operations, including tool manipulation and storage.

- The pools are equipped with leak chases or alternate means of leak detection at each weld located below normal operating water level range upper value. For a limited number of pool structural welds where the use of leak chase channels behind the weld are not practical, alternate means that establish a leak tight inspectable barrier between the pool water and these structural welds are provided. The leak chase collection system uses piping that is adequately sized to allow testing and to minimize the potential for blockage by encrustation of precipitates (boric acid), and facilitates removal of any such blockage.

- The pool leak detection system will be zoned to allow identification of the area of the pool liner which is leaking, even for very small leaks.

- To the extent possible, these pools are located entirely inside the auxiliary and containment building, so that any theoretical leakage from the tanks is into the building rather than having the potential for release to the environment. Specifically, for pools other than a portion of the fuel transfer canal, the concrete support structure of the pools may be inspected from rooms adjacent to or below (i.e., outside) the pool.

12.1.2.4.3 Equipment Layout

A video record of the equipment layout in areas where radiation fields are expected to be high following operations may be used to assist in ALARA planning and to facilitate decommissioning.

12.1.3 Operational Considerations

The ALARA program is based on mature programs in use at other operating commercial nuclear facilities. As such, it inherently incorporates lessons-learned from decades of operating experience. Industry operating experience is regularly reviewed, and applicable lessons-learned are incorporated into plans, procedures, and policies as warranted.

Functional Structure

The functional structure for the ALARA program is described in Section 12.5.

Organizational Structure

Organizational structure is discussed in Section 13.1 and/or Section 17.5, the Quality Assurance Program Description.

Radiation Protection Program

The station has a radiation protection program and an ALARA program which contain the operational ALARA philosophy. These programs, made available to plant personnel, define management's commitment to ALARA and designate those individuals who have the responsibility and authority to implement the ALARA program. Section 12.5 provides a complete description of the radiation protection program.

Training

ALARA training is described in Sections 12.5 and 13.2.

Procedures

Procedures are developed and maintained in accordance with Sections 13.5 and 17.5. During initial preparation, radiation protection personnel review the procedures with potential radiological impact for operations, maintenance, refueling, inservice inspections, and operation of the radwaste system for compliance with ALARA guidelines outlined in the radiation protection program.

ALARA Program Review and Improvement

The ALARA Committee and plant management perform periodic reviews of the radiation program content and implementation. In addition to evaluating the implementation of the radiation protection program, these reviews are used to monitor workgroup trends as a means of controlling and reducing personnel exposure. All employees are encouraged to submit suggestions on methods of reducing personnel exposure and improving the ALARA program. Operating procedures are revised, as necessary, to incorporate ALARA lessons-learned from these reviews and suggestions.

Plant Modifications

Modifications to plant equipment and facilities are made where they will substantially reduce exposures at a reasonable cost. Specifications for replacement equipment reflect modifications based on experience gained from using the original equipment. Written procedures direct that all proposed plant modifications are screened for potential adverse radiological impacts. The initial screening review of these proposed modifications is typically performed by engineering personnel. Radiological protection and management personnel perform further review as warranted by level of potential radiological impact.

Work Practices

Radiation protection training, the radiation protection plan, the RWP system, and procedure reviews all help to ensure that radiation exposure of personnel is maintained ALARA. The following examples illustrate the incorporation of ALARA work practices:

- Personnel required to be monitored for radiation exposure in accordance with 10 CFR 20.1502 are assigned appropriate dosimetry to establish exposure history.

- Workers are provided with direct-reading dosimeters on jobs, so that the worker can determine accumulated exposure at any time during a job.

- Dose rate meters are used as needed to identify elevated dose rates.

- Pre-job briefs are used to review radiological surveys and to plan work before personnel enter a radiation area. Written procedures provide guidelines regarding the amount of detail to be included in the pre-job briefings.

- Post-job debriefs are used to ascertain lessons learned. Incorporation of these lessons may result in lower personnel exposure on future jobs. The requirement for post-job debriefings is specified by procedures.
- For work involving high radiation areas, high collective doses, high levels of removable contamination relative to site posting criteria for contamination areas, or known or suspected airborne radioactivity areas:
  1. Work is preplanned to minimize personnel exposure as defined in ALARA program procedures.
  2. Radiation protection personnel provide coverage as required by radiation protection procedures.
- On complex jobs in high radiation areas, dry-run training may be utilized. In some cases, mockups are used to familiarize workers with the operations that they are to perform. These techniques are beneficial to improving worker efficiency and minimizing the amount of time spent in the radiation field.
- On jobs where general area radiation levels are greater than 1.5 rem/hour or when individual exposure greater than 500 mrem per entry is expected, stay times are considered as further protection against unnecessary exposure.
- As practical, work area entry and exit points are established in areas with low radiation levels. This is done to minimize dose accumulated while changing protective clothing and respiratory equipment. Control points are also established to minimize the spread of removable contamination from the job site.
- As much as practicable, jobs and activities such as reading instruction manuals or maintenance procedures, adjusting tools or jigs, repairing valve internals, and prefabricating components are performed outside radiation areas.
- Individuals working in radiologically controlled areas are trained to be aware of the varying intensities of radiation fields within the general vicinity of their job locations, and are instructed to remain in the areas of lower radiation levels as much as possible, consistent with performing their assigned tasks.
- For high radiation area jobs, maps, postings, and/or detailed instructions are provided to clearly delineate the source of radiation or to alert personnel concerning the location of elevated dose rates. Provided with this information, workers will be cognizant of their immediate radiological environment, and will minimize their stay times in areas of elevated dose rates, thus maintaining exposures ALARA.
- Protective clothing and respiratory equipment prescribed by radiation protection personnel are commensurate with the radiological hazards involved. These requirements cannot be modified without the permission of radiation protection personnel. Consideration is given to the discomfort of workers to minimize the effect of protective efforts on efficiency and the time spent in a radiation area.
- Contamination containments (e.g., glove bags, plastic bottles, tents) and special ventilation systems (e.g., HEPA units) are used where practicable when personnel are working on highly contaminated equipment.

- Special tools or jigs are used on jobs when their use permits the job to be performed more efficiently or prevents errors, thus reducing the time spent in a radiation area.
- Where applicable, special tools are used to increase the distance from the source to the worker, thereby reducing the exposure received.
- Consideration is given to the use of remote monitoring of personnel with various combinations of audio, visual and dose information to reduce exposure of personnel. Direct communications (e.g., radios) may be used to further enhance radiation protection.
- Some systems and components which are subject to buildup of activated corrosion products are equipped with flush connections to reduce hot-spot buildup. Prior to performing maintenance work on these systems or components, consideration is given to flushing and/or chemically decontaminating the system or piece of equipment in order to reduce the crud levels, thereby reducing dose rates which may result in lower personnel exposure.
- Permanent shielding is used, where practicable, to reduce radiation exposure at the work site and in designated "waiting areas" for personnel during periods when they are not actively involved in the work.
- On some jobs, temporary shielding such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment is used. Temporary shielding is used only if the estimated total exposure, which includes exposure received during installation and removal, is reduced. Experience with such operations is used in developing guidelines in this area.

12.1.4 Combined License Information

Operational considerations of ALARA, as well as operational policies and continued compliance with 10 CFR 20 and associated Regulatory Guides, are addressed in NEI 07-08A and Appendix 12AA.

12.1.5 References

201. 10 CFR 20, Standards for Protection Against Radiation.

202. USNRC, “Qualification and Training of Personnel for Nuclear Power Plants,” Regulatory Guide 1.8, Revision 3, May 2000.

203. USNRC, “Quality Assurance Program Requirements (Operation)”, Regulatory Guide 1.33, Revision 2, February 1978.

204. USNRC, “Combined License Applications for Nuclear Power Plants (LWR Edition)”, Regulatory Guide 1.206, June 2007.

205. USNRC, “Guide for Administrative Practices in Radiation Monitoring”, Regulatory Guide 8.2, February 1973.

206. USNRC, “Instructions for Recording and Reporting Occupational Radiation Dose Data”, Regulatory Guide 8.7, Revision 2, November 2005.

207. USNRC, “Information Relevant to Ensuring that Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable”, Regulatory Guide 8.8, Revision 3, June 1978.

208. USNRC, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassy Program", Regulatory Guide 8.9, Revision 1, July 1993.

209. USNRC, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, September 1975.

210. USNRC, "Instructions Concerning Prenatal Radiation Exposure", Regulatory Guide 8.13, Revision 3, June 1999.

211. USNRC, "Acceptable Programs for Respiratory Protection", Regulatory Guide 8.15, Revision 1, October 1999.

212. USNRC, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants", Regulatory Guide 8.27, March 1981.

213. USNRC, "Audible-Alarm Dosimeters", Regulatory Guide 8.28, August 1981.

214. USNRC, "Instructions Concerning Risks from Occupational Radiation Exposure", Regulatory Guide 8.29, Revision 1, February 1996.

215. USNRC, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", Regulatory Guide 8.34, July 1992.

216. USNRC, "Planned Special Exposures", Regulatory Guide 8.35, June 1992.

217. USNRC, "Radiation Dose to the Embryo/Fetus", Regulatory Guide 8.36, July 1992.

218. USNRC, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants", Regulatory Guide 8.38, Revision 1, May 2006.

219. USNRC, "Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation", NUREG-1736, October 2001.

12.2 Radiation Sources

This section describes the sources of radiation 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 the three plant conditions of normal full-power operation, shutdown, and design basis accident events.

12.2.1.1 Sources for Full-Power Operation

The primary sources of radioactivity during normal full-power operation are direct core radiation, coolant activation processes, leakage of fission products from pinhole defects in fuel rod cladding, and activation of reactor coolant corrosion products. The design basis for fission product activities is operation with cladding defects in fuel rods producing 0.25 percent of the core thermal power. The design basis for activation and corrosion product activities is derived from measurements at operating plants and is independent of the fuel defect level.

12.2.1.1.1 Reactor Core

The neutron and gamma flux from the reactor core is reduced by the reactor internals and by the reactor vessel. Table 12.2-1 lists the neutron and gamma energy flux spectra in the reactor cavity outside the reactor vessel for several energy groups. The values are maximum values on the inside surface of the primary shield concrete at the core midplane.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the reactor coolant system are fission products released from fuel and activation of the coolant and of corrosion products that are circulated in the reactor coolant. These sources and their bases are described in Section 11.1.

The activation product, nitrogen-16 (N-16), is the predominant contributor to the activity in the reactor coolant pumps, steam generators, and reactor coolant piping during operation. The N-16 activity in each of the components depends on the total transit time to the component and the average residence time in the component. Table 12.2-3 presents the reactor coolant N-16 activity as a function of transport time in a reactor coolant loop. The N-16 activity for the pressurizer is tabulated in Table 12.2-4.

Fission and corrosion product activities circulating in the reactor coolant system and out-of-core crud deposits comprise the remaining significant radiation sources during full-power operation. The fission and corrosion product activities circulating in the reactor coolant are given in Section 11.1. The fission and corrosion product source strengths and specific activities in the pressurizer liquid and vapor phases are given in Table 12.2-5.

The isotopic composition and specific activity of typical out-of-core crud deposits are given in Table 12.2-6. Typically, one milligram of deposited crud material is found on one square centimeter of a relatively smooth surface. This may be as much as 50 times higher in crud trap areas. Crud trap areas are generally locations of high turbulence, areas of high momentum change, gravitational sedimentation areas, high affinity material areas, and possibly thin boundary layer regions.

The N-16 activity is not a factor in the radiation sources for systems and components located outside containment. This is due to its short half-life (7.11 seconds) and the greater than one minute transport

time before flow exits the containment. The normal letdown flow path is entirely inside containment. Primary coolant is directed outside containment only when it is diverted to the liquid radwaste system (e.g., due to boron dilution operations or for degassing prior to shutdown).

12.2.1.1.3 Chemical and Volume Control System

Radiation sources in the chemical and volume control system consist of radionuclides carried in the reactor coolant. The chemical and volume control system components in the purification path are located inside containment. The chemical and volume control system carries radioactive fluid out of the containment only when reactor coolant is directed to the liquid radwaste system.

The shielding design of the chemical and volume control system components is based on processing reactor coolant having the design basis source term presented in Section 11.1. The regenerative and letdown heat exchanger sources include contributions from N-16. Owing to its short half-life, the concentration of N-16 is highly sensitive to the location of these heat exchangers with respect to the reactor coolant loop piping. The concentration of N-16 at the heat exchangers is assumed to be the value in the reactor coolant when it exits the steam generator (see Table 12.2-3). The radiation sources for the other components in the purification loop do not include a contribution from N-16. The N-16 contribution to the shielding source term for the filter and demineralizers is determined based on the additional decay afforded by the time delay resulting from the system layout. The chemical and volume control system component sources are provided in Table 12.2-7.

12.2.1.1.4 Service Water System and Component Cooling Water System

These systems are normally nonradioactive or, if there is inleakage of radioactive material into the systems, of very low level activity. For shielding and dose assessment purposes, these systems are assumed to be nonradioactive.

12.2.1.1.5 Spent Fuel Pool Cooling System

One of the functions of the spent fuel pool cooling system is to provide cleanup of the water in the spent fuel pool, the refueling cavity, and the in-containment refueling water storage tank. The equipment considered in designing shielding are the spent fuel pool cooling system demineralizers and filters which accumulate activity, primarily Co-58 and Co-60 from radioactive crud that is resuspended in the water during the course of fuel handling. The source terms for this equipment are provided in Table 12.2-8. Based on operating experience, the remainder of the spent fuel pool cooling system may contain a significant amount of crud and thus requires shielding. The composition of crud is provided in Table 12.2-6.

12.2.1.1.6 Main Steam Supply System

Potential radioactivity in the main steam supply system is a result of steam generator tube leaks and is sufficiently low so that radiation shielding is needed around secondary components, which are capable of concentrating the system’s activity, including the steam generator blowdown system (BDS) filter, the electrodeionization (EDI) unit, and the condensate polisher(s).

12.2.1.1.7 Liquid Radwaste System

Radioactive inputs include fission and activation product radionuclides produced in the core and reactor coolant. Shielding for each component of the liquid radwaste system is based on the sources listed on Table 12.2-9. Radiation sources for the various pumps in the liquid radwaste system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

12.2.1.1.8 Gaseous Radwaste System

Radioactive gases and hydrogen removed from the reactor coolant when coolant is discharged to the liquid radwaste system comprise the bulk of the gas processed by the gaseous radwaste system. There is no gas stripping performed in the reactor coolant purification loop of the chemical and volume control system. The result is that the volume of gases processed by the gaseous radwaste system is small. Table 12.2-10 lists the shielding sources for the components in the gaseous radwaste system.

12.2.1.1.9 Solid Radwaste System

The solid radwaste system handles various radioactive waste products ranging from relatively low activity materials to high activity spent resins and filter cartridges. Solid wastes are packaged for shipment to a burial or long-term storage facility.

Prior to packaging, the spent resin is stored in a spent resin storage tank. Two spent resin storage tanks are provided, one for high activity resins and the other for low activity resins. The initial gamma source strength in the high activity spent resin storage tank is assumed to be the same as that in the chemical and volume control system mixed bed demineralizer. After a 30-day decay period, only the cesium and cobalt isotopes are significant contributors to the radiation field. Table 12.2-11 lists the source strengths and specific activities both initially and after 30 days of decay.

Spent filter cartridge sources are as listed in Tables 12.2-7, 12.2-8, and 12.2-9.

12.2.1.1.10 Miscellaneous Sources

There are additional contained sources used for instrument calibration or for radiography. These sources will be identified as discussed in Subsection 12.2.3.

Licensed sources containing byproduct, source, and special nuclear material that warrant shielding design consideration meet the applicable requirements of 10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50, and 70.

There are byproduct and source materials with known isotopes and activity manufactured for the purpose of measuring, checking, calibrating, or controlling processes quantitatively or qualitatively.

These sources include but are not limited to:

- Sources in field monitoring equipment.
- Sources in radiation monitors to maintain a threshold sensitivity.
- Sources used for radiographic operations.
- Depleted uranium slabs used to determine beta response and correction factors for portable monitoring instrumentation.
- Sources used to calibrate and response check field monitoring equipment (portable and fixed).
- Liquid standards and liquids or gases used to calibrate and verify calibration of laboratory counting and analyzing equipment.
- Radioactive waste generated by the use of radioactive sources.

Specific details of these sources are maintained in a database on-site following procurement. This database, at a minimum, contains the following information:

- Isotopic composition
- Location in the plant
- Source strength
- Source geometry

Written procedures are established and implemented that address procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance and use of these radioactive sources. These procedures are developed in accordance with the radiation protection program to comply with 10 CFR Parts 19 and 20. A supplementary warning symbol is used in the presence of large sources of ionizing radiation consistent with the guidance in Regulatory Issue Summary (RIS) 2007-03.

Sources maintained on-site for instrument calibration purposes are shielded while in storage to keep personnel exposure ALARA. Sources used to service or calibrate plant instrumentation are also routinely brought on-site by contractors. Radiography is performed by the licensed utility group or licensed contractors. These sources are maintained and used in accordance with the provisions of the utility group's or contractor's license. Additional requirements and restrictions may apply depending on the type of source, use, and intended location of use. If the utility group or contractor source must be stored on-site, designated plant personnel must approve the storage location, and identify appropriate measures for maintaining security and personnel protection.

During the period prior to the implementation of the Emergency Plan (in preparation for the initial fuel loading following the 52.103(g) finding), no specific materials related emergency plan will be necessary because:

a) No byproduct material will be received, possessed, or used in a physical form that is "in unsealed form, on foils or plated sources, or sealed in glass," that exceeds the quantities in Schedule C in 10 CFR 30.72, and

b) No 10 CFR Part 40 specifically licensed source material, including natural uranium, depleted uranium and uranium hexafluoride will be received, possessed, or used during this period.

The following radioactive sources will be used for the Radiation Monitoring System and laboratory/ portable monitoring instrumentation:[1]

Radioactive Licensee Material (Element and Mass Number)[1]	Chemical and/or Physical Form[1]	Maximum Quantity That Licensee May Possess at Any One Time[1]
• Any byproduct material with atomic numbers 1 through 93 inclusive	Sealed Sources[2]	No single source to exceed 100 millicuries 5 Curies total
• Americium-241	Sealed Sources[2]	No single source to exceed 300 millicuries 500 millicuries total

Notes:

1. This information remains in effect between the issuance of the COL and the Commission's 52.103(g) finding for each unit, and will be designated historical information after that time.
2. Includes calibration and reference sources.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring permanent shielding consideration are the spent reactor fuel and the residual heat removal system. Individual components may require shielding during shutdown due to deposited crud material. Estimates of accumulated crud in the reactor coolant system are given in Subsection 12.2.1.1. The radiation sources in the reactor coolant system and other systems addressed in Subsection 12.2.1.1 are bounded by the sources given for full power operation with the exception of a short time period (less than 24 hours) following shutdown, during which crud bursts can result in increased radiation sources. Crud bursts are the resuspension of a portion of the accumulated deposited corrosion products into the reactor coolant system during shutdown operation. Activity increases also occur during planned coolant oxygenation procedures prior to refueling activities.

12.2.1.2.1 Normal Residual Heat Removal System

The maximum gamma ray source strengths in the normal residual heat removal system for four and eight hours after reactor shutdown are given in Table 12.2-12 along with the listing of contributing nuclides. The system may be placed in operation at the maximum flow rate at approximately four hours following a shutdown. The system removes decay heat from the reactor for the duration of the shutdown. The sources given are maximum values taking into account activity increases due to coolant oxygenation measures.

12.2.1.2.2 Reactor Core

The core average gamma ray and neutron source strengths are used in the evaluation of radiation levels within and around the shutdown reactor.

The basis for the core average source strengths is an equilibrium cycle core at end-of-life. Feed enrichment of 4.9 (68 assemblies) weight-percent U-235 was assumed. The regions operate at a specific power of 40.7 megawatts (thermal) per metric ton of uranium for 520, 1040, and 1561 effective full-power days, respectively (this is for an 18 month fuel cycle with a 95 percent capacity factor).

Core average gamma ray source strengths are presented in Table 12.2-2 for various times after shutdown. These source strengths may be put on a per-unit volume of homogenized core basis by multiplying them by the core power density (109.7 watts/cc).

Neutrons are produced in the shutdown reactor by spontaneous fission of the transplutonium isotopes and by (α, n) reactions of alpha particles with 0-17 and 0-18 in the uranium dioxide fuel.

Core average neutron source strengths are given in Table 12.2-13 for various times after shutdown. The neutron source strengths may be put on a per-unit volume of homogenized core basis by multiplying them by the power density.

12.2.1.2.3 Spent Fuel

Spent fuel gamma ray and neutron source strengths are used in the evaluation of radiation levels for fuels handling, spent fuel storage, and shipping.

The basis for the spent fuel data presented here is the discharge region of an equilibrium cycle core at end of life. A feed enrichment of 4.9 weight-percent U-235 is assumed. The discharge region was operated at a specific power of 40.7 megawatts (thermal) per metric ton of uranium for 1561 effective full-power days.

Spent fuel gamma ray source strengths are presented in Table 12.2-14 for various times after shutdown. These source strengths may be put on a per-unit volume of homogenized core basis by multiplying by the power density (109.7 watts/cc).

Spent fuel neutron source strengths are given in Table 12.2-13 for various times after shutdown. The neutron source strengths may be put on a per-unit volume of homogenized core basis by multiplying them by the power density.

12.2.1.2.4 Irradiated Control Rods, Gray Rods, and Secondary Source Rods

The gamma ray source strengths of the irradiated control rods, gray rods, and secondary source rods are used in establishing radiation shielding requirements during refueling operations and during shipping of irradiated rods.

The absorber material used in the control rods is silver-indium-cadmium (Ag-In-Cd). The gray rods contain tungsten in a nickel alloy 718 sleeve. The gamma ray source strengths associated with the irradiated Ag-In-Cd control rod absorber and gray rod materials are listed in Table 12.2-15 for various times after shutdown.

The photoneutron source material used in the secondary source rods is an equal volume mixture of antimony and beryllium (Sb-Be). The gamma ray source strengths associated with the secondary source rods are listed in Table 12.2-16 for various times after shutdown and Table 12.2-17 lists the neutron source strengths. The source values are per cubic centimeter of source material for an irradiation period of 400 days.

The material used for the control rod cladding, gray rod cladding and/or pellets and secondary source rod cladding is Type 304 stainless steel with a maximum cobalt content of 0.05 weight percent. The gamma ray source strengths associated with the irradiated stainless steel are listed in Table 12.2-18 for various times after shutdown.

12.2.1.2.5 Incore Flux Thimbles

Irradiated incore flux thimble gamma ray source strengths are given in Table 12.2-19.

12.2.1.3 Sources for the Core Melt Accident

The AP1000 is designed to provide adequate core cooling in the event of a postulated loss of coolant accident (LOCA) so that there is no significant core damage. Following a LOCA, the normal residual heat removal system could be used, if available, to provide post-accident cooling. Use of the normal residual heat removal system is acceptable only if the source term is close to the design basis source term (see Table 12.2-12).

For the evaluation of the radiological consequences of the LOCA, it is assumed that major degradation of the core takes place, including melting of the core. The source term used for the LOCA dose analysis assumes no core release for 10 minutes, then there is a gap release from a small number of fuel rods before the onset of core degradation. The first half hour of core release is restricted to releases from the fuel cladding gap; this gap release phase is followed by the in-vessel core melt phase that has a duration of 1.3 hours. After the in-vessel core melt phase, there is assumed to be no further release of activity from the core. This core activity release model is based on the source term model from NUREG-1465 (Reference 1). The source term is described in detail in Subsection 15.6.5.3.

If there is core degradation, core cooling would be provided by the passive core cooling system which is totally inside the containment such that no high activity sump solution would be recirculated

outside the containment. The shielding provided for the containment addresses this post-LOCA source term. The source strengths as a function of time are provided in Table 12.2-20 and the integrated source strengths are provided in Table 12.2-21.

12.2.2 Airborne Radioactive Material Sources

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

12.2.2.1 Containment Atmosphere

The main sources of airborne activity in the containment is leakage of primary coolant and activation of naturally occurring argon in the atmosphere. During normal power operation, excessive activity buildup in the containment atmosphere is prevented by periodic purging of the containment (approximately 20 hours per week). When the plant is shut down for refueling or maintenance, additional purging of the containment atmosphere is performed to further reduce the activity levels consistent with the increased level of worker presence in the containment. The assumptions and parameters used to determine the airborne activity levels in the containment are listed in Table 12.2-22. The airborne concentrations are provided in Table 12.2-23. Three situations are considered: normal power operation without purge, normal power operation with 20 hours of purge operation per week, and shutdown operation.

12.2.2.2 Fuel-Handling Area Atmosphere

The source of airborne activity in the fuel-handling area is leakage from stored spent fuel assemblies and the evaporation losses from the spent fuel pool. The maximum airborne concentration in the fuel-handling area is calculated using the assumptions and parameters in Table 12.2-24. The resulting airborne isotopic concentrations are provided in Table 12.2-25.

12.2.2.3 Auxiliary Building Atmosphere

The source of airborne activity in the auxiliary building atmosphere is primarily equipment leakage. The ventilation system constantly removes activity and discharges it to the plant vent. The maximum airborne concentration in the auxiliary building is calculated using the assumptions and parameters in Table 12.2-26. The resulting airborne isotopic concentrations are provided in Table 12.2-27.

12.2.2.4 Airborne Activity Model

The airborne concentration of each nuclide in the atmosphere is calculated by:

$$C_i(t) = \frac{(LR)_i\, A_i\,(PF)_i\,(1 - e^{-\lambda_{Ti}\, t})}{(V)\,(\lambda_{Ti})}$$

where:

$(LR)_i$ = leak or evaporation rate of the I^{th} radioisotope in the applicable region (g/s)

A_i = activity concentration of the I^{th} leaking or evaporating radioisotope (µCi/g)

$(PF)_i$ = partition factor or the fraction of the leaking activity that is airborne for the I^{th} radioisotope

λ_{Ti} = total removal rate constant for the I^{th} radioisotope from the applicable region (s^{-1})

= $\lambda_{di} + \lambda_e$, the removal rate constants in s^{-1} due to radioactive decay for the I^{th} radioisotope and the exhaust from the applicable region, respectively

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

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

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

From the above equation, the peak or equilibrium concentration, C_{Egi}, of the I^{th} radioisotope in the applicable volume is given by the following expression:

$$C_{Egi} = (LR)_I\, A_I\, (PF)_I\, /V\lambda_{Ti}$$

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

12.2.3 Combined License Information

The additional contained radiation sources not identified in Subsection 12.2.1, including radiation sources used for instrument calibration or radiography, are addressed in Subsection 12.2.1.1.10.

12.2.4 References

1. L. Soffer, et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

Table 12.2-1
Radiation Flux at the Primary Shield Concrete

Neutron Flux	
Energy	**Flux (n/cm^2-sec)**
≥ 1 Mev	2.4E+09
0.1 Mev to 1 Mev	1.8E+10
0.414 ev to 0.1 Mev	2.7E+10
< 0.414 ev	6.4E+9
Gamma Energy Flux	
Energy	**Flux (Mev/cm^2-sec)**
≥ 6.0 Mev	1.2E+10
3.0 Mev to 6.0 Mev	8.3×10^9
1.0 Mev to 3.0 Mev	5.9×10^9
< 1.0 Mev	3.9×10^9

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

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0.000-0.020	1.8E+08	1.5E+08	6.5E+07	3.1E+07	1.4E+06
0.020-0.030	4.2E+07	3.7E+07	2.2E+07	1.6E+07	3.9E+06
0.030-0.045	1.1E+08	1.0E+08	6.1E+07	4.5E+07	1.5E+07
0.045-0.070	6.5E+07	5.9E+07	3.2E+07	2.0E+07	1.9E+06
0.070-0.100	3.6E+08	3.2E+08	1.6E+08	9.4E+07	8.6E+06
0.100-0.150	1.5E+09	1.3E+09	6.3E+08	3.6E+08	8.7E+07
0.150-0.300	2.2E+09	1.8E+09	7.1E+08	3.5E+08	1.4E+07
0.300-0.450	9.3E+08	8.2E+08	5.2E+08	3.9E+08	7.6E+07
0.450-0.700	5.3E+09	4.3E+09	2.4E+09	1.9E+09	9.5E+08
0.700-1.000	6.8E+09	5.9E+09	4.3E+09	3.8E+09	2.6E+09
1.000-1.500	2.0E+09	1.2E+09	4.9E+08	3.3E+08	9.2E+07
1.500-2.000	3.4E+09	3.1E+09	2.6E+09	2.3E+09	6.6E+08
2.000-2.500	3.0E+08	2.0E+08	1.4E+08	1.2E+08	4.9E+07
2.500-3.000	1.9E+08	1.8E+08	1.5E+08	1.3E+08	3.9E+07
3.000-4.000	5.2E+06	1.9E+06	1.6E+06	1.4E+06	4.4E+05
4.000-6.000	3.5E+05	1.9E+04	4.6E+00	4.6E+00	4.4E+00
6.000-8.000	7.4E-01	7.4E-01	7.4E-01	7.4E-01	7.1E-01
8.000-11.000	1.2E-01	1.2E-01	1.2E-01	1.1E-01	1.1E-01

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

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)			
	3 Months	6 Months	1 Year	5 Years
0.000-0.020	3.0E+05	1.7E+05	1.0E+05	2.9E+04
0.020-0.030	9.6E+05	3.8E+05	2.2E+05	7.4E+04
0.030-0.045	6.7E+06	4.3E+06	2.8E+06	4.5E+05
0.045-0.070	2.5E+05	1.6E+05	1.1E+05	5.6E+04
0.070-0.100	2.5E+06	1.9E+06	1.3E+06	1.4E+05
0.100-0.150	3.4E+07	1.6E+07	8.7E+06	7.0E+05
0.150-0.300	1.5E+06	6.4E+05	3.8E+05	2.0E+05
0.300-0.450	5.6E+06	2.5E+06	1.9E+06	6.4E+05
0.450-0.700	4.8E+08	3.1E+08	2.3E+08	9.4E+07
0.700-1.000	1.5E+09	7.2E+08	2.0E+08	3.3E+07
1.000-1.500	4.3E+07	3.5E+07	2.7E+07	8.4E+06
1.500-2.000	2.9E+07	2.8E+06	1.8E+06	2.9E+05
2.000-2.500	2.1E+07	1.6E+07	1.0E+07	3.1E+05
2.500-3.000	1.7E+06	1.6E+05	1.0E+05	6.6E+03
3.000-4.000	4.8E+04	2.8E+04	1.9E+04	1.2E+03
4.000-6.000	4.1E+00	3.8E+00	3.4E+00	2.7E+00
6.000-8.000	6.7E-01	6.2E-01	5.5E-01	4.4E-01
8.000-11.000	1.0E-01	9.6E-02	8.6E-02	6.8E-02

Table 12.2-3
Reactor Coolant Nitrogen-16 Activity

Position in Loop	Loop Transit Time (sec)	Nitrogen-16 Activity (μCi/g)
Leaving core	0.0	280
Leaving reactor vessel	0.9	256
Entering steam generator	1.2	249
Leaving steam generator	6.8	144
Entering reactor vessel	8.0	128
Entering core	9.5	122
Leaving core	10.3	280

Table 12.2-4
Pressurizer Nitrogen-16 Source Strengths

Discrete Energy (Mev/gamma)	Energy Group (Mev/gamma)	Source Strength	
		Liquid Phase[(a)] (Mev/gram-sec)	Steam Phase[(b)] (Mev/cm^3-sec)
1.75	1.35 - 1.80	5.4E-02	1.4E-01
2.74	2.6 - 3.0	5.0E-01	1.3E+00
6.13	6.0 - 7.0	1.0E+02	2.6E+02
7.12	7.0 - 7.5	8.5E+00	2.2E+01

Notes:
a. Based on an insurge to the pressurizer following a ten percent step load power increase.
b. Based on a boron equalization spray rate of 80 gpm.

Table 12.2-5 (Sheet 1 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1000 Cubic Foot Liquid Phase Source Strengths	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	1.4E+03
0.02-0.03	1.6E+03
0.03-0.045	6.9E+04
0.045-0.07	3.2E+02
0.07-0.1	1.4E+05
0.1-0.15	1.1E+03
0.15-0.3	4.3E+04
0.3-0.45	2.3E+04
0.45-0.7	1.0E+05
0.7-1.0	1.3E+05
1.0-1.5	1.2E+05
1.5-2.0	5.9E+04
2.0-2.5	9.3E+04
2.5-3.0	1.5E+04
3.0-4.0	2.6E+03
4.0-6.0	5.0E+02
6.0-8.0	-
8.0-11.00	-
Total	7.9E+05

Table 12.2-5 (Sheet 2 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1000 Cubic Foot Liquid Phase Specific Activity	
Nuclide	**Activity (µCi/gram)**
Kr-87	4.7E-01
Kr-88	1.5E+00
Kr-89	3.5E-02
Xe-131m	1.3E+00
Xe-133m	1.7E+00
Xe-133	1.2E+02
Xe-135	3.5E+00
Br-84	1.7E-02
I-131	7.1E-01
I-132	9.3E-01
I-133	1.3E+00
I-134	2.2E-01
I-135	7.8E-01
Rb-88	1.5E+00
Cs-134	6.9E-01
Cs-136	1.0E+00
Cs-138	3.7E-01
Tc-99m	2.0E-01

Table 12.2-5 (Sheet 3 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1100 Cubic Foot Steam Phase Source Strengths (Mev/cm^3-sec)	
Energy Group (Mev/gamma)	**Normal 2-gpm Spray**
0-0.02	2.3E+03
0.02-0.03	2.1E+03
0.03-0.045	1.3E+05
0.045-0.07	2.0E-04
0.07-0.1	2.7E+05
0.1-0.15	1.9E+00
0.15-0.3	6.7E+03
0.3-0.45	5.0E+02
0.45-0.7	1.7E+03
0.7-1.0	8.4E+02
1.0-1.5	6.3E+02
1.5-2.0	6.3E+02
2.0-2.5	2.9E+03
2.5-3.0	1.3E+02
3.0-4.0	4.3E+00
4.0-6.0	2.2E-05
6.0-8.0	-
8.0-11.00	-
Total	4.1E+05

Table 12.2-5 (Sheet 4 of 4)
Pressurizer Liquid and Steam Phase
Source Strengths and Specific Activity

1100 Cubic Foot Steam Phase Specific Activity (μCi/cm^3)	
Nuclide	**Normal 2-gpm Spray**
Kr-85m	5.2E-02
Kr-85	9.0E+00
Kr-87	6.7E-03
Kr-88	5.5E-02
Kr-89	1.7E-08
Xe-131m	2.9E+00
Xe-133m	1.4E+00
Xe-133	2.4E+02
Xe-135	4.6E-01
Xe-138	1.7E-04
I-131	7.1E-03
I-132	9.3E-03
I-133	1.3E-02
I-134	2.2E-03
I-135	7.8E-03

Table 12.2-6
Isotopic Composition and Specific Activity of Typical Out-of-Core Crud Deposits[a]

Composition (Nuclide)	Activity (µCi/mg) of Deposited Crud for Effective Full Power Years of Plant Operation			
	1 Year	**2 Years**	**5 Years**	**10 Years**
Mn-54	1.0	1.1	1.3	1.4
Fe-59	0.5	0.5	0.5	0.5
Co-58	12.0	12.0	12.0	12.0
Co-60	1.5	2.3	4.0	6.0

Note:

a. In addition to corrosion products, about 1.0 µg of mixed actinides and fission products may be present for each 1 g of deposited crud.

Table 12.2-7 (Sheet 1 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

A. Regenerative heat exchanger		
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**	
	Tube Side	**Shell Side**
0-0.02	1.4E+03	1.4E+03
0.02-0.03	1.6E+03	1.6E+03
0.03-0.045	6.9E+04	6.9E+04
0.045-0.07	3.2E+02	3.2E+02
0.07-0.1	1.4E+05	1.4E+05
0.1-0.15	1.1E+03	1.1E+03
0.15-0.3	4.3E+04	4.2E+04
0.3-0.45	2.3E+04	1.5E+04
0.45-0.7	1.0E+05	4.7E+04
0.7-1.0	1.3E+05	7.5E+04
1.0-1.5	1.2E+05	7.7E+04
1.5-2.0	7.1E+04	4.2E+04
2.0-2.5	9.3E+04	8.9E+04
2.5-3.0	1.3E+05	1.5E+04
3.0-4.0	2.6E+03	2.2E+03
4.0-6.0	5.0E+02	5.0E+02
6.0-8.0	2.5E+07	-
8.0-11.00	-	-
Total	2.5E+07	6.2E+05

Table 12.2-7 (Sheet 2 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

A. Regenerative heat exchanger

Nuclide	Activity (µCi/gram)	
	Tube Side	**Shell Side**
Kr-87	4.7E-01	4.7E-01
Kr-88	1.5E+00	1.5E+00
Kr-89	3.5E-02	3.5E-02
Xe-131m	1.3E+00	1.3E+00
Xe-133m	1.7E+00	1.7E+00
Xe-133	1.2E+02	1.2E+02
Xe-135	3.5E+00	3.5E+00
Br-84	1.7E-02	--
I-131	7.1E-01	--
I-132	9.3E-01	--
I-133	1.3E+00	--
I-134	2.2E-01	--
I-135	7.8E-01	--
Rb-88	1.5E+00	1.5E+00
Cs-134	6.9E-01	6.9E-01
Cs-136	1.0E+00	1.0E+00
Cs-138	3.7E-01	3.7E-01
Tc-99m	2.0E-01	2.0E-01
Ba-137m	--	4.7E-01
N-16	1.4E+02	--

Table 12.2-7 (Sheet 3 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

B. Letdown heat exchanger	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	1.4E+03
0.02-0.03	1.6E+03
0.03-0.045	6.9E+04
0.045-0.07	3.2E+02
0.07-0.1	1.4E+05
0.1-0.15	1.1E+03
0.15-0.3	4.3E+04
0.3-0.45	2.3E+04
0.45-0.7	1.0E+05
0.7-1.0	1.3E+05
1.0-1.5	1.2E+05
1.5-2.0	7.1E+04
2.0-2.5	9.3E+04
2.5-3.0	1.3E+05
3.0-4.0	2.6E+03
4.0-6.0	5.0E+02
6.0-8.0	2.5E+07
8.0-11.00	-
Total	2.5E+07

Table 12.2-7 (Sheet 4 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

B. Letdown heat exchanger	
Nuclide	**Activity (µCi/gram)**
Kr-87	4.7E-01
Kr-88	1.5E+00
Kr-89	3.5E-02
Xe-131m	1.3E+00
Xe-133m	1.7E+00
Xe-133	1.2E+02
Xe-135	3.5E+00
Br-84	1.7E-02
I-131	7.1E-01
I-132	9.3E-01
I-133	1.3E+00
I-134	2.2E-01
I-135	7.8E-01
Rb-88	1.5E+00
Cs-134	6.9E-01
Cs-136	1.0E+00
Cs-138	3.7E-01
Tc-99m	2.0E-01
N-16	1.4E+02

Table 12.2-7 (Sheet 5 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

C. Mixed bed demineralizer (50 cubic feet of resin)	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	1.7E+04
0.02-0.03	1.1E+05
0.03-0.045	3.5E+05
0.045-0.07	4.8E+04
0.07-0.1	2.2E+05
0.1-0.15	4.8E+03
0.15-0.3	2.2E+06
0.3-0.45	2.8E+07
0.45-0.7	1.8E+08
0.7-1.0	9.9E+07
1.0-1.5	2.8E+07
1.5-2.0	4.2E+06
2.0-2.5	8.7E+05
2.5-3.0	7.4E+05
3.0-4.0	9.9E+04
4.0-6.0	2.7E+03
6.0-8.0	-
8.0-11.00	-
Total	3.4E+08

Table 12.2-7 (Sheet 6 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

C. Mixed bed demineralizer (50 cubic feet of resin)	
Nuclide	**Activity (µCi/gram)**
Mn-54	6.2E+01
Mn-56	1.3E+01
Co-58	7.2E+01
Co-60	7.8E+01
I-131	2.4E+03
I-132	4.3E+01
I-134	3.9E+00
I-135	9.9E+01
Rb-88	8.9E+00
Cs-134	3.0E+03
Cs-136	1.6E+02
Cs-137	3.0E+03
Ba-137m	2.8E+03

Table 12.2-7 (Sheet 7 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

D. Cation bed demineralizer (50 cubic feet of resin)	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	6.3E+03
0.02-0.03	-
0.03-0.045	3.3E+05
0.045-0.07	4.8E+04
0.07-0.1	3.1E+04
0.1-0.15	2.9E+03
0.15-0.3	5.1E+05
0.3-0.45	9.9E+05
0.45-0.7	1.6E+08
0.7-1.0	9.0E+07
1.0-1.5	1.6E+07
1.5-2.0	1.9E+06
2.0-2.5	4.4E+05
2.5-3.0	6.9E+05
3.0-4.0	9.3E+04
4.0-6.0	2.7E+03
6.0-8.0	-
8.0-11.00	-
Total	2.7E+08

Table 12.2-7 (Sheet 8 of 8)
Chemical and Volume Control System Components
Source Strengths and Specific Activity

D. Cation bed demineralizer (50 cubic feet of resin)	
Nuclide	**Activity (µCi/gram)**
Rb-88	8.9E+00
Cs-134	3.0E+03
Cs-136	1.6E+02
Cs-137	3.0E+03
Cs-138	3.7E+00
Ba-137m	2.8E+03
E. Reactor coolant filter	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	5.7E+07
0.9 - 1.35	1.5E+07

Table 12.2-8
Spent Fuel Pool Cooling System Component
Source Strengths and Specific Activity

A. Demineralizer (75 cubic feet of resin)	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	3.0E+06
0.9 - 1.35	4.7E+06
Nuclide	**Activity ($\mu Ci/cm^3$)**
Co-58	5.8E+01
Co-60	3.4E+01
B. Filters	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	1.1E+07
0.9 - 1.35	3.0E+06
Source Dimensions (inches)	**Source Composition (volume percent)**
Radius = 3.375	Air – 67
Length = 19	Water – 33

Table 12.2-9 (Sheet 1 of 7)
Liquid Radwaste System
Component Source Terms

A. Reactor coolant drain tank		
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**	
	Liquid Phase (450 gallons)	**Gas Space (60 cubic feet)**
0.000-0.020	1.4E+03	1.4E+02
0.020-0.030	1.6E+03	2.0E+02
0.030-0.045	6.9E+04	7.6E+03
0.045-0.070	3.2E+02	6.1E-05
0.070-0.100	1.4E+05	1.5E+04
0.100-0.150	1.1E+03	8.4E-02
0.150-0.300	4.3E+04	4.2E+02
0.300-0.450	2.3E+04	2.6E+01
0.450-0.700	1.0E+05	4.1E+03
0.700-1.000	1.3E+05	2.3E+01
1.000-1.500	1.2E+05	1.4E+01
1.500-2.000	5.9E+04	3.3E+01
2.000-2.500	9.3E+04	2.0E+02
2.500-3.000	1.5E+04	1.0E+01
3.000-4.000	2.6E+03	3.7E-01
4.000-6.000	5.0E+02	2.2E-03
6.000-8.000	-	-
8.000-11.000	-	-
Total	7.9E+05	2.8E+04

Table 12.2-9 (Sheet 2 of 7)
Liquid Radwaste System
Component Source Terms

A. Reactor coolant drain tank		
	Activity (µCi/cm^3)	
Nuclide	**Liquid Phase**	**Gas Space**
Mn-56	1.7E-01	-
I-132	9.3E-01	-
I-134	2.2E-01	-
Kr-87	4.7E-01	-
Kr-88	1.5E+00	-
I-131	7.1E-01	-
I-133	1.3E+00	-
Xe-133	1.2E+02	-
I-135	7.8E-01	-
Xe-135	3.5E+00	-
Xe-138	2.4E-01	-
Rb-88	1.5E+00	-
Cs-134	6.9E-01	-
Cs-136	1.0E+00	-
Ba-137m	4.7E-01	-
Cs-138	3.7E-01	-
Kr-85	-	5.0E+01
Kr-87	-	5.4E-04
Kr-88	-	3.8E-03
Xe-131m	-	3.4E-01
Xe-133m	-	8.0E-02
Xe-133	-	1.4E+01
Xe-135	-	2.8E-02

Notes:
The liquid activities listed are 99% of the total source strength.
The vapor activities listed are essentially 100% of the source strength.

Table 12.2-9 (Sheet 3 of 7)
Liquid Radwaste System
Component Source Terms

B. Effluent tank (28,000 gal) and waste holdup tank (15,000 gal)			
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**		
	Effluent Tank		**Holdup Tank Liquid**
	Liquid Phase	**Vapor Phase**	
0.000-0.020	1.9E+01	3.4E+03	1.9E+01
0.020-0.030	1.6E+01	4.0E+03	1.2E+01
0.030-0.045	2.7E+02	1.8E+05	2.7E+02
0.045-0.070	3.2E+02	4.5E-05	3.2E+02
0.070-0.100	2.2E+02	3.7E+05	2.1E+02
0.100-0.150	1.0E+03	3.6E+01	9.8E+02
0.150-0.300	3.4E+03	7.6E+04	3.3E+03
0.300-0.450	7.5E+03	5.4E+03	7.0E+03
0.450-0.700	4.1E+04	6.1E+03	3.8E+04
0.700-1.000	6.5E+04	1.0E+04	5.6E+04
1.000-1.500	7.1E+04	6.5E+03	4.7E+04
1.500-2.000	2.5E+04	1.5E+04	2.1E+03
2.000-2.500	6.5E+03	9.1E+04	5.7E+02
2.500-3.000	7.3E+03	4.0E+03	1.4E+02
3.000-4.000	1.5E+03	1.3E+02	1.7E+01
4.000-6.000	4.5E+02	9.8E-18	-
6.000-8.000	-	-	-
8.000-11.000	-	-	-
Total	2.3E+05	7.7E+05	1.6E+05

Table 12.2-9 (Sheet 4 of 7)
Liquid Radwaste System
Component Source Terms

B. Effluent tank (28,000 gal) and waste holdup tank (15,000 gal)			
	Activity (µCi/cm^3)		
	Effluent Tank		
Nuclides	**Liquid Phase**	**Vapor Phase**	**Holdup Tank Liquid**
Mn-56	1.7E-02	–	1.7E-02
I-131	-	–	7.1E-02
I-132	9.3E-02	–	9.3E-02
I-133	1.3E-01	–	1.3E-01
I-134	2.2E-02	–	–
I-135	7.8E-02	–	7.8E-02
Rb-88	1.5E+00	–	–
Rb-89	6.9E-02	–	–
Cs-134	6.9E-01	–	6.9E-01
Cs-136	1.0E+00	–	1.0E+00
Cs-137	5.0E-01	–	5.0E-01
Ba-137m	5.0E-01	–	5.0E-01
Sr-89	–	–	1.1E-04
Cs-138	3.7E-01	–	–
Mo-99	2.1E-01	–	2.1E-01
Kr-85m	–	1.3E+00	–
Kr-85	–	8.0E+00	–
Kr-87	–	2.1E-01	–
Kr-88	–	1.8E+00	–
Xe-131m	–	3.6E+00	–
Xe-133m	–	4.3E+00	–
Xe-133	–	3.3E+02	–
Xe-135	–	7.3E+00	–

Notes:
The liquid activities listed are 99% of the total source strength.
The vapor activities listed are essentially 100% of the source strength.

Table 12.2-9 (Sheet 5 of 7)
Liquid Radwaste System
Component Source Terms

C. Chemical waste tank (7,700 gal)	
Energy Group (Mev/gamma)	**Source Strength (Mev/gram-sec)**
0-0.02	6.3E-01
0.02-0.03	1.2E-01
0.03-0.045	2.5E+01
0.045-0.07	3.1E-04
0.07-0.1	5.3E-03
0.1-0.15	3.2E-02
0.15-0.3	3.8E+00
0.3-0.45	5.6E+00
0.45-0.7	1.6E+04
0.7-1.0	9.7E+03
1.0-1.5	1.0E+03
1.5-2.0	1.5E+02
2.0-2.5	3.2E+01
2.5-3.0	5.3E+01
3.0-4.0	6.9E+00
4.0-6.0	–
6.0-8.0	–
8.0-11.00	–
Total	2.7E+04
Nuclide	**Activity (μCi/gram)**
Co-58	1.9E-03
Cs-134	3.4E-01
Cs-137	2.5E-01
Te-127m	3.8E-04
Ba-137m	2.3E-01
Ce-144	5.8E-05

Table 12.2-9 (Sheet 6 of 7)
Liquid Radwaste System
Component Source Terms

D. Waste ion exchanger and charcoal deep bed filter vessel[(a)]	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0-0.02	1.2E+03
0.02-0.03	5.5E+03
0.03-0.045	5.1E+04
0.045-0.07	5.8E+04
0.07-0.1	4.8E+04
0.1-0.15	3.4E+03
0.15-0.3	6.7E+05
0.3-0.45	2.5E+06
0.45-0.7	9.1E+06
0.7-1.0	1.1E+07
1.0-1.5	9.3E+06
1.5-2.0	6.0E+05
2.0-2.5	1.4E+05
2.5-3.0	2.9E+04
3.0-4.0	3.5E+03
4.0-6.0	–
6.0-8.0	–
8.0-11.00	–
Total	3.3E+07

Note:
a. Source term for the charcoal deep bed filter vessel is based on operation charged with resin instead of charcoal since this is the most conservative mode of operation for source terms.

Table 12.2-9 (Sheet 7 of 7)
Liquid Radwaste System
Component Source Terms

D. Waste ion exchanger and charcoal deep bed filter vessel[a]	
Nuclide	**Activity (μCi/cm^3)**
Mn-56	3.4E+00
I-131	1.2E+02
I-132	1.7E+01
I-133	1.1E+02
I-135	2.8E+01
Cs-134	1.4E+02
Cs-136	1.9E+02
Cs-137	9.9E+01
Ba-137m	9.4E+01
E. Waste prefilter and waste after filter	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.4 - 0.9	1.1E+07
0.9 - 1.35	3.0E+06
Total	1.4E+07
Source Dimensions (inches)	**Source Composition (volume percent)**
Radius = 3.375	Air - 67
Length = 19	Water - 33

Note:

a. Source term for the charcoal deep bed filter vessel is based on operation charged with resin instead of charcoal since this is the most conservative mode of operation for source terms.

Table 12.2-10 (Sheet 1 of 4)
Gaseous Radwaste System
Component Source Terms

A. Gas cooler and moisture separator	
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**
0.000-0.020	1.1E+03
0.020-0.030	1.2E+03
0.030-0.045	5.4E+04
0.045-0.070	2.3E-01
0.070-0.100	1.1E+05
0.100-0.150	2.2E+01
0.150-0.300	3.1E+04
0.300-0.450	5.6E+03
0.450-0.700	5.3E+03
0.700-1.000	7.8E+03
1.000-1.500	5.2E+03
1.500-2.000	1.4E+04
2.000-2.500	6.5E+04
2.500-3.000	5.9E+03
3.000-4.000	5.9E+02
4.000-6.000	3.6E+01
6.000-8.000	-
8.000-11.000	-
Total	3.1E+05

Table 12.2-10 (Sheet 2 of 4)
Gaseous Radwaste System
Component Source Terms

A. Gas cooler and moisture separator	
Nuclide	**Activity (µCi/cm^3)**
Kr-85m	6.6E-01
Kr-87	3.8E-01
Kr-88	1.2E+00
Xe-133m	1.3E+00
Xe-133	9.8E+01
Xe-135m	1.4E-01
Xe-135	2.8E+00
Xe-138	1.9E-01

Note:
The activities listed are 99% of the total source strength.

Table 12.2-10 (Sheet 3 of 4)
Gaseous Radwaste System
Component Source Terms

B. Charcoal guard and delay beds (8 ft^3 guard bed and 80 ft^3 delay beds)		
Energy Group (Mev/gamma)	**Source Strength (Mev/cm^3-sec)**	
	Guard Bed	**Delay Beds**
0.000-0.020	1.2E+06	1.1E+05
0.020-0.030	1.4E+06	1.3E+05
0.030-0.045	6.2E+07	5.8E+06
0.045-0.070	9.2E+00	8.6E-01
0.070-0.100	1.3E+08	1.2E+07
0.100-0.150	1.2E+04	1.2E+03
0.150-0.300	4.1E+07	3.8E+06
0.300-0.450	2.1E+06	2.0E+05
0.450-0.700	3.4E+06	3.1E+05
0.700-1.000	3.3E+06	3.1E+05
1.000-1.500	2.1E+06	1.9E+05
1.500-2.000	4.9E+06	4.6E+05
2.000-2.500	2.9E+07	2.7E+06
2.500-3.000	1.5E+06	1.4E+05
3.000-4.000	5.5E+04	5.1E+03
4.000-6.000	3.2E+02	3.0E+01
6.000-8.000	–	–
8.000-11.000	–	–
Total	2.8E+08	2.6E+07

Table 12.2-10 (Sheet 4 of 4)
Gaseous Radwaste System
Component Source Terms

B. Charcoal guard and delay beds (8 ft^3 guard bed and 80 ft^3 delay beds)		
	Activity (µCi/cm3)	
Nuclide	**Guard Bed**	**Delay Beds**
Kr-85m	5.0E+02	4.6E+01
Kr-85	2.7E+03	2.5E+02
Kr-87	8.0E+01	7.4E+00
Kr-88	5.5E+02	5.2E+01
Xe-131m	1.2E+03	1.1E+02
Xe-133m	1.5E+03	1.4E+02
Xe-133	1.1E+05	1.1E+04
Xe-135	4.3E+03	4.0E+02
Xe-138	7.7E+00	7.2E-01

Note:
The activities listed are essentially 100% of the total source strength.

Table 12.2-11 (Sheet 1 of 2)
Spent Demineralizer Resin
Source Strengths and Specific Activities

Energy Group (Mev/gamma)	Spent Resin Source Strength (Mev/cm^3-sec)	
	Initial	After 30 Days
0-0.02	1.7E+04	1.3E+04
0.02-0.03	1.1E+05	8.1E+03
0.03-0.045	3.5E+05	3.0E+05
0.045-0.07	4.8E+04	9.7E+03
0.07-0.1	2.2E+05	2.1E+04
0.1-0.15	4.8E+03	5.7E+02
0.15-0.3	2.2E+06	2.6E+05
0.3-0.45	2.8E+07	2.3E+06
0.45-0.7	1.8E+08	1.6E+08
0.7-1.0	9.9E+07	8.8E+07
1.0-1.5	2.8E+07	1.7E+07
1.5-2.0	4.2E+06	1.9E+06
2.0-2.5	8.7E+05	3.8E+05
2.5-3.0	7.4E+05	6.4E+05
3.0-4.0	9.9E+04	8.3E+04
4.0-6.0	2.7E+03	–
6.0-8.0	–	–
8.0-11.00	–	–
Total	3.4E+08	2.7E+08

Table 12.2-11 (Sheet 2 of 2)
Spent Demineralizer Resin
Source Strengths and Specific Activities

Nuclide	Spent Resin Activity (μCi/cm^3)	
	Initial	**After 30 Days**
Mn-54	6.2E+01	5.8E+01
Mn-56	1.3E+01	-
Co-58	7.2E+01	5.4E+01
Co-60	7.8E+01	7.7E+01
I-131	2.4E+03	1.8E+02
I-132	4.3E+01	–
I-134	3.9E+00	–
I-135	9.9E+01	–
Rb-88	8.9E+00	–
Cs-134	3.0E+03	3.0E+03
Cs-136	1.6E+02	3.1E+01
Cs-137	3.0E+03	3.0E+03
Ba-137m	2.8E+03	2.8E+03

Table 12.2-12 (Sheet 1 of 2)
Normal Residual Heat Removal System
Source Strengths and Specific Activities

Energy Group (Mev/gamma)	Source Strength (Mev/gram-sec)	
	4 Hours After Shutdown	8 Hours After Shutdown
0.000 -0.020	1.3E+03	1.2E+03
0.020 -0.030	1.5E+03	1.4E+03
0.030 -0.045	6.7E+04	6.6E+04
0.045 -0.070	2.1E+02	1.4E+02
0.070 -0.100	1.4E+05	1.3E+05
0.100 -0.150	9.6E+02	9.2E+02
0.150 -0.300	3.0E+04	2.2E+04
0.300 -0.450	1.1E+04	7.2E+03
0.450 -0.700	4.6E+04	2.7E+04
0.700 -1.000	3.5E+05	3.3E+05
1.000 -1.500	5.6E+04	3.5E+04
1.500 -2.000	2.0E+04	7.8E+03
2.000 -2.500	3.0E+04	1.1E+04
2.500 -3.000	2.6E+03	7.9E+02
3.000 -4.000	4.7E+02	1.7E+02
4.000 -6.000	1.7E+02	6.2E+01
6.000 -8.000	–	–
8.000-11.000	–	–
Total	7.6E+05	6.4E+05

Table 12.2-12 (Sheet 2 of 2)
Normal Residual Heat Removal System
Source Strengths and Specific Activities

Nuclide	Activity (µCi/gram)	
	4 Hours After Shutdown	**8 Hours After Shutdown**
Kr-87	5.3E-02	6.0E-03
Kr-88	5.5E-01	2.1E-01
Xe-131m	1.3E+00	1.3E+00
Xe-133m	1.6E+00	1.5E+00
Xe-133	1.2E+02	1.2E+02
Xe-135	2.7E+00	2.0E+00
I-131	4.7E-01	3.1E-01
I-132	1.9E-01	4.0E-02
I-133	7.5E-01	4.4E-01
I-135	3.4E-01	1.5E-01
Rb-88	5.9E-01	2.2E-01
Cs-134	4.6E-01	3.1E-01
Cs-136	6.8E-01	4.5E-01
Tc-99m	1.9E-01	1.8E-01
Ba-137m	3.1E-01	2.1E-01
Co-58	1.0E+01	1.0E+01
Co-60	1.0E-01	1.0E-01

Table 12.2-13
Core Average and Spent Fuel Neutron Source Strengths at Various Times After Shutdown

Time After Shutdown	Core Average (n/watt-sec)	Spent Fuel (n/watt-sec)
12 hours	22	77
24 hours	22	77
100 hours	22	76
1 week	22	76
1 month	21	75
3 months	20	71
6 months	18	68
1 year	16	63
5 years	13	51

Table 12.2-14 (Sheet 1 of 2)
Spent Fuel Gamma Ray Source Strengths

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)				
	12 Hours	**24 Hours**	**100 Hours**	**1 Week**	**1 Month**
0-0.02	2.3E+08	2.0E+08	8.4E+07	4.1E+07	1.7E+06
0.02-0.03	4.3E+07	3.7E+07	2.2E+07	1.6E+07	4.1E+06
0.03-0.045	1.1E+08	1.0E+08	6.0E+07	4.4E+07	1.5E+07
0.045-0.07	8.5E+07	7.6E+07	4.3E+07	2.9E+07	2.9E+06
0.07-0.1	4.4E+08	3.9E+08	1.9E+08	1.1E+08	9.6E+06
0.1-0.15	1.8E+09	1.6E+09	7.4E+08	4.0E+08	8.2E+07
0.15-0.3	2.5E+09	2.1E+09	8.6E+08	4.2E+08	1.6E+07
0.3-0.45	1.0E+09	8.9E+08	5.4E+08	3.9E+08	7.5E+07
0.45-0.7	5.4E+09	4.5E+09	2.7E+09	2.2E+09	1.2E+09
0.7-1.0	6.7E+09	5.9E+09	4.2E+09	3.6E+09	2.5E+09
1.0-1.5	2.3E+09	1.5E+09	7.6E+08	5.6E+08	1.8E+08
1.5-2.0	3.3E+09	3.0E+09	2.4E+09	2.1E+09	6.1E+08
2.0-2.5	3.9E+08	3.1E+08	2.4E+08	2.0E+08	7.9E+07
2.5-3.0	1.8E+08	1.7E+08	1.4E+08	1.2E+08	3.5E+07
3.0-4.0	4.6E+06	1.8E+06	1.5E+06	1.3E+06	4.2E+05
4.0-6.0	2.5E+05	1.3E+04	1.6E+01	1.6E+01	1.6E+01
6.0-8.0	2.6E+00	2.6E+00	2.6E+00	2.6E+00	2.6E+00
8.0-11.00	4.1E-01	4.1E-01	4.1E-01	4.1E-01	4.0E-01

Table 12.2-14 (Sheet 2 of 2)
Spent Fuel Gamma Ray Source Strengths

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/watt-sec)			
	3 Months	**6 Months**	**1 Year**	**5 Years**
0-0.02	3.8E+05	2.4E+05	1.6E+05	6.4E+04
0.02-0.03	1.2E+06	5.2E+05	3.2E+05	1.1E+05
0.03-0.045	6.8E+06	4.6E+06	3.1E+06	6.9E+05
0.045-0.07	3.4E+05	2.2E+05	1.7E+05	9.6E+04
0.07-0.1	2.7E+06	2.1E+06	1.4E+06	2.5E+05
0.1-0.15	3.3E+07	1.6E+07	9.1E+06	1.2E+06
0.15-0.3	1.9E+06	9.6E+05	6.6E+05	3.7E+05
0.3-0.45	6.8E+06	3.7E+06	2.9E+06	9.7E+05
0.45-0.7	7.3E+08	5.3E+08	4.1E+08	1.6E+08
0.7-1.0	1.5E+09	7.9E+08	3.2E+08	6.8E+07
1.0-1.5	7.7E+07	6.2E+07	5.0E+07	1.7E+07
1.5-2.0	3.1E+07	5.2E+06	3.3E+06	5.6E+05
2.0-2.5	2.3E+07	1.6E+07	1.0E+07	3.2E+05
2.5-3.0	1.6E+06	2.4E+05	1.6E+05	1.0E+04
3.0-4.0	6.6E+04	4.4E+04	3.1E+04	2.0E+03
4.0-6.0	1.5E+01	1.4E+01	1.3E+01	1.1E+01
6.0-8.0	2.5E+00	2.3E+00	2.2E+00	1.8E+00
8.0-11.00	3.8E-01	3.6E-01	3.4E-01	2.8E-01

Table 12.2-15
Irradiated Control Rod and Gray Rod Source Strengths

Energy Group (Mev/gamma)	Silver-Indium-Cadmium Control Rod Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	1 Week	1 Month
0.20 - 0.40	2.3E+08	2.3E+08	2.2E+08
0.40 - 0.90	1.1E+12	1.1E+12	1.0E+12
0.90 - 1.35	2.0E+11	1.9E+11	1.8E+11
1.35 - 1.80	3.7E+11	3.7E+11	3.4E+11
(Mev/gamma)	6 Months	1 Year	5 Years
0.20 - 0.40	1.4E+08	8.5E+07	1.5E+06
0.40 - 0.90	6.6E+11	4.0E+11	7.1E+09
0.90 - 1.35	1.2E+11	7.2E+10	1.3E+09
1.35 - 1.80	2.3E+11	1.4E+11	2.5E+09

Note: The absorber cross-sectional area is 0.589 and 0.521 square centimeters for the upper and lower portions of the rods, respectively, and the absorber material density is 10.2 grams per cubic centimeter.

Energy Group (Mev/gamma)	Tungsten Gray Rod Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	1 Week	1 Month
0.00 - 0.25	6.5E+11	2.7E+11	8.3E+10
0.25 - 0.40	1.2E+11	5.2E+10	4.1E+10
0.40 - 0.90	5.1E+11	1.3E+11	1.1E+11
0.90 - 1.35	3.7E+10	7.9E+09	7.4E+09
1.35 - 1.80	9.2E+09	4.5E+07	8.7E+06
1.80 - 2.20	2.0E+09	7.2E+06	4.7E+05
(Mev/gamma)	6 Months	1 Year	5 Years
0.00 - 0.25	8.1E+09	1.7E+09	1.9E+06
0.25 - 0.40	2.1E+10	9.8E+09	2.6E+07
0.40 - 0.90	5.5E+10	2.5E+10	6.8E+07
0.90 - 1.35	5.5E+09	4.3E+09	2.1E+09
1.35 - 1.80	3.2E+06	1.0E+06	2.6E+03
1.80 - 2.20	1.3E+05	5.0E+04	2.1E+04

Note: The absorber cross-sectional area is 0.197 square centimeters and the absorber material density is 19.2 grams per cubic centimeter.

Table 12.2-15 (continued)
Irradiated Control Rod and Gray Rod Source Strengths

Energy Group (Mev/gamma)	Gray Rod Sleeve (Nickel Alloy 718) Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	**1 Week**	**1 Month**
0.00 - 0.25	8.2E+09	2.9E+09	1.3E+09
0.25 - 0.40	1.3E+10	1.1E+10	5.9E+09
0.40 - 0.90	4.1E+11	3.6E+11	2.4E+11
0.90 - 1.35	2.1E+12	2.1E+12	2.1E+12
1.35 - 1.80	8.0E+08	5.6E+08	4.4E+08
1.80 - 2.20	1.9E+08	2.0E+07	2.0E+07
2.20 - 2.60	7.9E+06	4.2E+04	4.1E+04
2.60 - 3.00	1.0E+08	1.0E+08	-

(Mev/gamma)	6 Months	1 Year	5 Years
0.00 - 0.25	9.4E+08	8.4E+08	4.9E+08
0.25 - 0.40	1.8E+08	3.1E+07	1.6E+07
0.40 - 0.90	2.5E+10	5.1E+09	4.5E+08
0.90 - 1.35	2.0E+12	1.8E+12	1.1E+12
1.35 - 1.80	1.0E+08	1.6E+07	4.4E+02
1.80 - 2.20	1.9E+07	1.8E+07	1.0E+07
2.20 - 2.60	3.9E+04	3.7E+04	2.2E+04
2.60 - 3.00	-	-	-

Note: The absorber cross-sectional area is 0.29 square centimeters and the absorber material density is 8.2 grams per cubic centimeter.

Table 12.2-16
Irradiated SB-BE Secondary Source Rod
Gamma Ray Source Strengths

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm^3-sec)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.020 - 0.035	5.9E+10	2.4E+10	1.1E+10	2.5E+09	5.7E+08	5.5E+07
0.035 - 0.05	4.2E+10	1.7E+10	7.9E+09	1.4E+09	1.8E+08	6.0E+06
0.05 - 0.075	5.7E+10	2.3E+10	1.1E+10	1.9E+09	2.2E+08	6.0E+05
0.075 - 0.125	1.0E+11	4.0E+10	1.9E+10	3.3E+09	4.0E+08	1.7E+06
0.125 - 0.175	7.3E+10	3.8E+10	2.4E+10	7.8E+09	2.3E+09	1.7E+07
0.175 - 0.250	6.6E+10	2.6E+10	1.2E+10	2.2E+09	3.1E+08	2.0E+07
0.250 - 0.40	1.4E+11	5.8E+10	2.9E+10	5.2E+09	6.6E+08	2.0E+07
0.40 - 0.90	1.3E+13	7.1E+12	4.4E+12	7.8E+11	9.5E+10	1.0E+09
0.90 - 1.35	8.6E+11	5.9E+11	4.0E+11	7.2E+10	8.5E+09	4.3E+02
1.35 - 1.80	7.3E+12	6.8E+12	5.2E+12	9.2E+11	1.1E+11	5.5E+03
1.80 - 2.20	9.8E+11	9.2E+11	7.0E+11	1.3E+11	1.5E+10	7.4E+02
2.20 - 2.60	8.1E+09	7.6E+09	5.8E+09	1.0E+09	1.2E+08	6.1E+00
2.60 - 3.00	9.8E+08	9.2E+08	7.0E+08	1.3E+08	1.5E+07	7.4E-01

Notes:

The Sb-Be material density is 3.56 g/cm^3.

The secondary source rod cross-sectional area is 0.432 cm^2 per rod.

The average neutron energy is 30 kev.

Table 12.2-17
Irradiated SB-BE Secondary Source Rod
Neutron Source Strengths

Time After Shutdown	Sb-124 Concentration (curies/cm^3)	Neutron Source (N/cm^3-sec)
1 day	2.2E+02	4.4E+08
1 week	2.0E+02	4.1E+08
1 month	1.6E+02	3.1E+08
6 months	2.8E+01	5.6E+07
1 year	3.3E+00	6.6E+06
5 years	1.6E-07	3.3E-01

Notes:
The Sb-Be material density is 3.56 g/cm^3.
The secondary source rod cross-sectional area is 0.432 cm^2 per rod.
The average neutron energy is 30 kev.

Table 12.2-18
Irradiated Stainless Steel Source Strengths
(0.05 Weight Percent Cobalt)

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm^3-sec)		
	1 Day	1 Week	1 Month
0.00 - 0.25	4.7E+08	4.2E+08	3.2E+08
0.25 - 0.40	1.1E+10	9.6E+09	5.4E+09
0.40 - 0.90	3.7E+10	3.5E+10	3.2E+10
0.90 - 1.35	2.3E+11	2.2E+11	2.1E+11
1.35 - 1.80	3.6E+08	1.3E+08	1.0E+08
1.80 - 2.20			
2.20 - 2.60			
(Mev/gamma)	6 Months	1 Year	5 Years
0.00 - 0.25	1.1E+08	7.8E+07	4.4E+07
0.25 - 0.40	1.3E+08	4.2E+06	1.5E+06
0.40 - 0.90	1.8E+10	1.1E+10	4.3E+08
0.90 - 1.35	1.8E+11	1.6E+11	9.7E+10
1.35 - 1.80	2.1E+07	3.2E+06	4.0E+01
1.80 - 2.20	1.7E+06	1.6E+06	9.3E+05
2.20 - 2.60	3.5E+10	3.3E+03	1.9E+03

Notes:

The various cross-section areas per rod are as follows:

– Ag-In-Cd control rod cladding - 0.136 cm^2

– Sb-Be secondary source rod cladding - 0.136 cm^2 (outer) and 0.118 cm^2 (inner)

– Gray rod cladding - 0.184 cm^2

Table 12.2-19
Irradiated Flux Thimble
Source Strengths

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm^3-sec)			
	12 Hours	**1 Day**	**70 Hours**	**1 Week**
<0.15	4.0E+09	3.2E+09	2.2E+09	1.2E+09
0.15 - 0.45	7.9E+09	7.2E+09	6.4E+09	5.4E+09
0.45 - 1.0	4.2E+10	2.4E+10	2.2E+10	2.0E+10
1.0 - 1.5	1.2E+11	1.2E+11	1.2E+11	1.2E+11
1.5 - 2.0	1.0E+10	4.8E+08	7.5E+07	7.2E+07
2.0 - 2.5	6.3E+09	2.5E+08	9.9E+05	9.8E+05
2.5 - 3.0	1.1E+09	4.3E+07	1.2E+03	1.0E+03
3.0 - 4.0	1.2E+08	4.7E+06	2.0E+01	–
(MeV/gamma)	**1 Month**	**6 Months**	**1 Year**	**5 Years**
<0.15	5.5E+08	3.6E+08	3.1E+08	1.2E+08
0.15 - 0.45	2.9E+09	7.5E+07	5.1E+06	2.1E+06
0.45 - 1.0	1.7E+10	9.3E+09	5.5E+09	2.1E+08
1.0 - 1.5	1.1E+11	9.7E+10	9.0E+10	5.3E+10
1.5 - 2.0	5.7E+07	1.3E+07	2.2E+06	1.7E+00
2.0 - 2.5	9.8E+05	9.2E+05	8.6E+05	5.1E+05
2.5 - 3.0	1.0E+03	9.6E+02	9.0E+02	5.3E+02
3.0 - 4.0	–	–	–	–

Note:
The flux thimble cross-sectional area is 0.682 cm^2.

Table 12.2-20
Core Melt Accident Source Strengths
in Containment Atmosphere as a Function of Time

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/watt-sec)				
	10 Min.	**40 Min**	**1.97 Hours**	**3.97 Hours**	**5.97 Hours**
<0.15	3.9E+06	6.1E+06	1.2E+08	1.3E+08	1.4E+08
0.15 - 0.45	3.8E+07	4.8E+07	5.1E+08	5.9E+08	6.6E+08
0.45 - 1.0	3.2E+08	4.2E+08	2.6E+09	3.1E+09	2.8E+09
1.0 - 1.5	2.2E+08	2.6E+08	1.2E+09	1.4E+09	1.2E+09
1.5 - 2.0	7.8E+07	9.4E+07	7.1E+08	7.3E+08	5.4E+08
2.0 - 2.5	8.6E+07	9.0E+07	8.2E+08	6.6E+08	3.5E+08
2.5 - 3.0	3.0E+07	2.7E+07	1.3E+08	8.8E+07	2.9E+07
3.0 - 4.0	1.1E+07	5.6E+06	1.9E+07	1.3E+07	4.9E+06
4.0 - 6.0	3.3E+06	4.2E+05	3.9E+06	3.4E+06	1.7E+06
6.0 - 11.0	-	-	1.7E-04	2.3E-03	4.4E-03
Beta	4.2E+08	4.3E+08	3.5E+09	3.6E+09	3.0E+09
(Mev/gamma)	**11.97 Hours**	**1 Day**	**1 Week**	**1 Month**	**1 Year**
<0.15	1.4E+08	1.3E+08	5.8E+07	4.2E+06	4.0E+05
0.15 - 0.45	6.0E+08	4.7E+08	1.9E+08	2.8E+07	5.5E+05
0.45 - 1.0	2.3E+09	1.7E+09	6.8E+08	3.4E+08	2.2E+08
1.0 - 1.5	7.4E+08	3.5E+08	7.8E+07	2.2E+07	8.3E+06
1.5 - 2.0	3.0E+08	1.8E+08	2.5E+08	7.7E+07	1.3E+04
2.0 - 2.5	1.1E+08	2.9E+07	7.9E+06	1.3E+06	5.5E+04
2.5 - 3.0	8.8E+06	7.4E+06	1.5E+07	4.6E+06	5.2E+02
3.00 - 4.00	1.1E+06	1.2E+05	1.5E+05	4.8E+04	9.7E+01
4.0 - 6.0	3.9E+05	2.1E+04	2.4E-02	2.3E-02	1.8E-02
6.0 - 11.0	4.4E-03	4.4E-03	4.4E-03	4.3E-03	3.3E-03
Beta	2.2E+09	1.5E+09	6.2E+08	2.0E+08	6.3E+07

Note:
No release from core until 10 minutes after incident (see Subsection 15.6.5.3).

Table 12.2-21
Core Melt Accident Integrated Source Strengths in Containment Atmosphere

Energy Group (Mev/gamma)	Source Strength at Time After Release (Mev/watt)			
	10 Min.	**40 Min.**	**1.97 Hours**	**3.97 Hours**
<0.15	3.9E+06	8.8E+09	3.6E+11	1.3E+12
0.15 - 0.45	3.8E+07	7.8E+10	1.6E+12	5.9E+12
0.45 - 1.0	3.2E+08	6.6E+11	8.6E+12	3.0E+13
1.0 - 1.5	2.2E+08	4.3E+11	4.4E+12	1.4E+13
1.5 - 2.0	7.8E+07	1.6E+11	2.4E+12	7.6E+12
2.0 - 2.5	8.6E+07	1.6E+11	3.2E+12	7.9E+12
2.5 - 3.0	3.0E+07	5.1E+10	6.1E+11	1.3E+12
3.0 - 4.0	1.1E+07	1.4E+10	9.2E+10	1.9E+11
4.0 - 6.0	3.3E+06	2.5E+09	1.5E+10	3.9E+10
6.0 - 11.0	-	-	4.0E-01	1.7E+01
Beta	4.2E+08	7.6E+11	1.2E+13	3.8E+13

(Mev/gamma)	11.97 Hours	1 Day	1 Week	1 Month	1 Year
<0.15	5.3E+12	1.1E+13	5.7E+13	9.8E+13	1.4E+14
0.15 - 0.45	2.5E+13	4.9E+13	1.9E+14	3.6E+14	5.6E+14
0.45 - 1.0	1.2E+14	2.0E+14	7.2E+14	1.7E+15	9.7E+15
1.0 - 1.5	5.0E+13	7.2E+13	1.5E+14	2.4E+14	6.4E+14
1.5 - 2.0	2.4E+13	3.5E+13	1.6E+14	4.5E+14	7.1E+14
2.0 - 2.5	2.0E+13	2.2E+13	2.9E+13	3.7E+13	4.8E+13
2.5 - 3.0	2.4E+12	2.8E+12	9.6E+12	2.7E+13	4.2E+13
3.0 - 4.0	3.7E+11	3.9E+11	4.6E+11	6.5E+11	8.7E+11
4.0 - 6.0	9.5E+10	1.0E+11	1.0E+11	1.0E+11	1.0E+11
6.0 - 11.0	1.4E+02	3.4E+02	2.6E+03	1.0E+04	9.5E+04
Beta	1.3E+14	2.1E+14	6.7E+14	1.4E+15	4.9E+15

Note:
No release from core until 10 minutes after incident (see Subsection 15.6.5.3)

Table 12.2-22
Parameters and Assumptions Used for Calculating
Containment Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Reactor coolant leakage rate	30 lb/day
Time used to estimate equilibrium concentration	100 days
Containment free air volume	2.06E6 cu. ft
Flashing fraction	0.40
Fuel defects	0.250%
Reactor coolant tritium concentration	3.5 μCi/g
Normal operation purge flow rate	4,000 cfm
Normal operation purge duration	20 hrs/week
Shutdown purge flow rate	8,000 cfm

Table 12.2-23 (Sheet 1 of 3)
Containment Airborne Radioactivity Concentrations
(μCi/cm^3)

Isotope	Equilibrium Activity (no purge)	Maximum Activity (with normal purge)	Shutdown Activity (shutdown purge for 24 hrs)
Cr-51	1.0E-11	9.1E-12	2.1E-13
Mn-54	5.1E-12	4.6E-12	1.1E-13
Mn-56	0.0E+00	0.0E+00	0.0E+00
Fe-55	3.9E-12	3.5E-12	8.2E-14
Fe-59	9.7E-13	8.8E-13	2.0E-14
Co-58	1.5E-11	1.4E-11	3.2E-13
Co-60	1.7E-12	1.6E-12	3.6E-14
Br-83	0.0E+00	0.0E+00	0.0E+00
Br-84	3.4E-11	3.2E-11	2.5E-26
Br-85	0.0E+00	0.0E+00	0.0E+00
Kr-83m	0.0E+00	0.0E+00	0.0E+00
Kr-85m	1.3E-08	7.9E-09	2.2E-10
Kr-85	3.3E-05	4.1E-07	6.5E-08
Kr-87	3.4E-09	2.9E-09	1.5E-14
Kr-88	1.4E-08	1.0E-08	4.4E-11
Kr-89	0.0E+00	0.0E+00	0.0E+00
Rb-88	2.8E-10	2.7E-10	4.1E-36
Rb-89	0.0E+00	0.0E+00	0.0E+00
Sr-89	4.7E-13	4.2E-13	9.7E-15
Sr-90	3.9E-14	3.5E-14	8.2E-16
Sr-91	3.6E-12	3.2E-12	1.4E-14
Sr-92	0.0E+00	0.0E+00	0.0E+00
Y-90	4.6E-15	4.2E-15	7.6E-17
Y-91m	1.2E-12	1.1E-12	6.8E-23
Y-91	1.7E-14	1.6E-14	3.6E-16
Y-92	0.0E+00	0.0E+00	0.0E+00
Y-93	1.6E-11	1.4E-11	6.9E-14

Table 12.2-23 (Sheet 2 of 3)
Containment Airborne Radioactivity Concentrations
(μCi/cm^3)

Isotope	Equilibrium Activity (no purge)	Maximum Activity (with normal purge)	Shutdown Activity (shutdown purge for 24 hrs)
Zr-95	1.3E-12	1.2E-12	2.7E-14
Nb-95	9.3E-13	8.4E-13	1.9E-14
Mo-99	2.2E-11	2.0E-11	3.6E-13
Tc-99m	1.8E-11	1.6E-11	2.5E-14
Ru-103	2.5E-11	2.2E-11	5.1E-13
Ag-110m	4.3E-12	3.9E-12	9.0E-14
Te-127m	0.0E+00	0.0E+00	0.0E+00
Te-129m	6.2E-13	5.6E-13	1.3E-14
Te-129	7.1E-11	6.6E-11	1.1E-18
Te-131m	5.3E-12	4.8E-12	6.5E-14
Te-131	1.4E-11	1.4E-11	2.1E-30
Te-132	5.8E-12	5.2E-12	9.9E-14
TE-134	0.0E+00	0.0E+00	0.0E+00
I-129	0.0E+00	0.0E+00	0.0E+00
I-130	0.0E+00	0.0E+00	0.0E+00
I-131	2.9E-09	1.4E-10	8.8E-12
I-132	6.5E-09	6.9E-10	3.7E-14
I-133	8.9E-09	4.8E-10	1.5E-11
I-134	4.1E-09	8.6E-10	4.2E-19
I-135	1.4E-08	9.0E-10	5.2E-12
Xe-131m	4.4E-06	2.7E-07	4.8E-08
Xe-133m	6.9E-08	1.3E-08	3.1E-09
Xe-133	6.4E-06	7.1E-07	1.5E-07
Xe-135m	6.2E-10	6.0E-10	5.8E-37
Xe-135	1.4E-07	6.1E-08	8.0E-09
Xe-137	3.9E-11	3.9E-11	0.0E+00
Xe-138	5.0E-10	4.8E-10	0.0E+00

Table 12.2-23 (Sheet 3 of 3)
Containment Airborne Radioactivity Concentrations
(μCi/cm^3)

Isotope	Equilibrium Activity (no purge)	Maximum Activity (with normal purge)	Shutdown Activity (shutdown purge for 24 hrs)
Cs-134	2.3E-11	2.1E-11	4.8E-13
Cs-136	2.9E-12	2.6E-12	5.8E-14
Cs-137	3.1E-11	2.8E-11	6.5E-13
Cs-138	0.0E+00	0.0E+00	0.0E+00
Ba-137m	1.7E-12	1.7E-12	0.0E+00
Ba-140	4.3E-11	3.9E-11	8.5E-13
La-140	8.8E-11	8.0E-11	1.2E-12
Ce-141	5.1E-13	4.6E-13	1.0E-14
Ce-143	9.9E-12	9.0E-12	1.3E-13
Pr-143	1.2E-11	1.1E-11	2.3E-13
Ce-144	1.3E-11	1.2E-11	2.8E-13
Pr-144	3.9E-12	3.8E-12	0.0E+00
H-3	9.4E-06	1.2E-07	1.7E-09
Ar-41	3.8E-07	2.9E-07	2.2E-13
Na-24	1.7E-10	1.6E-10	1.2E-12
Zn-65	1.7E-12	1.5E-12	3.5E-14
Ru-106	2.9E-10	2.6E-10	6.1E-12
Rh-103m	1.4E-11	1.3E-11	7.3E-21
W-187	8.7E-12	7.9E-12	9.3E-14
Np-239	7.7E-12	7.0E-12	1.2E-13
Total	5.4E-05	1.9E-06	2.7E-07
Iodines	3.6E-08	3.1E-09	2.9E-11
Particulates	1.3E-09	1.2E-09	1.4E-11
Noble Gases	4.4E-05	1.8E-06	2.7E-07

Table 12.2-24
Parameters and Assumptions Used for Calculating
Fuel Handling Area Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Assumed fuel load	Full core offload
Ventilation flow through fuel handling area[(1)]	17,000 cfm[(2)]
Iodine filter efficiency	0
Particulate filter efficiency	0.99
Fuel handling area free air volume	200,000 ft^3
Fuel defects	0.25%
Time from shutdown to reactor vessel head removal	100 hours
Refueling time	10 days
Spent fuel pool purification flow rate	250 gpm
Decontamination factors of mixed-bed demineralizer for spent fuel pool purification system:	
Iodines	100
Cs and Rb	2
Others	50
Spent fuel pool temperature	120°F
Evaporation rate of spent fuel pool water	486 lbs/hr
Spent fuel pool tritium concentration	1.0 μCi/g

Notes:
1. This flow rate is defined as the sum of the fuel area exhaust fan flows minus the rail car bay/solid radwaste system exhaust flow.
2. This is the nominal expected ventilation flow rate. For conservatism, the calculated airborne radioactivity concentrations are based on a 10% lower flow rate.

Table 12.2-25 (Sheet 1 of 2)
Fuel Handling Area Airborne Radioactivity Concentrations[(1)]
(μCi/cm^3)

Isotope	Activity[(2)]
Cr-51	8.7E-12
Mn-54	4.8E-12
Fe-55	3.7E-12
Fe-59	8.7E-13
Co-58	1.4E-11
Co-60	1.6E-12
Kr-85m	7.6E-16
Kr-85	2.2E-10
Kr-88	2.7E-19
Sr-89	4.2E-12
Sr-90	3.7E-13
Sr-91	2.1E-14
Y-90	1.5E-14
Y-91	1.6E-13
Y-93	1.6E-13
Zr-95	1.2E-11
Nb-95	8.2E-12
Mo-99	7.1E-11
Tc-99m	1.4E-15
Ru-103	2.2E-10
Ag-110m	4.0E-11
Te-127m	2.9E-18
Te-129m	5.4E-12
Te-131m	4.9E-12
Te-132	2.3E-11
I-130	3.5E-18
I-131	1.0E-08
I-133	1.8E-09
I-135	2.3E-12
Xe-131m	1.7E-10
Xe-133m	3.1E-10
Xe-133	2.2E-08
Xe-135	4.2E-12
Cs-134	2.2E-10

Table 12.2-25 (Sheet 2 of 2)
Fuel Handling Area Airborne Radioactivity Concentrations[(1)]
(μCi/cm^3)

Isotope	**Activity[(2)]**
Cs-136	2.3E-11
Cs-137	3.0E-10
Ba-140	3.2E-10
La-140	1.5E-10
Ce-141	4.4E-12
Ce-143	1.1E-11
Pr-143	9.0E-11
Ce-144	1.3E-10
H-3	3.9E-06
Total (excluding tritium)	3.7E-08
Iodines	1.2E-08
Particulates	1.7E-09
Noble Gases	2.3E-08

Notes:
1. The maximum activity concentration is calculated to occur 2 hours after removal of the head, or 102 hours after shutdown in this case.
2. The following nuclides are expected to exist in the FHA at the time of maximum airborne concentrations with individual nuclide activity concentrations less that 1.0E-20 μCi/cm^3:
^{56}Mn, ^{83}Br, ^{84}Br, ^{85}Br, ^{83m}Kr, ^{87}Kr, ^{89}Kr, ^{88}Rb, ^{89}Rb, ^{92}Sr, ^{91m}Y, ^{92}Y, ^{129}Te, ^{131}Te, ^{134}Te, ^{129}I, ^{132}I, ^{134}I, ^{135m}Xe, ^{137}Xe, ^{138}Xe, ^{138}Cs, ^{137m}Ba, and ^{144}Pr.

Table 12.2-26
Parameters and Assumptions Used for Calculating Auxiliary Building Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Ventilation exhaust flow[(1)]	25,000 cfm[(2)]
Free air volume	365,400 ft^3
Primary coolant leakage to auxiliary building	20 lb/day
Flashing fraction	0.4
Primary coolant source term	See Table 11.1-2.
Fuel defects	0.25%

Notes:

1. This flow rate is defined as the sum of the aux/annex exhaust fan flow minus the annex building exhaust flow minus room 12555 (VES) and 12556 (containment access) exhaust flow.
2. This is the nominal expected ventilation flow rate. For conservatism, the calculated airborne radioactivity concentrations are based on a 10% lower flow rate.

Table 12.2-27 (Sheet 1 of 3)
Auxiliary Building Airborne Radioactivity Concentrations
(μCi/cm^3)

Isotope	Activity
Cr-51	5.1E-12
Mn-54	2.7E-12
Mn-56	6.7E-10
Fe-55	2.0E-12
Fe-59	5.1E-13
Co-58	7.5E-12
Co-60	8.7E-13
Br-83	1.3E-10
Br-84	6.6E-11
Br-85	7.8E-12
Kr-83m	1.8E-09
Kr-85m	8.3E-09
Kr-85	2.9E-08
Kr-87	4.7E-09
Kr-88	1.5E-08
Kr-89	3.5E-10
Rb-88	6.0E-09
Rb-89	2.7E-10
Sr-89	4.3E-12
Sr-90	1.9E-13
Sr-91	6.9E-12
Sr-92	1.6E-12
Y-90	5.0E-14
Y-91m	3.7E-12
Y-91	5.4E-13
Y-92	1.3E-12

Table 12.2-27 (Sheet 2 of 3)
Auxiliary Building Airborne Radioactivity Concentrations
($\mu Ci/cm^3$)

Isotope	Activity
Y-93	4.5E-13
Zr-95	6.3E-13
Nb-95	6.3E-13
Mo-99	8.4E-10
Tc-99m	7.7E-10
Ru-103	5.4E-13
Ag-110m	1.6E-12
Te-127m	3.0E-12
Te-129m	1.0E-11
Te-129	1.5E-11
Te-131m	2.7E-11
Te-131	1.7E-11
Te-132	3.1E-11
Te-134	4.2E-11
I-129	5.9E-17
I-130	4.2E-11
I-131	2.8E-09
I-132	3.7E-09
I-133	5.1E-09
I-134	8.6E-10
I-135	3.1E-09
Xe-131m	1.3E-08
Xe-133m	1.7E-08
Xe-133	1.2E-06

Table 12.2-27 (Sheet 3 of 3)
Auxiliary Building Airborne Radioactivity Concentrations
(µCi/cm³)

Isotope	**Activity**
Xe-135m	2.3E-09
Xe-135	3.5E-08
Xe-137	6.6E-10
Xe-138	2.4E-09
Cs-134	2.7E-09
Cs-136	4.0E-09
Cs-137	2.0E-09
Cs-138	1.5E-09
Ba-137m	1.9E-09
Ba-140	4.1E-12
La-140	1.0E-11
Ce-141	6.1E-13
Ce-143	5.6E-13
Pr-143	5.9E-13
Ce-144	4.6E-13
Pr-144	4.6E-13
H-3	1.4E-08
Total	1.4E-06
Iodines	1.6E-08
Particulates	2.1E-08
Noble Gases	1.4E-06

12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

Specific design features for maintaining personnel exposure as low as reasonably achievable (ALARA) are presented in this subsection. The design feature recommendations given in Regulatory Guide 8.8 are utilized to minimize exposures to personnel.

12.3.1.1 Plant Design Features for ALARA

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of Subsection 12.1.2 and are outlined in this subsection.

12.3.1.1.1 Common Equipment and Component Designs for ALARA

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

Reactor Vessel

The reactor vessel design includes an integrated head package which combines the head lifting rig, control and gray rod drive mechanism (CRDM/GRDM), lift columns, control rod drive mechanism cooling system and power and instrumentation cabling into an effective, one-package reactor vessel head design. Mounted directly on the reactor vessel head assembly, the system helps to minimize the time, manpower, and radiation exposure associated with head removal and replacement during refueling. Integral in the design is permanent shielding for reducing work area dose rates from the control rod drive mechanism drive shafts.

The combination thermocouple/incore detector system is not kept with head assembly during refueling, but instead remains with the upper internals. This allows the thermocouple/incore detector system to be shielded underwater in the refueling cavity during a majority of refueling operations, reducing dose rates around the head assembly.

The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the reinforced areas to provide a smooth transition, and pipe branch locations are selected to avoid interference from one branch to the next. Weld-to-pipe interfaces require a smooth, high quality finish.

Reactor Coolant Pumps

The sealless high-inertia reactor coolant pumps are designed to require infrequent maintenance and inspection. When maintenance or replacement is required, the pump can be removed and moved to a low radiation background work area using a specially provided pump removal cart.

Reactor Vessel Insulation

Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate removal of the insulation. Permanent identification markings of the sections of insulation are provided to accommodate rapid reinstallation.

Steam Generators

The steam generator incorporates many design features to facilitate maintenance and inspection in reduced radiation fields. The tube ends are designed to be flush with the tube sheet in the steam

generator channel head to eliminate a potential crud trap. The steam generator manways (entrance to channel head) are sized for easy entrance and exit of workers with protective clothing, and to facilitate the installation and removal of tooling.

The specification of low cobalt tubing material for the AP1000 steam generator design is an important feature of the design; not only in terms of reduced exposure relative to the steam generator, but to the total plant radiation source term. The cobalt content has been substantially reduced to 0.015 weight percent for the AP1000 steam generator tubing.

The steam generator design includes a sludge control system/mud drum which is designed to reduce the need for sludge lancing, and reduces tube and tube support degradation. Steam generator tube support plates design and full depth tubesheet expansion of tubes reduce corrosion and occupational exposure.

Reactor Coolant Pipe Connections

To minimize crud buildup in branch lines, piping connections to the reactor coolant loops are located on or above the horizontal centerline of the pipe wherever practicable.

Filters

Cartridges and filter bags that accumulate radioactivity are removed with semi-remote tools. Adequate space is provided to allow removing, and transporting the cartridge to storage and packaging areas as described in Section 11.4.

Liquid systems containing radioactive filters are provided with remote or semi-remote filter handling systems for the removal of spent radioactive filter elements from their housings and for their transfer to temporary storage or for packaging and shipment from the site for burial. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. The filter handling is designed to be simple, with a minimum of components susceptible to malfunction.

Demineralizers

Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks prior to processing and so that fresh resin can be loaded into the demineralizer remotely. The demineralizers and piping include provisions for being flushed with demineralized water. The system design prevents inadvertent flushing of the resin into the purification loop through the demineralizer inlet.

Pumps

Air operated diaphragm, sealless pumps or pumps with mechanical seals are used in radioactive systems to reduce leakage and seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal if necessary. Large pumps are selected with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping. Pumps in radioactive waste systems are provided with flanged or threaded connections for ease of removal.

Tanks and Sumps

Tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to control contamination within plant structures. Tanks containing radioactivity are fabricated from stainless steel, and sumps which can contain radioactive liquid are lined with stainless steel to facilitate decontamination.

Heat Exchangers

Vertical heat exchangers are designed so that the shell-to-tube sheet joint need not be broken for inspection. The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection and testing for leaks.

Heat exchangers are provided with corrosion-resistant tubes of stainless steel to reduce leakage. Impingement plates are provided and as necessary and tube side and shell side velocities are limited to minimize erosive effects. Wherever practicable, the radioactive fluid passes through the tube side of the heat exchanger.

Instruments

Instrument devices are located in low radiation zones away from radiation sources whenever practicable. Primary instrument devices, which for functional reasons are located in high radiation zones, are designed for easy removal to a lower radiation zone for calibration. Transmitters and readout devices are located in low radiation zones, such as corridors for servicing. Non-contact type instruments or self cleaning instruments are used whenever possible.

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

Integral radiation check sources for response verification for airborne radiation monitors and area radiation monitors are provided.

Chemical seals are provided on the instrument sensing lines on process piping, which may contain highly radioactive solids, to reduce the servicing time required to keep the lines free of solids. Instrument and sensing line connections are located slightly above the pipe midplane wherever practicable to minimize radioactive crud buildup.

Valves

To minimize personnel exposures from valve operations, motor-operated, air-operated, or other remotely actuated valves are used where justified by the activity levels and frequency of use. Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding is provided such that personnel need not enter a high radiation area for valve operation.

Wherever testing is required, valves of the bolted body-to-bonnet forging type are used to permit the use of ultrasonic testing in place of radiography. This facilitates inspection and maintenance time. Valves under 2 inches in diameter located in the piping carrying radioactive fluids in containment or carrying highly radioactive fluids outside containment are hermetically sealed valves to preclude radioactive releases to the environment, except for some passive core cooling system test header valves, where the valve stems are not normally pressurized, and hermetically sealed valves are impractical due to design constraints and availability limitations. The design of large-bore valves includes live-loaded packing and graphite packing materials to reduce the potential for steam leakage.

When equipment in high radiation areas is operated infrequently, those valves associated with normal processing are provided with remote-manual operators or reach rods. Other valve operations are performed with equipment in the shutdown mode.

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

Manually operated valves in the filter and demineralizer modules required for normal operation and shutdown are equipped with reach rods extending through the shield plates. Personnel do not enter the module during spent resin or cartridge transfer operations. The modules are designed to reduce personnel exposure during maintenance of components within or adjacent to the modules and to protect personnel who operate the valves.

Piping

The piping in pipe chases is designed for 60 year design objective with consideration for corrosion and operating environment. Pipe bends are used instead of elbows where practicable to reduce potential crud traps. Welds are made smooth to prevent crud traps from forming. Butt welds are used to the extent practicable. When radioactive piping is routed through areas where routine maintenance is required, pipe chases or distance separation are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection or maintenance requirements. Piping containing radioactive material is routed to minimize radiation exposure to plant personnel.

Floor and Equipment Drains

Floor drains and sloped floors are provided, where practicable, for rooms or cubicles containing serviceable components which contain radioactive liquids on plant elevations above the basemat. Floor drains are also provided on the Nuclear Island basemat in potentially contaminated areas, but sloping of floors around the drains is not required. If floor sloping is not incorporated on top of the Nuclear Island basemat, housekeeping activities shall be used to convey radioactive leakage to the available floor drains. When practicable, shielded pipe chases are used for radioactive pipes. Floor coatings are specified which simplify cleanup of spills. If a radioactive drain line must pass through a plant area requiring personnel access, shielding or distance separation is provided as necessary to maintain radiation levels consistent with the required access.

Lighting

Wherever practicable, multiple electric lights are provided for rooms containing highly radioactive components so that the burnout of a single lamp does not require entry and immediate replacement of the defective lamp since sufficient illumination is still available. Incandescent lights are provided inside containment. They require less time for servicing and, hence, the personnel exposure is reduced. The fluorescent lights which are used outside containment do not require frequent service due to the increased life of the tubes. High Pressure Sodium (HPS) underwater lighting, with appropriate barriers and protection to prevent contact with the water, or long life lighting not containing restricted materials, are used for underwater lighting in the fuel handling areas outside containment as they provide longer lamp life, longer service intervals and less personnel exposure.

Heating, Ventilation, and Air-Conditioning

The heating, ventilation, and air-conditioning (HVAC) system design facilitates replacement of the filter elements. Ventilation airflow is routed from areas of lower potential airborne contamination to areas of potentially higher contamination. In the radiologically-controlled area ventilation system (VAS) high airborne activity causes the exhaust air to be rerouted through HEPA and charcoal filters in the containment air filtration system (VFS).

Sample Stations

Proper shielding and ventilation are provided at the primary sample room to minimize personnel exposure during sampling. The counting room and laboratory facilities are described in Section 12.5. The fly ash used in the concrete for these areas is screened before use to confirm the radioactivity is less than the design limit.

Clean Services

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

Materials

Equipment specifications for components exposed to high temperature reactor coolant as part of normal plant operations contain limitations on the cobalt content of the base metal as given in Table 12.3-1. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations. Nickel-based alloys in the reactor coolant system (Co-58 is produced from activation of Ni-58) are similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the reactor coolant system is the inconel steam generator tubes.

General prohibitions on antimony and other low melting point metals are contained in Subsection 6.1.1. In addition, the reactor coolant pump mechanical design criteria prohibits antimony completely from the reactor coolant pump and its bearings.

Improved Head Closure System

The head closure system is designed to minimize the reactor head stud tensioning time.

12.3.1.1.2 Common Facility and Layout Designs for ALARA

This subsection describes the design features utilized for standard plant process and layout situations. These features are employed in conjunction with the general equipment described in Subsection 12.3.1.1.1 and include the features described in the following paragraphs.

Valve Modules

Selected valve modules are provided with shielded entrances for personnel protection. Floor drains are provided to control radioactive leakage. To facilitate decontamination, concrete surfaces are covered with a smooth surface coating which allows decontamination.

Piping

Pipes carrying radioactive materials are routed through controlled access areas properly zoned for that level of activity. Radioactive piping runs are analyzed to determine the potential radioactivity level and the dose rate 1 foot from the surface, in accordance with 10 CFR 20. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways or distance separation are provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Equipment compartments are used as pipeways for those pipes associated with equipment in the compartment.

When practicable, radioactive and nonradioactive piping are separated to minimize personnel exposure. Should maintenance be required, provision is made to isolate and drain radioactive piping and associated equipment.

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

Piping which carries resin slurries is run vertically and horizontal runs carrying spent resin are sloped toward the spent resin tanks, as much as practicable. Large radius bends are utilized instead of

elbows. Where sloped lines or large radius bends are impractical, adequate flush and drain capability is provided to prevent flow blockage and minimize crud traps.

The use of embedded pipes is minimized to the extent possible, consistent with maintaining radiation doses ALARA. To the extent possible, pipes are routed in accessible areas, such as dedicated pipe routing tunnels or pipe trenches, which provide good conditions for decommissioning.

Wall Penetrations

To minimize radiation streaming through wall penetrations, as many wall penetrations as practicable are located with offsets between the radioactive source and the normally accessible areas. If offsets are not practicable, penetrations are located as far as practicable above the floor elevation to reduce radiation exposure to personnel. If these two methods are not used, alternate means are employed, such as baffle shield walls or grouting the penetration annulus.

Contamination Control

Access control and traffic patterns are considered in the plant layout to reduce the spread of contamination. Equipment vents and drains from highly radioactive systems are piped directly to the collection system to minimize airborne and floor contamination. Welded piping systems are employed on radioactive systems to the maximum extent practicable to reduce system leakage and crud buildup at joints.

The number of passageways (doors) between the radiologically controlled area and the environment has been minimized. When such doors are incorporated, systems of drains and floor and exterior concrete sloping are used to prevent (potentially radioactive) fluid from the interior of the buildings from exiting the buildings, and also to prevent surface water from entering the buildings.

Decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of epoxy paints and suitable smooth-surface coatings to the concrete floors and walls. Sloping floors with floor drains are provided, where practicable, in potentially contaminated areas of the plant on plant elevations above the basemat. Floor drains are also provided on the Nuclear Island basemat in potentially contaminated areas, but sloping of floors around the drains is not required. If floor sloping is not incorporated on top of the Nuclear Island basemat, housekeeping activities shall be used to convey radioactive leakage to the available floor drains. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

In radiologically controlled areas where contamination is expected, radiation monitoring equipment is provided (Section 11.5). Those systems that become highly radioactive, such as the spent resin lines in the radwaste system, are provided with flush and drain connections.

Because of the potential for adsorption of contaminated fluids, the use of concrete block walls in the radiologically controlled areas of the plant is minimized. Where such walls are used, they are fully sealed at the ceiling or top of the block in order to prevent liquid incursion.

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

Equipment Layout

In those systems where process equipment is a major radiation source; pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. Control panels are located in low radiation zones.

Major components such as tanks, demineralizers, and filters in radioactive systems are located in shielded compartments insofar as practical. Labyrinth shields or shielding doors are provided for

compartments where radiation could stream or scatter to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as ion exchangers, filters and spent resin tanks), shielded compartments with hatch openings or removable shield walls are used. Equipment in nonradioactive systems that requires lubrication is located in low radiation zones. Wherever practicable, lubrication of equipment in high radiation areas is achieved with the use of tube-type extensions to reduce exposure during maintenance.

Exposure from routine in-plant inspection is controlled by locating, whenever practicable, inspection points in low-background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, space for portable shielding is provided.

Field Run Piping

Field run radioactive piping is minimized in the plant design. Radioactive process piping is routed dimensionally on orthographic drawings. Fabrication isometrics of radioactive process piping are reviewed to provide adequate shielding.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard area is regulated and controlled by posting of radiation signs, control of personnel, and use of alarms and locks (Section 12.5). During plant operation, access to radiologically restricted areas is through the access control area in the annex building.

Plant areas are categorized into radiation zones according to design basis radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Rooms, corridors, and pipeways are evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Each radiation zone defines the radiation level range expected in the zone. The radiation zone categories employed and zoning for each plant area under normal conditions is shown in Figure 12.3-1. The zoning for each plant area under accident conditions is shown in Figure 12.3-2. Radiation zones shown in the figures are based upon conservative design data. Actual in-plant zones and control of personnel access are based upon surveys conducted by the Combined License holder. Access control provisions for each plant area under normal expected conditions are shown in Figure 12.3-3. These provisions implement the requirements of 10 CFR 20 and utilize the alternative access control methods outlined in Regulatory Guide 8.38.

Figure 12.3-201, Figure 12.3-202, and Figure 12.3-203 replace Figure 12.3-1 (sheet 11), Figure 12.3-2 (sheet 11), and Figure 12.3-3 (sheet 11), respectively, to reflect the relocation of the Operations Support Center.

Based on actual operating plant data, ingress or egress of plant operating personnel to radiologically restricted areas is controlled and monitored as discussed in Subsection 12.3.5 such that radiation levels and exposures are within the limits prescribed in 10 CFR 20.

Posting of radiation signs, control of personnel access, and use of alarms and locks are discussed in Subsection 12.3.5.

12.3.2 Shielding

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

12.3.2.1 Design Objectives

The objective of the plant radiation shielding is to minimize personnel and population exposures, while maintaining a program of controlled personnel access to and occupancy of radiation areas. Radiation levels are within the requirements of 10 CFR 50 during design basis accidents and ALARA within the requirements of 10 CFR 20 during normal operation. Shielding and equipment layout and design are considered in providing confidence that exposures are kept ALARA during anticipated personnel activities in areas of the plant containing radioactive materials. Design recommendations given in Regulatory Guide 8.8 are utilized where practicable.

The nuclear radiation shielding is designed to provide personnel protection and is based on the following operating states:

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

The shielding design objectives for the plant during these operating states are:

- Radiation exposure to plant operating personnel, contractors, administrators, visitors, and site boundary occupants is ALARA and within the limits of 10 CFR 20.

- Sufficient personnel access and occupancy time is provided to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

- Reduce potential equipment neutron activation and mitigate the effects of radiation on materials.

- Provide sufficient shielding for the control room so that for design basis accidents (DBAs) the direct dose plus the inhalation dose (calculated in Chapter 15) does not exceed the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

12.3.2.2 General Shielding Design

Systems containing radioactivity and other sources of radiation are identified for four plant conditions defined in Subsection 12.3.2.1. Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area shown in Figure 12.3-1. Design criteria for shield penetrations are consistent with the recommendations of Regulatory Guide 8.8 and are described in Subsection 12.3.1.1.2.

Materials used in shielding typically include lead, steel, water, and concrete. The material used for most of the plant shielding is ordinary concrete with a bulk density of approximately 140 lb/ft^3. Whenever poured-in-place concrete has been replaced by concrete blocks, an equivalent shielding basis as determined by the density of the concrete block is selected. Steel is used as shielding in the chemical and volume control system and other modules, as well as around the reactor vessel flange at the floor of the refueling cavity. Water is used as the primary shield material for areas above the spent fuel storage area and refueling cavity during refueling operations.

12.3.2.2.1 Containment Shielding Design

During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor coolant system components.

For design basis accidents, the shield building and the main control room shielding reduce the plant radiation intensities from fission products inside the containment to acceptable levels, as defined by 10 CFR 50, Appendix A, General Design Criterion 19, for the main control room. (See Subsection 12.3.2.2.7.)

Where personnel locks and equipment hatches or penetrations pass through the shield building wall, additional shielding is provided to attenuate radiation to the level defined by the outside radiation zone during normal operation and shutdown, and to acceptable levels during design basis accidents as defined by General Design Criterion 19.

12.3.2.2.2 Containment Interior Shielding Design

During reactor operation, many areas inside the containment are Zone V or greater and are normally inaccessible. Shielding is provided to reduce dose rates to approximately 100 mrem/hr or less in areas of the containment that potentially require access at power. These are the Zone IV or lower areas shown in Figure 12.3-1.

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

The primary shield is a large mass of reinforced concrete surrounding the reactor vessel. The primary shield meets the following objectives:

- In conjunction with the secondary shield, reduce the radiation level from sources within the reactor vessel and reactor coolant system to allow limited access to the containment during normal, full-power operation.
- After shutdown, limit the radiation level from sources within the reactor vessel, permit limited access to the reactor vessel and the reactor coolant system equipment.
- Limit neutron activation of component and structural materials.

The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps, and steam generators. This shield protects personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full-power operation.

The reactor cavity has been designed so that the dose rates on the operating deck due to neutron streaming are less than 100 mrem/hr.

Components of the purification portion of the chemical and volume control system (CVS) in the containment are located in a shielded compartment. Shielding is provided for equipment in the purification system consistent with its postulated maximum activity (Subsection 12.2.1) and with the access and zoning requirements of adjacent areas. This equipment includes the regenerative heat exchanger, the letdown heat exchanger, chemical and volume control system filters and demineralizers, and the letdown lines.

After shutdown, the containment is accessible for limited periods of time and access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1000 mrem/hr, depending on the location inside the containment (excluding reactor cavity). These dose rates result from residual fission products and neutron activation products (components and corrosion products) in the reactor coolant system.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels to below zone levels specified for adjacent areas. Water provides the shielding over the spent fuel assemblies during fuel handling.

12.3.2.2.3 Auxiliary Building Shielding

During normal operations, the major components in the auxiliary building with potentially high radioactivity are those in liquid radwaste, gaseous radwaste, and spent resin handling systems. Shielding is provided consistent with the postulated maximum activity (See Sections 11.1, 11.2, 11.3, and 12.2) and with the access and zoning requirements of adjacent areas. (See Figure 12.3-1.)

Depending on the equipment in the compartments, the radiation zones vary. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access.

Concrete plugs are utilized to provide necessary access for equipment maintenance and spent filter cartridge replacement. Where necessary, labyrinth entrances with provisions for adequate ingress and egress for equipment maintenance and inspection are provided and are designed to be consistent with the access and zoning requirements of adjacent areas.

Following reactor shutdown, the normal residual heat removal (RNS) system pumps and heat exchangers are in operation to remove heat from the reactor coolant system. The radiation levels in the vicinity of this equipment temporarily reach Zone V or higher levels due to corrosion and fission products in the reactor coolant water. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas.

12.3.2.2.4 Fuel Handling Area Shielding Design

The concrete shield walls surrounding the spent fuel cask loading and decontamination areas, and the shield walls surrounding the fuel transfer and storage areas are sufficiently thick to limit radiation levels outside the shield walls in accessible areas to Zone II. The building external walls are sufficient to shield external plant areas which are not controlled to Zone I.

Spent fuel removal and transfer operations are performed under borated water to provide radiation protection and maintain subcriticality. Minimum allowable water depths above active fuel in a fuel assembly during fuel handling are 8.75 feet in the reactor cavity and 8.75 feet in the fuel transfer canal and spent fuel pool. This limits the dose to personnel on the spent fuel pool handling machine to less than 2.5 mrem/hr for an assembly in a vertical position. Minimum water depth above the

stored assemblies is about 26 feet, and for this depth the dose rate at the pool surface is insignificant. The concrete walls of the fuel transfer canal and spent fuel pool walls supplement the water shielding and limit the maximum radiation dose levels in working areas to less than 2.5 mrem/hr.

The spent fuel pit cooling system (SFS) shielding (Section 9.1) is based on the activity discussed in Subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the spent fuel pit cooling system to be shielded includes the spent fuel cooling system heat exchangers, pumps, piping, filters and demineralizers which may be contaminated with radioactive crud.

12.3.2.2.5 Radwaste Building Shielding Design

Shielding is provided as necessary for the waste storage areas in the radwaste building to meet the radiation zone and access requirements. Depending on the equipment in the compartments, the radiation zoning varies from Zone I through IV as shown on the radiation zone drawing of Figure 12.3-1. Temporary partitions and shield walls will be provided, as required, to supplement the permanent shield walls surrounding the waste accumulation room inside the radwaste building. The three bunkers described in Subsection 11.4.2.5.2 include removable steel shield bunker roof plates that provide vertical shielding during storage of moderate and high activity waste to maintain radiation levels as Zone II to Zone III on the normally unoccupied and access controlled radwaste building roof, and removable steel shield bunker door plates, that may be installed for ALARA considerations. The removable steel shield bunker door plates are not required to be installed to maintain radiation levels in the worker occupied areas in the radwaste building and in the adjacent plant yard areas as Zone I to Zone IV as defined in Figure 12.3-1 (Sheet 14).

12.3.2.2.6 Turbine Building Shielding Design

The steam generator blowdown demineralizers and the resin columns associated with the secondary sampling system (SSS) steam generator blowdown sample panel are shielded when required to meet the radiation zone and access requirements. Radiation shielding is not required for other process equipment located in the turbine building. Space has been provided so that shielding may be added around the steam generator blowdown system (BDS) filter, the electrodeionization (EDI) unit, and the condensate polishing demineralizers if they become radioactive.

12.3.2.2.7 Control Room Shielding Design

The design basis loss-of-coolant accident dictates the shielding requirements for the control room. Consideration is given to shielding provided by the shield building structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criterion 19.

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for plant buildings containing radiation sources so that radiation levels at the outside surfaces of the buildings are maintained below Zone I levels. Plant yard areas that are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. Tanks containing radioactive materials are not located in the yard.

12.3.2.2.9 Spent Fuel Transfer Canal and Tube Shielding

The spent fuel transfer tube is shielded to within adjacent area radiation zone limits. This is primarily achieved through the use of concrete and water. The only removable shielding consists of concrete

or steel hatches which reduce radiation in accessible areas to within those levels prescribed in the normal operation radiation zone maps (Figure 12.3-1).

The spent fuel transfer tube is completely enclosed in concrete and there is no unshielded portion of the spent fuel transfer tube during the refueling operation. The only potential radiation streaming path associated with the tube shielding configuration is the 2 inch (5.08 cm) seismic gap between the fuel transfer tube shielding and the steel containment wall. Shielding of this gap is provided by a water-filled bladder. This "expansion gap" radiation shield provides effective reduction of the radiation fields during fuel transfer and accommodates relative movement between the containment and the concrete transfer tube shielding with no loss in shield integrity. A removable hatch in the shield configuration provides access for inspection of the fuel transfer tube welds. The opening of this hatch is administratively controlled and is treated as an entrance to a very high radiation area under 10 CFR 20. This hatch is in place during the spent fuel transfer operation.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided for compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in Chapter 11 and Section 12.2. The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as practicable the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard references (References 1 through 6).

The geometric model assumed for shielding evaluation of most tanks, heat exchangers, filters, ion exchangers, and the containment is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source.

Computer codes based on point kernel and Monte Carlo methods are used to calculate gamma dose rates. Most dose rates for non-complex geometries are calculated with a point kernel code MicroShield 6.20 (Reference 22), which is a PC shielding code with a menu-guided user-interface. For complex geometries, Monte Carlo or discrete ordinate methods were used for radiation analysis. Some simplifications are made in the modeling, concerning non-active components connected to the sources, and shielding. As a rule, these simplifications result in conservative dose rate estimates, but do not significantly affect the overall evaluation of the radiological conditions in the containment. Non-homogenous sources, such as fuel assemblies, ion exchange resin beds are homogenized, where this does not underestimate the dose rates.

Complex geometries are modeled in MCNP code (Reference 21). Due to the need of larger computer and work resources MCNP is used only in those cases that cannot be calculated by methods based on line-of-sight attenuation such as point kernel method. Such cases may involve labyrinth structures, penetrations, dominance of scattered radiation etc.

For very simple geometries also analytical formulas using gamma energy yields of radioactive isotopes are used.

The source activity (Ci) and gamma ray source strengths (MeV/sec) are calculated using one of the following computer codes: ORIGEN (Reference 17), SOURCE2/ACCUM (Reference 12), or RADGAS3 (Reference 13). ACCUM (Reference 12) is an option within SOURCE2 that computes isotope accumulation for several time periods from a given flow of isotopes in curies per second. This accumulated activity may then be decayed for any number of decay times at which gamma energy spectra and isotope Curie activity are computed. The generation of daughter products is included during the accumulation and decay periods. FIPCO, CORA, and RADGAS3 compute isotopic activity in radioactive liquid and gaseous systems. The total activity in system lines or equipment is

computed from the initial isotope flow, equipment accumulating (operating) time, and parameters which describe the physical accounts for instantaneous mixing or uniform flow and plateout of particulate daughter products. Isotope data is based on the Table of Isotopes (Reference 5) and ORIGEN library data (Reference 6).

The shielding thicknesses of walls and slabs are selected to reduce the aggregate computed radiation level from the contributing sources below the upper limit of the radiation zone specified for each plant area. The labyrinths are constructed so that the scattered dose rate, plus the transmitted dose rate through the shield wall from all contributing sources, is below the upper limit of the radiation zone specified for each plant area. Shielding requirements in each plant area are evaluated at the point of maximum radiation dose through any wall. In addition, for shielding design purposes the concrete density of 140 lb/ft^3 was assumed. Therefore, the actual anticipated radiation level in each plant area is less than this maximum dose and consequently less than the radiation zone upper limit.

Neutron radiation is calculated either with MCNP code or hand calculation methods combined to literature data on neutron attenuation (References 7 and 8).

12.3.3 Ventilation

The plant heating, ventilating, and air-conditioning systems are designed to provide a suitable environment for personnel and equipment during normal operation.

12.3.3.1 Design Objectives

The plant heating, ventilating, and air-conditioning systems for normal operation are designed to meet the requirements of 10 CFR 20 and 10 CFR 50.

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

- During normal operation the average and maximum airborne radioactivity levels to which plant personnel are exposed in restricted areas of the plant are ALARA and within the limits specified in 10 CFR 20. The average and maximum airborne radioactivity levels in unrestricted areas of the plant during normal operation, are ALARA and within the limits of 10 CFR 20.

- During normal operations the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary is ALARA and within the limits specified in 10 CFR 20 and 10 CFR 50, Appendix I.

12.3.3.3 Design Features

To accomplish the design objectives and to conform to the design criteria, the following design features are incorporated wherever practicable.

12.3.3.3.1 Design Features to Minimize Airborne Radioactivity

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

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

- Welded piping systems are employed on systems containing radioactive fluids to the maximum extent practicable. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.

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

- Design of equipment incorporates features that minimize the spread of radioactivity during maintenance operations. These features include flush and drain connections on pump casings for draining and flushing the pump prior to maintenance and flush connections on piping systems that could become highly radioactive.

12.3.3.3.2 Design Features to Control Airborne Radioactivity

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

- In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than is supplied to that area. This minimizes the amount of uncontrolled exfiltration from the area.

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

- The ventilation system design for radiologically controlled areas is discussed in Subsections 9.4.3, 9.4.7, 9.4.8, and 9.4.11. The exhaust air from these areas is normally unfiltered except for the containment atmosphere which is filtered by the containment air filtration system exhaust filters. A description of these filter units is given in Subsection 12.3.3.5.

- Air discharged from the containment is passed through high efficiency particulate air filters and charcoal adsorbers to remove particulates and halogens. Air exhausted from the auxiliary building, fuel handling area of the auxiliary building, and the annex building is monitored for high airborne activity. Means are provided to shut off supply air and divert exhaust air through high efficiency particulate air filters and charcoal adsorbers upon detection of high airborne activity. Alarms are provided in the main control room for these discharge flows and for flows from the radwaste building and the health physics/hot machine shop area. These alarms alert the operator of high radioactivity concentrations in the air. This minimizes the discharge of contaminants to the environment and in-plant exposures.

- Atmospheric tanks which contain radioactive materials are vented to the respective building ventilation system for release to the monitored plant vent.

12.3.3.3.3 Design Features to Minimize Personnel Exposure from HVAC Equipment

- The guidelines of Regulatory Guide 8.8 have been utilized, as practicable, in the design of the plant ventilation systems.

- Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The HVAC system is designed to allow rapid replacement of components.

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

- Ventilating air for radiologically controlled areas of the plant is a once-through design.

- Access to ventilation systems in potentially radioactive areas can result in operator exposure during maintenance, inspection, and testing. Equipment locations are selected to minimize personnel exposures. The outside air supply units and building exhaust system components are located in ventilation equipment rooms. These equipment rooms are accessible to the operators. Work space is provided around each unit for anticipated maintenance, testing, and inspection.

12.3.3.4 Design Description

The ventilation systems serving the following structures are considered to be potentially radioactive and are discussed in detail in Section 9.4.

- Containment building (See Subsection 9.4.7)
- Auxiliary building (See Subsection 9.4.3)
- Fuel handling area of the auxiliary building (See Subsection 9.4.3)
- Annex building (See Subsection 9.4.3)
- Radwaste building (See Subsection 9.4.8)
- Health physics and hot machine shop (See Subsection 9.4.11)

The main control room is considered to be a nonradioactive area. The associated ventilation system design is described in Section 6.4 and Subsection 9.4.1.

Other structures contain insignificant sources of airborne radioactivity and are not addressed in this chapter.

12.3.3.5 Air Filtration Units

The guidance and recommendations of Regulatory Guide 1.140 concerning maintenance and inplace testing provisions for atmospheric cleanup systems, air filtration, and adsorption units are used as a guide in the design of the various ventilation systems. The extent to which Regulatory Guide 1.140 has been incorporated is discussed in Subsection 1.9.1. Figure 12.3-3 shows the typical layout of an air filtration unit.

Provisions specifically included to minimize personnel exposures and to facilitate maintenance or inplace testing operations are as follows.

A. The loading of the filters and adsorbers with radioactive material during normal plant operation is a slow process. Therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity on a scheduled maintenance basis with portable equipment. The filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard. No shielding is provided since it is not required for the level of radioactivity accumulation during normal operation. In case of excessive radioactivity caused by a postulated accident, the filter is replaced before normal personnel access is resumed. It is not necessary for workers to handle filter units immediately after a design basis accident, so exposures can be minimized by allowing the short-lived isotopes to decay before changing the filter.

B. Active components of the atmospheric cleanup systems are designed for ease of removal.

C. Access to active components is direct from working platforms to simplify element handling. Ample space is provided on the platforms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling equipment and inplace testing devices.

D. No filter bank is more than three filter cells high, where each filter cell is 2 feet by 2 feet. The access to the level or platform at which the filter is serviced is by stairs.

E. The clear space for access to filter banks and active components is a minimum of 20 inches by 50 inches.

F. The HEPA filter banks are designed with replaceable cells that are clamped in place against compression seals. The charcoal adsorbers are designed to be replaced with bulk charcoal using a vacuum transfer system. The filter housing is designed and tested to be airtight with bulkhead type doors that are closed against compression seals.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

For a description of the radiation monitoring system (RMS), refer to Section 11.5.

Procedures detail the criteria and methods for obtaining representative measurement of radiological conditions, including in-plant airborne radioactivity concentrations in accordance with applicable portions of 10 CFR Part 20 and consistent with the guidance in Regulatory Guides 1.21-Appendix A, 8.2, 8.8, and 8.10. Additional discussion of radiological surveillance practices is included in the radiation protection program description provided in Appendix 12AA.

Surveillance requirements are determined by the functional manager in charge of radiation protection based on actual or potential radiological conditions encountered by personnel and the need to identify and control radiation, contamination, and airborne radioactivity. These requirements are consistent with the operational philosophy in Regulatory Guide 8.10. Frequency of scheduled surveillance may be altered by permission of the functional manager in charge of radiation protection or their designee. Radiation Protection periodically provides cognizant personnel with survey data that identifies radiation exposure gradients in area resulting from identified components. This data includes recent reports, with survey data, location and component information.

The following are typical criteria for frequencies and types of surveys:

<u>Job Coverage Surveys</u>

- Radiation, contamination, and/or airborne surveys are performed and documented to support job coverage.
- Radiation surveys are sufficient in detail for Radiation Protection to assess the radiological hazards associated with the work area and the intended/specified work scope.
- Surveys are performed commensurate with radiological hazard, nature and location of work being conducted.
- Job coverage activities may require surveys to be conducted on a daily basis where conditions are likely to change.

Radiation Surveys

- Radiation surveys are performed at least monthly in any radiological controlled area (RCA) where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.
- Radiation surveys are performed prior to or during entry into known or suspected high radiation areas for which up to date survey data does not exist.
- Radiation surveys are performed prior to work involving highly contaminated or activated materials or equipment.
- Radiation surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.
- Radiation surveys are performed to support movement of highly radioactive material.
- Neutron radiation surveys are performed when personnel may be exposed to neutron emitting sources.

Contamination Surveys

- Contamination surveys are performed at least monthly in any RCA where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.
- Contamination surveys are performed during initial entry into known or suspected contamination area(s) for which up to date survey data does not exist.
- Contamination surveys are performed at least daily at access points, change areas, and high traffic walkways in RCAs that contain contaminated areas. Area access points to a High Radiation Area or Very High Radiation Area are surveyed prior to or upon access by plant personnel or if access has occurred.
- Contamination surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.
- A routine surveillance is conducted in areas designated by the functional manager in charge of radiation protection or their designee likely to indicate alpha radioactivity. If alpha contamination is identified, frequency and scope of the routine surveillance is increased.

Airborne Radioactivity Surveys

- Airborne radioactivity surveys are performed during any work or operation in the RCA known or suspected to cause airborne radioactivity (e.g., grinding, welding, burning, cutting, hydrolazing, vacuuming, sweeping, use of compressed air, using volatiles on contaminated material, waste processing, or insulation).
- Airborne radioactivity surveys are performed during a breach of a radioactive system, which contains or is suspected of containing significant levels of contamination.
- Airborne radioactivity surveys are performed during initial entry (and periodically thereafter) into any known or suspected airborne radioactivity area.

- Airborne radioactivity surveys are performed immediately following the discovery of a significant radioactive spill or spread of radioactive contamination, as determined by the functional manager in charge of radiation protection.

- Airborne radioactivity surveys are performed daily in occupied radiological controlled areas where the potential for airborne radioactivity exists, including containment.

- Airborne radioactivity surveys are performed any time respiratory protection devices, alternative tracking methods such as derived air concentration-hour (DAC-hr), and/or engineering controls are used to control internal exposure.

- Airborne radioactivity surveys are performed using continuous air monitors (CAMs) for situations in which airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalations of radioactivity by workers. Determination of air flow patterns are considered for locating air samplers.

- Airborne radioactivity surveys are performed prior to use and monthly during use on plant service air systems used to supply air for respiratory protection to verify the air is free of radioactivity.

- Tritium sampling is performed near the spent fuel pit when irradiated fuel is in the pit and other areas of the plant where primary system leaks occur and tritium is suspected.

Appropriate counting equipment is used based on the sample type and the suspected identity of the radionuclides for which the sample is being done. Survey results are documented, retrievable, and processed per site document control and records requirements consistent with Regulatory Guide 8.2. Completion of survey documentation includes the update of room/area posting maps and revising area or room postings and barricades as needed.

Air samples indicating activity levels greater than a procedure specified percentage of DAC are forwarded to the radiochemistry laboratory for isotopic analysis. Samples which cannot be analyzed on-site are forwarded to an offsite laboratory or a contractor for analysis; or, the DAC percentage may be hand calculated using appropriate values from 10 CFR Part 20, Appendix B.

The responsible radiation protection personnel review survey documentation to evaluate if surveys are appropriate and obtained when required, records are complete and accurate, and adverse trends are identified and addressed.

An in-plant radiation monitoring program maintains the capability to accurately determine the airborne iodine concentration in areas within the facility where personnel may be present under accident conditions. This program includes the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment consistent with Regulatory Guides 1.21 (Appendix A) and 8.8. Training and personnel qualifications are discussed in Appendix 12AA.

Emergency operating procedures include provisions for use of a portable monitoring system, consistent with the criteria in NUREG-0737, Item III.D.3.3, to sample and analyze for radioiodine in areas of the plant during and following an accident. Procedures include methods for taking and analyzing samples in the field, as well as for analyzing samples in the count room facility, accounting for techniques to reduce counting system saturation from a high-activity sample. Accident monitoring instrumentation complies with applicable parts of 10 CFR Part 50, Appendix A.

Sampling cartridges can be removed to a low background area for further analysis. These cartridge samples can be purged of any entrapped noble gases, when necessary, prior to being analyzed.

12.3.5 Combined License Information

12.3.5.1 Administrative Controls for Radiological Protection

The administrative controls for use of the design features provided to control access to radiologically restricted areas, including potentially very high radiation areas, are addressed in Subsection 12.5.4 and Appendix 12AA.

12.3.5.2 Criteria and Methods for Radiological Protection

The criteria and methods for obtaining representative measurement of radiological conditions, including airborne radioactivity concentrations in work areas, and the use of portable instruments, and the associated training and procedures, are addressed in Subsection 12.3.4.

12.3.5.3 Groundwater Monitoring Program

In accordance with Reference 23, the groundwater monitoring program is addressed in Appendix 12AA.

12.3.5.4 Record of Operational Events of Interest for Decommissioning

In accordance with Reference 23, the program to provide documentation of operational events deemed to be of interest for decommissioning, including remediation of any leaks that have the potential to contaminate groundwater, is addressed in Appendix 12AA.

12.3.6 References

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25. EPRI NP-2681, Project 1784-3, “Evaluation of Cobalt Sources in Westinghouse-Designed Three- and Four-Loop Plants,” Final Report, October 1982.

Table 12.3-1
Equipment Specification Limits for Cobalt Impurity Levels(1)

Region, Component, or Application (Specific inclusions and examples are noted parenthetically.)			Maximum Weight Percent of Cobalt
Primary Components	Steam generator tubing	(Inconel tubing)	0.015
	Primary components other than steam generators	(Primary weld cladding surfaces core shroud, lower and upper core plates, lower core barrel and neutron panels or thermal shields)	0.05
Fuel Assembly Components		Inconel and stainless steel components in the fuel assembly	
Auxiliary Heat Exchangers			
Other Components in Regions of High Neutron Flux			
Steam Generator Surfaces Other than Tubing			0.10
Small Components	In regions of high neutron flux		0.20
Bolting Materials or Fasteners	Outside of regions of high neutron flux	(Bolting materials in primary and auxiliary components)	Not limited
Weld Material Other than Cladding(2)			
Bearing and Hardfacing Materials(3)			
Other Auxiliary Components(2)		(Valves, piping, instrumentation, and tanks)	

Notes:

1. Parts or components identified to exceed the cobalt limits are evaluated for acceptability considering actual cobalt content, the wetted surface area of the part, the location of the part within the primary system, the estimated plant cobalt input, and the estimates for personnel dose presented in Section 12.4. The evaluation considers industry ALARA guidance in Reference 24, along with the methods of Reference 25 to evaluate cobalt input to the RCS, and in conjunction with ALARA principles from Regulatory Guide 8.8, the estimated dose impact from each exceedance is estimated. The dose impact for the part or component is then used to determine whether re-fabrication, replacement, or use of the part as-is will be warranted to satisfy ALARA design principles. The evaluation methods maintain part and plant design conformance with ALARA design principles as described in this section of the licensing basis and Regulatory Guide 8.8.
2. Stainless steels manufactured without cobalt limitations generally contain 0.15 to 0.20 weight percent of cobalt.
3. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations.

LEGEND:

A. PLANT RADIATION ZONES:

DESIGNATION	MAXIMUM DESIGN DOSE RATE	DESCRIPTION
0	≤ 0.05 mRem/hr	NO RADIATION SOURCES; UNLIMITED GENERAL OCCUPANCY; OUTSIDE "CONTROLLED AREA"
I	≤ 0.25 mRem/hr	VERY LOW OR NO RADIATION SOURCES; INSIDE "CONTROLLED AREA" AND OUTSIDE "RESTRICTED AREA"
"RESTRICTED AREA" ZONES		
II	≤ 2.5 mRem/hr	LOW RADIATION SOURCES; UNLIMITED WORKER OCCUPANCY
III	≤ 15.0 mRem/hr	LOW-TO-MODERATE RADIATION SOURCES; LIMITED WORKER OCCUPANCY
IV	≤ 100 mRem/hr	MODERATE RADIATION SOURCES; LIMITED WORKER OCCUPANCY
V	≤ 1 Rem/hr	HIGH RADIATION SOURCES; LIMITED WORKER OCCUPANCY
VI	≤ 10 Rem/hr	SAME AS ZONE V ABOVE
VII	≤ 100 Rem/hr	SAME AS ZONE V ABOVE
VIII	≤ 500 Rad/hr	SAME AS ZONE V ABOVE
IX	> 500 Rad/hr	VERY HIGH RADIATION SOURCES; VERY LIMITED WORKER ACCESS

B. DRAWING SYMBOLS:

V/IV - UPPER RADIATION ZONE NUMERAL FOR FULL POWER OPERATION/ LOWER NUMERAL FOR 24 HOURS AFTER PLANT SHUTDOWN (IF DIFFERENT)

- RADIATION ZONE BOUNDARY

C. GENERAL DRAWING NOTES

1. ACCESS CONTROL REQUIREMENTS AND TRAFFIC PATTERNS ARE SHOWN IN SERIES 201 DRAWINGS.

2. DOSE RATES INSIDE CONTAINMENT DURING POWER OPERATION ARE SUBJECT TO SIGNIFICANT VARIABILITY OWING TO LOCALIZED NEUTRON STREAMING/SCATTERING EFFECTS. ACTUAL RADIATION FIELDS WILL BE DETERMINED FROM RADIATION SURVEYS AND ACCESS TO THE CONTAINMENT DURING POWER OPERATION WILL BE STRICTLY CONTROLLED.

Figure 12.3-1 (Sheet 1 of 16)
Radiation Zones, Normal Operation/Shutdown Legend

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 2 of 16)
Site Radiation Zones, Normal Operations/Shutdown

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 3 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 66'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 4 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 82'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 5 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 96'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 6 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 100'-0" & 107'-2"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 7 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 117'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 8 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 135'-3"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 9 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 153'-0" & 160'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 10 of 16)
Radiation Zones, Normal Operations/Shutdown
Nuclear Island, Elevation 160'-6" & 180'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

(Note: This figure replaces Figure 12.3-1 Sheet 11 of 16.)

Figure 12.3-201
Radiation Zones, Normal Operations/Shutdown
Annex Building, Elevation 100'-0" & 107'-2"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 12 of 16)
Radiation Zones, Normal Operations/Shutdown
Annex Building, Elevation 117'-6" & 126'-3"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 13 of 16)
Radiation Zones, Normal Operations/Shutdown
Annex Building, Elevation 135'-3", 150'-3", 156'-0" & 158'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 14 of 16)
Radiation Zones, Normal Operations/Shutdown
Radwaste Building, Elevation 100'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 15 of 16)
Radiation Zones, Normal Operations/Shutdown
Turbine Building, Elevation 100'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-1 (Sheet 16 of 16)
Radiation Zones, Normal Operations/Shutdown
Turbine Building, Elevation 120'-6"

LEGEND:

A. POST-ACCIDENT RADIATION ZONES:

DESIGNATION	MAXIMUM DESIGN DOSE RATE	DESCRIPTION
0	≤ 0.05 mRem/hr	NO RADIATION SOURCES
I	≤ 0.25 mRem/hr	VERY LOW OR NO RADIATION SOURCES
II	≤ 2.5 mRem/hr	LOW RADIATION SOURCES
III	≤ 15.0 mRem/hr	LOW-TO-MODERATE RADIATION SOURCES
IV	≤ 100 mRem/hr	MODERATE RADIATION SOURCES
V	≤ 1 Rem/hr	HIGH RADIATION SOURCES
VI	≤ 10 Rem/hr	SAME AS ZONE V ABOVE
VII	≤ 100 Rem/hr	SAME AS ZONE V ABOVE
VIII	≤ 500 Rad/hr	SAME AS ZONE V ABOVE
IX	> 500 Rad/hr	VERY HIGH RADIATION SOURCES

B. DRAWING SYMBOLS:

VI	RADIATION ZONE NUMERAL AT POST-ACCIDENT PEAK
ECS	DOMINANT POST-ACCIDENT RADIATION SOURCE(S)

– NON-RADIOACTIVE AREA BOUNDARY

– RADIOACTIVE AREA BOUNDARY

– ANNEX AREA BOUNDARY

– RADIATION ZONE BOUNDARY

– POST-ACCIDENT ACCESS ROUTE

C. POST-ACCIDENT SOURCES:

SYMBOL	POST-ACCIDENT RADIATION SOURCE
ECS	EXTERNAL CLOUD SHINE
NRA	NON-RADIOACTIVE AUXILIARY BUILDING AREA CLOUD
RAC	RADIOACTIVE AUXILIARY BUILDING AREA CLOUD
SCC	SHIELDED CONTAINMENT CLOUD
UCC	UNSHIELDED CONTAINMENT CLOUD
CPS	CONTAINMENT AND PENETRATION RADIATION STREAMING
AXC	ANNEX BUILDING AREA CLOUD
PAS	POST-ACCIDENT SAMPLE PIPING

D. GENERAL DRAWING NOTES:

1. ZONING IS BASED ON PEAK POST-ACCIDENT DOSE RATES IN THE DESIGNATED AREA.
2. INCLUDES CONTRIBUTIONS FROM POST-ACCIDENT CONTAINED AND AIRBRONE CLOUD SOURCES.

Figure 12.3-2 (Sheet 1 of 15)
Radiation Zones, Post-Accident Legend

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 2 of 15)
Site Radiation Zones, Post-Accident

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 3 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 66'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 4 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 82'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 5 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 96'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 6 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 100'-0" & 107'-2"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 7 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 117'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 8 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 135'-3"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 9 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 153'-0" & 160'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 10 of 15)
Radiation Zones, Post-Accident
Nuclear Island, Elevation 160'-6" & 180'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

(Note: This figure replaces Figure 12.3-2 Sheet 11 of 15.)

Figure 12.3-202
Radiation Zones, Post-Accident
Annex Building, Elevation 100'-0" & 107'-2"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 12 of 15)
Radiation Zones, Post-Accident
Annex Building, Elevation 117'-6" & 126'-3"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 13 of 15)
Radiation Zones, Post-Accident
Annex Building, Elevation 135'-3", 150'-3", 156'-0" & 158'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 14 of 15)
Radiation Zones, Post-Accident
Radwaste Building, Elevation 100'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-2 (Sheet 15 of 15)
Radiation Zones, Post-Accident
Turbine Building, Elevation 100'-0"

LEGEND:

A. PLANT ACCESS CONTROL PROVISIONS:

AREA TYPE	DOSE RATE	SINGLE AREA	MULTIPLE AREAS
RADIATION AREA	> 5 mRem/hr		
HIGH RADIATION AREA	> 100 mRem/hr	BARRICADED OR ALARMED	BARRICADED OR ALARMED
HIGH RADIATION AREA	> 1 Rem/hr	LOCKED OR (IF OPEN AREA) BARRICADED WITH LOCAL ALARM	LOCKED OR BARRICADED WITH LOCKED COMMON ENTRY AND LOCAL CONTROL POINT
VERY HIGH RADIATION AREA	> 500 Rad/hr	LOCKED	LOCKED OR BARRICADED WITH LOCKED COMMON ENTRY, LOCAL CONTROL POINT AND SURVEILLANCE

B. DRAWING SYMBOLS:

⇨ - PERSONNEL TRAFFIC PATTERN

▲ - ENTRANCE BARRICADE (e.g. ROPE, CHAIN, ETC.)

● - LOCKED ENTRANCE

C - LOCAL ACCESS CONTROL POINT

A - ALARM LOCATION

S - SURVEILLANCE POINT

- ACCESS CONTROL BARRIER (e.g. CHAIN LINK FENCE, ETC.)

- "RESTRICTED" AREA BOUNDARY

- "CONTROLLED" AREA BOUNDARY

C. GENERAL DRAWING NOTES:

1. ACCESS CONTROL PROVISIONS ARE BASED ON NORMAL EXPECTED RADIATION SOURCES.

Figure 12.3-3 (Sheet 1 of 16)
Radiological Access Controls Legend

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 2 of 16)
Site Radiation Access Controls, Normal Operations/Shutdown

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 3 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 66'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 4 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 82'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 5 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 96'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 6 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 100'-0" & 107'-2"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 7 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 117'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 8 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 135'-3"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 9 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 153'-0" & 160'-6"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 10 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Nuclear Island, Elevation 160'-6" & 180'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

(Note: This figure replaces Figure 12.3-3 Sheet 11 of 16.)

Figure 12.3-203
Radiological Access Controls, Normal Operations/Shutdown
Annex Building, Elevation 100'-0" & 107'-2"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 12 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Annex Building, Elevation 117'-6" & 126'-3"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 13 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Annex Building Elevation 135'-3", 150'-3", 156'-0" & 158'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 14 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Radwaste Building, Elevation 100'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 15 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Turbine Building, Elevation 100'-0"

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 12.3-3 (Sheet 16 of 16)
Radiological Access Controls, Normal Operations/Shutdown
Turbine Building, Elevation 120'-6"

12.4 Dose Assessment

Radiation exposures in the AP1000 are primarily due to direct radiation from components and equipment containing radioactive material. In addition, in some areas of the plant there can be radiation exposure to personnel due to the presence of airborne radionuclides. This section addresses the anticipated occupational radiation exposure (ORE) due to normal operation and anticipated inspection and maintenance.

12.4.1 Occupational Radiation Exposure

Radiation exposures to operating personnel are restricted to be within the limits of 10 CFR 20. The health physics program in Section 12.5 and the radiation protection features described in Section 12.3 together maintain occupational radiation exposures as low as reasonably achievable (ALARA).

In the analysis of occupational radiation exposure data from operating plants (domestic plants having Westinghouse designed nuclear steam supply systems), the best operating plant performance is 0.1 man-rem per MWe-year of electricity produced. Major factors contributing to this level of occupational radiation exposure include low plant radiation fields, good layout and access provisions, and operational practices and procedures that minimize time spent in radiation fields.

As discussed in Section 12.3, the AP1000 design incorporates features to reduce occupational radiation exposure that go beyond the designs provided for plants currently in operation.

The estimated annual occupational radiation exposures are developed within the following categories:

- Reactor operations and maintenance
- Routine maintenance
- Inservice inspection
- Special maintenance
- Waste processing
- Fuel handling operations

Exposure data obtained from operating plants have been reviewed to obtain a breakdown of the doses incurred within each category. For several routinely performed operations, this information has been used to develop detailed dose predictive models. These models identify the various steps that are included in the operation, the radiation zones, the required number of workers, and the time to perform each step. This information has been used to develop dose estimates for each of the preceding categories.

There is no separate determination of doses due to airborne activity. Past experience demonstrates that the dose from airborne activity is not a significant contributor to the total doses.

12.4.1.1 Reactor Operations and Surveillance

To support plant operations, the performance of various systems and components is monitored. Also, operation of some manual valves requires personnel to enter radiation fields. Examples of activities in this category are:

- On-Line Exposure
- Routine patrols and inspections
- Tagging
- Testing

- Health physics
- Shielding
- Calibration of instrumentation

When the plant is at power, the containment radiation fields are significantly higher than at plant shutdown. The frequency and duration of at-power containment entries is dependent on the plant operator. Based on review of current plant operations and on the AP1000 design changes and reliability improvements, it is assumed that 100 worker-hours per year spent in the containment during power operations.

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

12.4.1.2 Routine Inspection and Maintenance

Routine inspection and maintenance are required for mechanical and electrical components. Table 12.4-2 provides a breakdown of the collective doses for routine inspection and maintenance. These estimates are based on having good access to equipment (a characteristic of the AP1000 layout).

Table 12.4-4 itemizes the doses estimated to be incurred from steam generator sludge lancing operations and Table 12.4-5 lists the doses resulting from the visual examination of the secondary side of the steam generators.

12.4.1.3 Inservice Inspection

ASME Code, Section XI requires periodic inservice inspection (ISI) on plant safety-related components. The Code defines the inservice inspection interval as a 10-year period and sets requirements for each one-third interval (each 40 months). In general, at least 25 percent (with credit for no more than 33-1/3 percent) of the specified inspections must be performed in each 40-month testing interval. The amount of inspection required for an area varies according to the category but is explicitly defined in the Code. Table 12.4-6 provides the doses for inservice inspection activities.

Detailed listings of the doses associated with certain major inservice inspection activities appear in Table 12.4-7 (eddy current inspection of 33 1/3 percent of the steam generator tubes and plugging of three tubes) and Table 12.4-8 (steam generator exterior). The dose estimates in Table 12.4-7 reflect the dose-reducing features of the AP1000 design, such as:

- Permanent work platforms
- Manway cover handling device
- Improved manway insert fasteners (tapered-end type)
- Trailer-mounted data collection station
- Use of robotics to perform eddy current inspection and tube plugging

12.4.1.4 Special Maintenance

Maintenance that goes beyond the routine scheduled maintenance is considered to be special maintenance. This category includes both the modification of equipment to upgrade the plant and repairs to failed components. Dose estimates assume no significant equipment upgrade efforts. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will be lower for the AP1000 than for current plant designs because of the reduced radiation fields, increased equipment reliability, and the reduced number of components relative to currently operating plants.

In the past, special maintenance of the steam generators has resulted in significant personnel doses. The AP1000 benefits both from design changes and from improved primary and secondary water chemistry. The plugging of three tubes per steam generator each time eddy current examination is performed is included in the inservice inspection category.

No special maintenance activities are forecast for the sealless motor reactor coolant pumps.

Table 12.4-9 provides the estimated doses due to special maintenance operations.

12.4.1.5 Waste Processing

The AP1000 radwaste system designs incorporate an uncomplicated approach to waste processing. The AP1000 design does not include waste or boron recycle evaporators and it does not include a catalytic hydrogen recombiner in the gaseous radwaste system. Elimination of high maintenance components contributes significantly to lower anticipated doses due to waste processing activities.

Estimated annual doses from waste processing operations appear in Table 12.4-10.

12.4.1.6 Fuel Handling

Criticality monitoring of the new fuel handling and storage areas is performed in accordance with 10 CFR 70.24. Details of the fuel handling area monitoring are provided in Subsections 11.5.6 and 11.5.6.4. A criticality excursion will produce an audible local alarm and an alarm in the plant MCR.

The refueling process is labor intensive. Detailed planning and coordination of effort are essential in order to maintain personnel doses as-low-as-reasonably achievable. Incorporation of advanced technology into the refueling process also reduces doses. Table 12.4-11 lists some of the AP1000 features that reduce doses during refueling operations.

Table 12.4-12 provides dose estimates for the various refueling activities.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the six activity categories discussed above are summarized below:

Category	Percent of Total	Estimated Annual (man-rem)
Reactor operations and surveillance	11.9	3.397
Routine inspection and maintenance	37.1	10.608
Inservice inspections	22.0	6.282
Special maintenance	6.1	1.731
Waste processing	10.4	2.971
Refueling	12.6	3.590
Total	100.0	28.579

These dose estimates are based on operation with an 18-month fuel cycle and are bounding for operation with a 24-month fuel cycle.

12.4.1.8 Post-Accident Actions

Requirements of 10 CFR 52.79(b) relative to plant area access and post-accident sampling (10 CFR 50.34 (f) (2)(viii) are included in Subsection 1.9.3. If procedures are followed, the design prevents radiation exposures to any individual from exceeding 5 rem to the whole body or 50 rem to the extremities. Figure 12.3-2 in Section 12.3 contains radiation zone maps for plant areas including those areas requiring post-accident access. This figure shows projected radiation zones in areas requiring access and access routes or ingress, egress and performance of actions at these locations. The radiation zone maps reflect maximum radiation fields over the course of an accident. The analyses that confirm that the individual personnel exposure limits following an accident are not exceeded reflect the time-dependency of the area dose rates and the required post-accident access times. The areas that require post-accident accessibility are:

- Main control room
- Class 1E regulating transformer areas
- Ventilation control area for MCR and I & C rooms with PAMS equipment
- Valve area to align spent fuel pool makeup
- Ancillary diesel generator room
- Passive containment water inventory makeup area

12.4.1.9 Radiation Exposure to Construction Workers

12.4.1.9.1 Site Layout

The physical location of VEGP Units 3 and 4 relative to Units 1 and 2 is depicted on Figure 1.1-202. As shown, Units 3 and 4 will be immediately west of Units 1 and 2. Construction activity will take place outside the Units 1 and 2 protected area, but inside the restricted area boundary.

12.4.1.9.2 Radiation Sources

During the construction of Units 3 and 4, the construction workers could be exposed to radiation sources from the routine operation of Units 1 and 2. Furthermore, Unit 4 construction workers could be exposed to radiation from Unit 3 operation.

12.4.1.9.2.1 Direct Radiation

The principal sources from Units 1 and 2 that contribute to direct radiation exposure at the construction site include the reactor buildings and the planned Independent Spent Fuel Storage Installation (ISFSI), which will be located east of Unit 1 (See Figure 1.1-202). In addition, workers constructing Unit 4 could be exposed to direct radiation from the Unit 3 reactor building.

12.4.1.9.2.2 Gaseous Effluents

Sources of gaseous releases for Units 1 and 2 are currently confined to the following paths: plant vents (Unit 1 and Unit 2), the condenser air ejector, the steam packing exhauster systems (Unit 1 and Unit 2), Radwaste Processing Facility and the DAW (Dry Active Waste Building). Waste gas decay tanks are batch released through the Unit 1 plant vent. The containment purges are released through their respective plant vents. (Reference 203)

The annual releases for the 2002 were reported as 26.3 Ci of fission and activation products, 0.0207 Ci of I-131, 1.67 x 10^{-5} Ci of particulates with half-lives greater than eight days, and 105 Ci of tritium (Reference 202). The annual releases for 2002 were selected because they resulted in the maximum exposure to the public among the years 2001-2004.

Unit 4 construction workers could also be exposed to radioactivity in gaseous effluents from Unit 3. Table 11.3-3 presents the projected gaseous effluent releases for Unit 3.

12.4.1.9.2.3 Liquid Effluents

Effluents from the liquid waste disposal system result in small amounts of radioactivity in the Savannah River. The annual liquid radioactivity releases for 2001 were reported as 0.220 Ci of fission and activation products, 1,490 Ci tritium, and 0.000423 Ci of dissolved and entrained gases (Reference 201). The annual releases for 2001 were selected because they were reported as the maximum exposure to the public among the years 2001-2004.

Unit 4 construction workers could be exposed to radioactivity in liquid effluents from Unit 3, but that is unlikely given that drinking water is derived from sources other than the Savannah River. Table 11.2-7 presents the projected liquid effluent releases for Unit 3. Applying the Units 1, 2, and 3 liquid effluent doses to Unit 4 construction workers is conservative in that it assumes these construction workers engage in the same activities that lead to the calculated liquid effluent doses (i.e., consuming fish and drinking surface water).

12.4.1.9.3 Measured and Calculated Dose Rates

The measured or calculated dose rates used to estimate worker doses are presented below.

12.4.1.9.3.1 Direct Radiation

Units 1 and 2 External Radiation Exposure

TLD data from 2003 is representative of annual results from Units 1 and 2, based on the completeness of the data set and having operated with a 95 percent plant capacity factor for that year. The average accumulated exposure from the six thermoluminescent dosimeters (TLDs) along the Units 1 and 2 Protected Area Fence closest to the construction site over a 365 day period is 115.9 mrem. The average TLD exposure from sixteen environmental locations surrounding the site over a 365 day period is 49.0 mrem. The measured radiation dose from the Protected Area Fence TLDs minus the Surrounding Environmental Site TLD's, is:

115.9 mrem per year - 49.0 mrem per year = 66.9 mrem per year

Independent Spent Fuel Storage Installation (ISFSI)

The dose to construction workers from the planned ISFSI is negligible for the Units 3 and 4 construction workforce.

Unit 3 Direct Radiation Exposure to Unit 4 Construction Workers

Conservatively assuming that the 66.9 mrem per year value presented above for Units 1 and 2 is attributable only to direct radiation from these units, and assuming this would be representative of the direct radiation dose from Unit 3 to Unit 4 construction workers gives a direct radiation dose to Unit 4 construction workers from Unit 3 operations of:

66.9 mrem per year / 2 units = 33.5 mrem per year (for one unit)

Summary of External Radiation

From all of the above sources discussed above, the highest direct radiation dose to construction workers will be during Unit 4 construction and is estimated to be 100.4 mrem per year (66.9 mrem from Units 1 and 2 + 33.5 mrem from Unit 3). The highest direct radiation exposure during Unit 3

construction is estimated to be 66.9 mrem per year (from Units 1 and 2). Therefore the Unit 4 construction workers doses would be bounding and are discussed in the remainder of this section.

12.4.1.9.3.2 Gaseous Effluents

Units 1 and 2

The XOQDOQ and GASPAR II codes were used to calculate the dose to Unit 4 workers from Units 1 and 2 gaseous effluents. The calculation is analogous to that for Units 3 and 4 as described in Subsection 11.3.3.4. Unit 4 construction workers would receive a total body radiation dose of 0.077 mrem per year and a maximum organ (lung) dose of 0.16 mrem per year from Units 1 and 2 normal radiological releases.

Unit 3 Gaseous Effluent Exposure to Unit 4 Construction Workers

Using the XOQDOQ and GASPAR II codes, as described in Subsection 11.3.3.4, Unit 4 construction workers would receive a total body radiation dose of 0.74 mrem per year and a maximum organ (skin) dose of 2.51 mrem per year from Unit 3 normal radiological releases.

12.4.1.9.3.3 Liquid Effluents

Units 1 and 2

The Annual Radioactive Effluent Release Report for 2001 (Reference 201) reports a total body dose of 0.0907 mrem and a critical organ dose (GI-LLI) of 0.153 mrem to the maximally exposed member of the public due to the release of liquid effluents from Units 1 and 2, calculated in accordance with Units 1 and 2 Offsite Dose Calculation Manual (Reference 204). SNC assumes this dose rate represents the rate for construction workers from Units 1 and 2 releases.

Unit 3 Liquid Effluent Exposure to Unit 4 Construction Workers

Using the LADTAP II code, as described in Subsection 11.2.3.5, the maximally exposed member of the public would receive a total body radiation dose of 0.017 mrem per year and a maximum organ (liver) dose of 0.021 mrem per year from normal Unit 3 liquid radiological releases.

12.4.1.9.4 Construction Worker Doses

Construction worker doses were conservatively estimated using the following information:

- The estimated maximum dose rate for each pathway
- An exposure time of 2000 hours per year
- All gaseous releases assumed at ground level
- A peak loading of 4,400 construction workers per year total for two AP1000 units

The estimated maximum annual dose for each pathway as well as the total dose is shown in Table 12.4-201.

12.4.1.9.4.1 Direct Radiation

Subsection 12.4.1.9.3.1 indicates an average annual direct radiation dose of 100.4 mrem based on TLD measurements. These TLD measurements and calculated doses reflect continuous exposures for long periods of time. The average measured dose rate of 100.4 mrem/yr is based on continuous exposure.

Adjusting for an exposure time of 2000 hours/year yields an annual worker whole body dose or total effective dose equivalent (TEDE) of 22.9 mrem.

12.4.1.9.4.2 Gaseous Effluents

The annual gaseous effluent doses to a Unit 4 construction worker after Unit 3 is operating (Subsection 12.4.1.9.3.2), which accounts for an exposure time of 2,000 hours per year, are 0.077 mrem for the total body, and 0.16 mrem for the critical organ (lung) from Units 1 and 2 gaseous effluent releases and 0.74 mrem for the total body, and 2.51 mrem (skin) for the critical organ from Unit 3 gaseous effluent releases. The total dose is 0.81 mrem total body and 2.60 mrem to the critical organ (skin).

12.4.1.9.4.3 Liquid Effluents

As the annual liquid effluent doses to the maximally exposed member of the public in Subsection 12.4.1.9.3 are based on continuous occupancy, they were adjusted for an exposure time of 2000 hr/yr. Although it is unlikely that the construction workers will be exposed to liquid effluent pathways, it is assumed that the liquid effluent dose rates to which the workers will be exposed are the same as those for the maximally exposed member of the public.

The resulting doses are 0.021 mrem for the total body and 0.035 mrem for the critical organ (GL-LLI) from Units 1 and 2 liquid effluent releases and 0.0038 mrem for the total body, and 0.0047 mrem for the critical organ (liver) from Unit 3 liquid effluent releases. The total annual dose is 0.025 mrem total body and 0.037 mrem to the critical organ (GI-LLI).

12.4.1.9.4.4 Total Doses

The annual doses from all three pathways are summarized in Table 12.4-201 and compared to the public dose criteria in the 10 CFR 20.1301 and 40 CFR 190 in Table 12.4-202 and Table 12.4-203, respectively. The unrestricted area dose rate in Table 12.4-202 was estimated from the annual TLD doses. Since the calculated doses (24.1 mrem per year and 0.012 mrem per hour) meet the public dose criteria of the 10 CFR 20.1301 and 40 CFR 190, the workers will not need to be classified as radiation workers. Table 12.4-204 provides documentation confirming that the doses also meet the design objectives of 10 CFR 50, Appendix I, for gaseous and liquid effluents.

The maximum annual collective dose to the AP1000 construction work force (4,400 workers) is estimated to be 106 person-rem. The calculated doses are based on available dose rate measurements and calculations. It is possible that these dose rates will increase in the future as site conditions change. However, the VEGP site will be continually monitored during the construction period and appropriate actions will be taken as necessary to ensure that the construction workers are protected from radiation.

12.4.1.9.4.5 Operating Unit Radiological Surveys

The operating unit conducts radiological surveys in the unrestricted and controlled area and radiological surveys for radioactive materials in effluents discharged to unrestricted and controlled areas in implementing 10 CFR 20.1302. These surveys demonstrate compliance with the dose limits of 10 CFR 20.1301 for construction workers.

12.4.2 Radiation Exposure at the Site Boundary

12.4.2.1 Direct Radiation

The direct radiation from the containment and other plant buildings is negligible. The AP1000 design also provides storage of refueling water inside the containment instead of in an outside storage tank that eliminates it as a radiation source.

12.4.2.2 Doses due to Airborne Radioactivity

Subsection 11.3.3 discusses doses at the site boundary due to activity released as a result of normal operations.

12.4.3 Combined License Information

This section contained no requirement for additional information.

12.4.4 References

201. Southern Nuclear Company, Vogtle Electric Generating Plant - Units 1 and 2, NRC Docket Nos. 50-424 and 50-425, Facility Operating License Nos. NPF-68 and NPF-81, Annual Radioactive Effluent Release Report for January 1, 2001 to December 31, 2001.

202. Southern Nuclear Company, Vogtle Electric Generating Plant - Units 1 and 2, NRC Docket Nos. 50-424 and 50-425, Facility Operating License Nos. NPF-68 and NPF-81, Annual Radioactive Effluent Release Report for January 1, 2002 to December 31, 2002.

203. Southern Nuclear Company, Vogtle Electric Generating Plant - Units 1 and 2, NRC Docket Nos. 50-424 and 50-425, Facility Operating License Nos. NPF-68 and NPF-81, Annual Radioactive Effluent Release Report for January 1, 2003 to December 31, 2003.

204. Southern Nuclear Company, Offsite Dose Calculation Manual for Southern Nuclear Operating Company Vogtle Electric Generating Plant, Version 22, June 25, 2004.

Table 12.4-1
Dose Estimate for Reactor Operations and Surveillance

Work Description	Cycle Dose (man-rem)
Operation Supervision	
On-Line Exposure	2.39
Routine Patrols and Inspections	0.387
Tagging	0.072
Testing	0.177
Health Physics	1.168
Shielding	0.356
Calibration of Instrumentation	0.545
Total Collective Dose (annual)	3.397
Total Collective Dose (per fuel cycle)	5.095

Table 12.4-2
Dose Estimate for Routine Inspection and Maintenance

Work Description	Annual Dose (man-Rem)
Valve Adjustment / Repacking	2.117
Pressurizer Spray Valve Maintenance	0.379
SG Sludge Lance	2.543
Scaffolding Work (erection, removal)	0.253
Snubber Inspections	0.308
Insulation-related Tasks	0.287
Miscellaneous Work	1.775
SG Secondary Side Inspection	0.963
Ex-core Detector Work (removal, maintenance, replacement)	1.880
IRWST / Annulus Maintenance	0.103
Total Collective Dose (annual)	10.608
Total Collective Dose (per fuel cycle)	15.912

Table 12.4-3
Not Used

Table 12.4-4
Dose Estimate for Sludge Lancing of Steam Generators

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Move equipment into containment(c)	2.5	10	4	0.1
Remove insulation and handhole cover(b)	15	2	3	0.09
Initial sludge lance containment equipment setup(c)	6.25(c)	5	12	0.375
Install lance on handholes(b)	15	2	4	0.12
Operate water lance(b)	15	2	12	0.36
Cleanliness Inspection(b)	15	2	2	0.06
Foreign object search and retrieval(b)	15	2	8	0.24
Sludge lance equipment move across containment (SG1 to SG2)(c)	2.5	5	12	0.15
Final tear down and stage sludge lance equipment(c)	15	5	12	0.90
Remove equipment(c)	6.25(d)	10	4	0.25
Install handhole cover and insulation(b)	15	2	5	0.15
Total ORE for both SG = 3.815 man-Rem / 18 month Annual total ORE for two SGs = 2.543 man-Rem/year(a)				

Notes:

a. The dose calculated based on an 18-month fuel cycle bounds plant operation with a 24-month fuel cycle.
b. These activities are performed twice per outage; once for each steam generator.
c. These activities are performed once per outage.
d. Only about 30% of the time for setup and demobilization is spent at the handholes, while the rest of the time is spent outisde of the SG compartment shield.

Table 12.4-5
Dose Estimate for Visual Examination of
Steam Generator Secondary Side

Activity[d]	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Remove insulation and two manway covers[a]	0.4	2	7	0.0056
Inspection equipment pack-in and set-up / check-out	4	2	8	0.064
Inspect separator orifices and feedwater ring[a]	4	2	8	0.064
Install two manway covers and insulation[a]	0.4	2	9	0.0072
Remove handhole cover and insulation	15	2	1	0.030
Photograph support plates	15	2	12	0.360
Install handhole cover and insulation	15	2	5	0.150
Teardown / move and set-up inspection equipment (SG A to SG B)	6.875[c]	2	6	0.083
Total ORE for two SGs = 1.444 man-Rem / 18 months Annual total ORE for two SGs = 0.963 man-Rem/year[b]				

Notes:

a. Secondary side water level at the lower deck plate.
b. The dose calculated based on an 18-month fuel cycle bounds plant operation with a 24-month fuel cycle.
c. Data from existing plants show that 34-35% of the time will be spent at the SG, while the rest of the time will be spent moving the equipment through containment.
d. All of the activities are completed twice per outage, except for the “Teardown / move and set-up inspection equipment (SG A to SG B)” activity, which is performed once per outage.

Table 12.4-6
Dose Estimate for Inservice Inspection

Component	Cycle Dose (man-Rem)
Valve Bodies and Boltings	1.05
Reactor Coolant Loop Piping and Supports	0.15
Other Piping	0.30
Heat Exchanger Shells	0.20
Pressurizer Shell	0.18
Pumps	0.06
Tank Shells and Supports	0.03
Filter Housing and Supports	0.02
SG Primary Side In-service Inspection	2.971
SG Eddy Current Testing	3.862
Reactor Head and Reactor Vessel	0.523
PRHR Inspection in IRWST	0.077
Total Dose (per Fuel Cycle)	9.423
Total Dose (Annual)	6.282

Table 12.4-7 (Sheet 1 of 2)
Dose Estimate for Steam Generator Eddy Current Tube Inspection and Tube Plugging

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Move equipment into containment	0.8	8	4	0.0256
Install ventilation equipment	15	4	2	0.120
Remove insulation on both manway covers	15	2	1	0.030
Remove both manway covers with handling fixture	20	3	4	0.240
Remove both manway inserts	60	2	0.5	0.060
Insert nozzel dams cold & hot legs (manual entries)	2,000/20[a]	3	1.5	0.1567
Install robotic manipulator hot leg, set up eddy current (EC)	20	2	4	0.160
Perform EC exam of 3340 tubes (33 1/3%)	0.4	1	8	0.0032
Remove EC end effector and replace with mechanical plugging end effector	20	1	1	0.020
Insert plugs in three tubes	20	1	0.75	0.015
Transfer robotic arm to cold leg	20	2	2	0.080
Cold leg eddy current	0.4	1	2	0.0008
Remove EC end effector and replace with mechanical plugging end effector	20	1	1	0.020

Note:
a. Nozzle dam installation involves a limited amount of manual access to the SG, requiring an individual to spend about two minutes in a 2,000 mrem/hr field.

Table 12.4-7 (Sheet 2 of 2)
Dose Estimate for Steam Generator Eddy Current Tube Inspection and Tube Plugging

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Insert plugs in 3 tubes	20	1	0.75	0.015
Remove robotic manipulator, equipment teardown	20	2	3	0.120
Remove nozzle dams hot & cold legs	2,000/20[(b)]	3	2	0.1867
Install manway inserts	60	2	1	0.120
Install both manway covers with handling fixture	20	3	6	0.360
Replace insulation both manway covers	20	2	2	0.080
Remove ventilation equipment	20	2	2	0.080
Move equipment out of containment	0.8	8	6	0.0384
Total SG special maintenance ORE = 1.931 man-Rem / 18 months Total SG special maintenance ORE for two SGs = 3.862 man-Rem / 18 months Annual total ORE for two SGs = 2.575 man-Rem/year[(c)]				

Note:

b. This task involves a limited amount of manual access to the SG, requiring an individual to spend about two minutes in a 2,000 mrem/hr field.

c. The dose calculated based on an 18-month fuel cycle bounds plant operation a 24-month fuel cycle.

Table 12.4-8
Dose Estimate for Steam Generator
Inservice Inspection (10-Year Interval)

Activity	Average Dose Rate (mRem/h)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-Rem)
Ultrasonic Inspection				
Tubesheet-to-channel head weld	40	2	48	3.84
Inlet nozzle-to-safe end butt weld	40	2	12	0.960
Outlet nozzle-to-RCP pump casing butt weld	2000/0.04[(a)]	2	120	0.343
PRHR nozzle-to-safe end butt weld	40	2	12	0.960
Elliptical head knuckle-to-upper shell weld	0.4	2	48	0.0384
Tubesheet-to-lower shell weld	40	2	48	3.84
Start-up feedwater nozzle-to-shell weld	4	2	12	0.096
Main feedwater nozzle-to-shell weld	4	2	12	0.096
Main feedwater nozzle inside radius section	4	2	12	0.096
Dye Penetrant Inspection				
Inlet nozzle-to-safe end butt weld	40	2	4	0.320
Outlet nozzle-to-RCP pump casing butt weld	40	2	8	0.640
PRHR nozzle-to-safe end butt weld	40	2	4	0.320
CVS nozzle-to-channel head weld	40	1	1	0.040
Visual Inspection				
Inlet nozzle-to-safe end butt weld	60	1	4	0.240
Outlet nozzle inside radius section	60	1	8	0.480
PRHR nozzle inside radius section	60	1	1	0.060
Studs and nuts less than 2" diameter	20	2	4	0.160
Pedestal steam generator support	15	1	2	0.030
Magnetic Particle Inspection				
Start-up feedwater nozzle-to-shell weld	4	2	4	0.032
Main feedwater nozzle-to-shell weld	4	2	4	0.032
Welded attachments; intermediate and upper lateral support trunnions and support pads	15	2	16	0.480
Total ORE from first 10-year inspection interval = 26.2068 man-Rem. Total ORE for remaining five 10-year inspection intervals = 18.527 man-Rem. Outage average over 60-year (40 18-month outages) life of plant = 2.971 man-Rem/outage. Annual average over 60-year life of plant = 1.980 man-Rem/year				

Note:
a. 10 minutes of this task requires a worker inside of the channel head to set up equipment.

Table 12.4-9
Dose Estimate for Special Maintenance Operations

Work Description	Annual Dose (man-rem)
Emergent Work	2.135
Modifications	0.461
Total Collective Dose (per Fuel Cycle)	2.597
Total Collective Dose (Annual)	1.731

Table 12.4-10
Dose Estimate for Waste Processing

Work Description	Annual Dose (man-rem)
Routine rad waste handling and decontamination	0.740
Dry active waste processing	0.586
Waste liquid processing	0.237
Trash, laundry, and decon activities	1.206
Filter change-outs / Resin sluicing	1.478
Tri-nuclear filters	0.209
Total Collective Dose (per Fuel Cycle)	4.456
Total Collective Dose (Annual)	2.971

Table 12.4-11
Design Improvements That Reduce Refueling Doses

Improved Design/Method	Reference Design/Method
Integrated RV Head Package	Conventional RV head package
RV Head Insulation with Suitcase-Type Fasteners and Permanent ID Markings	Insulation fastened with screws (no markings)
Combination Thermocouples and Flux Detectors	Top-mounted thermocouples and bottom-mounted flux detectors
Quick-Opening Fuel Transfer Tube Closure System	Bolted cover
Quick-Acting Stud Tensioner	Threaded-on stud tensioner
Pass and One-Half Stud Tensioning Procedure	Three-pass stud tensioning procedure
Electrical-Driven Stud Spin-Out Tool	Air-driven, spin-out tool
Permanent Cavity Seal Ring	Bolted or inflatable seal ring
Expandable Stud Hole Plugs	Threaded stud hole plugs
Shielded RV Head Storage Stand	Nonshielded stand
Smooth-Finish Reactor Cavity Liner (#1 Finish)	Rough-finish reactor cavity liner

Table 12.4-12
Dose Estimate for Refueling Activities

Refueling Operations Work Description	Dose (man-rem)
Preparation	0.207
Reactor Disassembly	1.859
Fuel Shuffle	0.571
Reactor Reassembly	2.171
Clean-Up	0.088
Total Refueling Dose:	5.385[b]
Average Annual Dose:	3.590

Note:

a. Based on an 18-month fuel cycle. The stated dose bounds operation with a 24-month fuel cycle.

b. The sum of refueling activity dose estimates is increased by 10% to consider small changes in work activity time durations, the number of personnel exposed, and the addition of unplanned outage tasks.

Table 12.4-201
Annual Construction Worker Doses

	Annual Dose (mrem)		
	Total Body	Critical Organ	Total Effective Dose Equivalent (TEDE)
Direct radiation	22.9	NA	22.9
Gaseous effluents	0.81	2.6 (skin)	1.16
Liquid effluents	0.025	0.037 (GI-LLI)	0.034
Total	23.8	2.6 (skin)	24.1

Table 12.4-202
Comparison with 10 CFR 20.1301 Criteria for Doses to Members of the Public

Criterion	Dose Limit	Estimated Dose (TEDE)
Annual dose (mrem)	100	24.1
Unrestricted area dose rate (mrem/hour)	2	0.012

Table 12.4-203
Comparison with 40 CFR 190 Criteria for Doses to Members of the Public

	Annual Dose (mrem)	
Organ	Limit	Estimated
Total body	25	23.8
Thyroid	75	1.4
Other organ	25	2.6 (skin)

Table 12.4-204
Comparison with 10 CFR 50,
Appendix I Criteria for Effluent Doses

	Annual Dose (mrem)	
	Limit	Estimated
Total body dose from liquid effluents	3	0.025
Organ dose from liquid effluents	10	0.037 (GI-LLI)
Total body dose from gaseous effluents	5	0.81
Organ dose from radioactive iodine and radioactive particulates in gaseous effluents	15	0.81 (thyroid)

12.5 Health Physics Facilities Design

12.5.1 Objectives

The health physics (HP) facilities are designed with the objectives of:

- Providing capability for administrative control of the activities of plant personnel to limit personnel exposure to radiation and radioactive materials as low as reasonably achievable (ALARA) and within the guidelines of 10 CFR 20.
- Providing capability for administrative control of effluent releases from the plant to maintain the releases ALARA and within the limits of 10 CFR 20 and the plant Technical Specifications.
- Providing capability for administrative control of waste shipments from the plant to meet applicable requirements for the shipment and receipt of the material at the storage or burial site.

12.5.2 Equipment, Instrumentation, and Facilities

The health physics (HP) facilities are located at elevation 100′-0″ in the annex building. See Figure 1.2-201 for a plan view of elevation 100′-0″ of the annex building.

12.5.2.1 Access and Exit of Radiologically Controlled Areas

Access to the radiologically controlled area (RCA) encompassing the containment and potentially contaminated areas of the annex, auxiliary, and radwaste buildings is normally through the entry/exit area of the health physics section of the annex building. Exit from the RCA is at the same location.

12.5.2.2 Facilities

The ALARA briefing room is located off the main corridor immediately beyond the main entry to the annex building. Near this room are several offices that may be used for other health physics functions.

Changes rooms are provided where radiation workers remove street clothes and put on modesty garments. These rooms are provided with lockers, wash sinks, showers and toilet facilities.

Radiation workers don anti-contamination clothing in the protective clothing pickup and suitup room. Workers then proceed to the central health physics booth.

Personnel access to and from the RCA is controlled at the health physics booth at the entry/exit points of the health physics area. Logging into the Radiation Permit System and issuance of dosimetry is also handled at this location. The health physics booth is equipped with computer terminals, desks, filing cabinets, and shelves, and other facilities needed for effective control and monitoring of radiation workers in the RCA. Workers are logged into a radiation exposure tracking system. The health physics and security log-in functions are integrated. Facilities and equipment are provided at the health physics booth for the following functions:

- Issuing respirators, as needed
- Issuing radiation dosimetry, as required

- Updating radiation work permits as needed based on information provided by health physics at local control points and at the work locations

The booth has a counter such that the health physics personnel can easily monitor the flow of workers. It is located adjacent to and visible from the health physics pickup and suitup room.

As radiation workers exit the work areas they go through personnel contamination monitors, shower for decontamination if needed, and receive radiologically controlled first-aid if needed. The health physics area contains the personnel contamination monitoring equipment, decontamination shower facilities, and first-aid equipment.

The hot machine shop is located at elevation 107′-2″ in the south end of the annex building. Contaminated equipment can be decontaminated at the facility and maintenance and repair operations can be performed in a low radiation background area with in the RCA and with appropriate radiation protection and contamination control measures in place.

12.5.2.3 Whole Body Counting Instrumentation

The whole body counter(s) is located in a low background radiation area of the plant. The whole body counting equipment is capable of detecting fractional body burdens of gamma emitting radionuclides.

12.5.2.4 Portable Survey Instrumentation

Portable radiation survey instrumentation is stored at the access control health physics booth and at in-plant control points. This instrumentation allows plant personnel to perform radiation, contamination, and neutron surveys, as needed, as well as collect samples for airborne analysis. Adequate facilities are provided in accordance with Subsection 12AA.5.3 for radioactivity analysis laboratory facilities and for calibration of survey instruments.

12.5.2.5 Other Health Physics Instrumentation

The area radiation monitoring system is installed in areas where it is desirable to have constant dose rate information. Monitors indicate dose rate in the control room and provide appropriate alarms upon reaching a preset dose rate. Fixed continuous airborne radioactivity monitors are also provided at strategic locations, where personnel exposure to airborne radionuclides is likely. More information on these fixed instruments is given in Sections 12.3 and 11.5.

12.5.3 Other Design Features

12.5.3.1 Radiation Protection Design Features

Specific design features for maintaining personnel exposure ALARA and plant shielding provisions are incorporated into the plant design. These features are described in Section 12.3.

12.5.3.2 Job Planning Facilities

Areas are provided where personnel may study, as appropriate: blueprints, drawings, photographs, videotapes, previous inspection reports, previous radiation and contamination surveys, or previous RWPs appropriate to the particular job prior to entry into radiation areas to perform inspections. Work rooms are provided where equipment is checked or calibrated to verify it is operating properly prior to entry into the radiation area. The ALARA briefing and operational support room in the annex building is an example of such a facility where job planning and ALARA briefing and debriefing activities can take place.

12.5.3.3 Radwaste Handling

The handling of radwaste has been minimized by plant design. Some of the activities involving radwaste or radioactively contaminated materials are performed offsite or using mobile equipment brought onsite. Cleaning of protective clothing and respiratory protective equipment are activities that are performed offsite or in mobile equipment.

The radwaste system is shielded and incorporates remotely operated liquid and solid radwaste systems. The systems are designed to minimize operator exposure in waste processing and handling operations. The liquid radwaste system and solid waste handling system are described in Chapter 11.

12.5.3.4 Spent Fuel Cask Loading and Shipping

Spent fuel handling and loading of a shipping cask is designed to be performed underwater, using the fuel handling cranes and/or manual extension tools.

Some of the design features included to maintain exposure ALARA are:

- Maintenance of at least 8.75 feet of water above the active fuel in a fuel assembly to minimize direct radiation.
- Purification of fuel pool water to minimize exposure due to water activity.
- Cooling of the spent fuel pool water.
- Providing continuous air sampling while moving fuel to evaluate airborne activity.

12.5.3.5 Normal Operation

The plant is designed so that significant radiation sources are minimized, locally shielded, and/or located in shield cubicles. Much of the instrumentation required for normal operation reads out remotely in the control room or in other low radiation areas. Instrumentation that cannot be placed remotely or that is read infrequently is situated, where possible, so that it can be read from the entrance to the cubicle or from a low radiation area within the cubicle.

Area radiation monitoring equipment, which is included as part of the radiological monitoring system, is available and provides indication of radiation levels and local alarms. The ventilation system is designed to minimize spread of airborne contamination.

12.5.3.6 Sampling

Provisions are made for sampling of radioactive systems in the sampling room. Protective clothing and gloves are available when sampling radioactive systems to prevent contamination of personnel.

12.5.3.7 Surface Coatings

Special coatings are applied to walls and floors of areas containing radioactive fluids, which aid in decontaminating these areas.

12.5.4 Controlling Access and Stay Time

Areas in the plant are classified as non-radiation areas and restricted radiologically controlled areas for radiation protection purposes. Restricted areas are further categorized as radiation areas, high

radiation areas, airborne radioactivity areas, contamination areas, and radioactive materials areas, to comply with 10 CFR 20 and plant procedures and instructions.

Entrance to the RCA area is normally through the access control area at the health physics area entry/exit location in the annex building, see Subsection 12.5.2.

High and very high radiation areas are segregated and identified in accordance with 10 CFR 20. The entrances to high and very high radiation areas are locked or barricaded and equipped with audible and/or visible alarms, as required.

A closed circuit television system may be installed in high radiation areas to allow remote monitoring of individuals entering high radiation areas by personnel qualified in radiation protection procedures.

12.5.5 Combined License Information

The organization and procedures used for radiological protection, and to provide methods so that personnel radiation exposures are maintained ALARA, are addressed in Appendix 12AA.

Radiation protection program information is included in Appendix 12AA.

Appendix 12AA Radiation Protection Program Description

This appendix incorporates NEI 07-03A, Generic FSAR Template Guidance for Radiation Protection Program Description. The numbering of NEI 07-03A is revised from 12.5# to 12AA.5#. Table 13.4-201 provides milestones for radiation protection program implementation.

12AA.5 Radiation Protection Program

A radiation protection program is developed, documented, and implemented through plant procedures that address quality requirements commensurate with the scope and extent of licensed activities, sufficient to ensure compliance with the provisions of 10 CFR Parts 19, 20, 50, 52, and 71 and consistent with the guidance in Regulatory Guides 1.8, 1.206, 8.2, 8.4, 8.5, 8.6, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, 8.38, and the consolidated guidance in NUREG-1736.

In accordance with 10 CFR 20, Subpart B, the purpose of the radiation protection program is to maintain occupational and public doses below regulatory limits and as low as is reasonably achievable (ALARA). To achieve this, the program will include:

I. A documented management commitment to keep exposures ALARA;

II. A trained and qualified organization with sufficient authority and well-defined responsibilities; and

III. Adequate facilities, equipment, and procedures to effectively implement the program.

The operational radiation protection program is implemented in stages consistent with the following milestones:

1. Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18), and thereafter, when such radioactive materials are possessed under this license, the following radiation protection program elements will be in place:

    a. Organization – A radiation protection supervisor and at least one (1) radiation protection technician, each selected, trained and qualified consistent with the guidance in Regulatory Guide 1.8.

    b. Facilities – A facility or facilities to support the receipt, storage and control of non-exempt radioactive sources in accordance with 10 CFR 20.1801, 20.1802, and 20.1906.

    c. Instrumentation and Equipment – Adequate types and quantities of instrumentation and equipment will be selected, maintained, and used to provide for the appropriate detection capabilities, ranges, sensitivities, and accuracies to conduct radiation surveys and monitoring (in accordance with 10 CFR 20.1501 and 20.1502) for the types and levels of radiation anticipated for the non-exempt sources possessed under this license.

    d. Procedures – Procedures will be established, implemented and maintained sufficient to maintain adequate control over the receipt, storage, and use of radioactive materials possessed under this license and as necessary to assure compliance with 10 CFR 19.11 and 19.12 and the applicable portions of 10 CFR Part 20, commensurate with the types and quantities of radioactive materials received and possessed under this license.

e. Training – Initial and periodic training will be provided to individuals responsible for the receipt, control or use of non-exempt radioactive sources possessed under this license in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.27, and 8.29.

2. Prior to receiving reactor fuel under this license, and thereafter, when reactor fuel is possessed under this license, plant procedures on criticality accident requirements will be established, implemented and maintained and radiation monitoring will be provided in accordance with 10 CFR 50.68, in addition to the radiation protection program elements specified under item 1, above.

3. Prior to initial loading of fuel in the reactor, all of the radiation program functional areas described in Appendix 12AA will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, the position of radiation protection manager (as described in Section 13.1) will be filled and at least one (1) radiation protection technician for each operating shift, selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor.

4. Prior to initial transfer, transport or disposal of radioactive materials, the organization, facilities, equipment, instrumentation, and procedures will be in place as necessary to assure compliance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71.

The radiation protection program content and effectiveness of implementation are reviewed periodically (at least annually) pursuant to plant procedures.

12AA.5.1 Management Policy

Plant management will establish written policy on radiation protection that is consistent with the guidance in Regulatory Guides 8.8 and 8.10, including management's commitment to:

I. Assure that the plant is designed, constructed, and operated such that occupational and public radiation exposures and releases of licensed radioactive materials are ALARA;

II. Comply with regulatory radiation requirements, dose limits, and limits on release of radioactive materials;

III. Implement and maintain a radiation protection program to keep radiation doses below regulatory limits and ALARA;

IV. Assure that each manager and supervisor in the plant organization understands and is held accountable for implementing his or her responsibility to integrate appropriate radiation protection controls into work activities;

V. Assure that each individual working at the facility understands and accepts the responsibility to follow radiation protection procedures and instructions provided by radiation protection staff and to maintain his or her dose ALARA;

VI. Provide the radiation protection manager the delegable authority to stop work or order an area evacuated (in accordance with approved procedures) when, in his or her judgment, the radiation conditions warrant such an action and such actions are consistent with plant safety;

VII. Establish a direct reporting chain of the Radiation Protection Manager to the Plant Manager that is at the same reporting level as, but independent of, the reporting chains for Operations and Maintenance.

VIII. Establish an ALARA Committee with delegated authority from the Plant Manager that includes, at a minimum, the managers of Operations, Maintenance, Work Control, Engineering and Radiation Protection to help assure effective implementation of line organization responsibilities for maintaining worker doses ALARA.

12AA.5.2 Organization

Qualification and training criteria for site personnel are consistent with the guidance in Regulatory Guide 1.8 and are described in Chapter 13. Specific radiation protection responsibilities for key positions within the plant organization are described in Section 13.1.

Subsections 12.5.2.1 through 12.5.2.5 of NEI 07-03A are not incorporated into Appendix 12AA.

12AA.5.3 Facilities, Instrumentation and Equipment

Adequate facilities, instrumentation and equipment are provided to support implementation of the radiation protection program during routine operations, refueling and other outages, abnormal occurrences, and accident conditions. The types and characteristics of facilities, instrumentation, and equipment provided are consistent with the guidance in Regulatory Guides 1.97 (and guidance provided in Branch Technical Position 7-10, Revision 5 to NUREG 0800), 8.2, 8.4, 8.6, 8.8, 8.9, 8.10, 8.15, and 8.28 and the criteria in NUREG-0737, Items II.B.3 and III.D.3.3.

Subsection 12.5.3.1 of NEI 07-03A is not incorporated into Appendix 12AA. Facilities are described in Subsection 12.5.2.2.

12AA.5.3.1 Monitoring Instrumentation and Equipment

Radiation monitoring instrumentation and equipment are selected, maintained and used to provide the appropriate detection capabilities, ranges, sensitivities and accuracies required for the types and levels of radiation anticipated at the plant and in the environs during routine operations, major outages, abnormal occurrences, and postulated accident conditions. The quantities of instrumentation and equipment are sufficient to meet the anticipated needs of the plant during all anticipated conditions –taking into account the amount of instrumentation and equipment that may be unavailable at any one time due to periodic testing and calibration, maintenance, and repair.

The types and nominal characteristics of the instrumentation are as follows:

<u>Laboratory and Fixed Instrumentation</u>

- Multi-channel gamma analysis system to identify and measure gamma emitting radionuclides in solid, liquid and gaseous samples. Some of the sample types analyzed include primary reactor coolant, liquid and gaseous waste and airborne contaminants.
- Counters to measure gross beta and gamma activity.
- A low background counter to detect and measure gross alpha activity.
- A liquid scintillation counter to measure tritium in liquid and gaseous samples.
- A whole-body counter to detect and quantify personnel intakes of radioactivity.

- Fixed instrumentation, such as small article monitors, hand and foot monitors, and portal monitors, to monitor for contamination on personnel, materials, and equipment.

Portable Monitoring Instrumentation and Equipment (nominal ranges are given in parentheses for illustrative purposes only)

- Beta-gamma count rate survey meters (0-50,000 cpm) to detect radioactive contamination on surfaces and for low level exposure rate measurements.
- Low-range (0-50 mR/hr) and high range (0-1,000 R/hr) beta-gamma survey meters and ion chamber survey meters (0-50 R/hr) are used to measure the full range of dose rates necessary for radiation protection purposes during routine operations (including underwater operations), abnormal occurrences and accidents.
- Beta-gamma survey meters (0-10,000 R/hr) to monitor the plant and environs during and following an accident.
- Count rate meters (0-500,000 cpm) to monitor directly for alpha activity.
- Neutron survey instruments (0-5 rem/hr) to measure neutron dose rates for radiation protection purposes (including underwater operations).
- High and low volume air samplers equipped with appropriate filter media are used to take grab samples that are analyzed to assess airborne radioactivity concentrations, estimate actual or potential exposure, and to determine respiratory protection measures.
- Continuous air monitors (CAMs) provide a means to observe trends in airborne radioactivity concentrations. CAMs equipped with local alarm capability are used in occupied areas where needed to alert personnel to sudden changes in airborne radioactivity concentrations.
- Hand-held friskers to detect radioactive contamination.
- Portable air sampling and analysis system to determine airborne radioiodine concentrations during and following an accident consistent with the criteria in NUREG-0737, Item III.D.3.3.
- Portable sampling and onsite analysis capability to assess airborne radiohalogens and particulates released during and following an accident consistent with the criteria in Regulatory Guide 1.97 (and guidance provided in Branch Technical Position 7-10, Revision 5 to NUREG 0800).

Personnel Monitoring Instrumentation and Equipment

- Individual personnel dosimeters to measure gamma, beta and neutron radiation dose with sensitivities and ranges appropriate to measure the expected levels and types of radiation.
- Direct-reading dosimeters to provide real-time gamma dose information with sensitivities and ranges appropriate to measure the expected levels and types of radiation.
- Special dosimeters to monitor extremity dose with sensitivities and ranges appropriate to measure the expected levels and types of radiation.
- Personnel air samplers to monitor individual exposure to airborne radioactivity.
- Remote and local reading alarm dosimeters (which may be coupled with direct or electronic surveillance equipment, as necessary).

12AA.5.3.2 Personnel Protective Clothing and Equipment

A sufficient inventory of serviceable personnel protective clothing and equipment is maintained for use during plant operations, refueling and other outages, abnormal conditions, and accidents. Only respirators that are tested and certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA), or otherwise approved by the NRC, are used.

If circumstances arise in which NIOSH tested and certified respiratory equipment is not used, compliance with 10 CFR 20.1703(b) and 20.1705 is maintained.

Personnel protective clothing and equipment includes the following:

- Anti-contamination clothing for both dry and wet work conditions, including heat stress reduction accessories
- Head covers, shoe covers, gloves, and safety-related items
- Full facemask respirators with high-efficiency particulate and charcoal filters
- Pressure demand full facemask air line respirators
- Pressure demand full facemask self-contained breathing apparatus

12AA.5.3.3 Other Protective Equipment

- Portable ventilation systems with HEPA filters
- Temporary containments, tents, and enclosures
- Heat-stress reduction equipment
- Vacuums with HEPA filters
- Portable liquid filtration equipment
- Temporary shielding such as lead and/or tungsten shield bricks, blankets, and curtains.

12AA.5.4 Procedures

Radiation protection procedures are established, implemented and maintained sufficient to provide adequate control over the receipt, possession, use, transfer, and disposal of byproduct, source, and special nuclear material and assure compliance with applicable requirements in 10 CFR Parts 19, 20, 50, 70, and 71. Procedures for radiation protection that include quality assurance requirements are prepared consistent with the guidance in Regulatory Guides 1.8, 8.2, 8.7, 8.8 and 8.10 and the consolidated guidance referenced in NUREG-1736 that is applicable to power reactors. The procedures are implemented by Radiation Protection staff trained and qualified in accordance with the requirements in 10 CFR 50.120 and consistent with the guidance in Regulatory Guide 1.8. Additionally, some procedures are implemented by plant staff trained in accordance with the requirements of 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 8.13, 8.27, and 8.29.

12AA.5.4.1 Radiological Surveillance

Radiological surveillance procedures comply with 10 CFR 20.1501 and are consistent with the guidance in Regulatory Guides 8.2, 8.8, and 8.10.

Trained and qualified radiation protection staff will routinely survey accessible areas in the plant and environs to assess the presence and levels of radiation, radioactive contamination, and airborne radioactivity. The instrumentation and techniques used for these surveys are selected based upon the purpose of the survey and the anticipated types and levels of radiation and radioactivity involved. Surveys are performed using effective practices to minimize personnel exposure and avoid the spread of contamination.

The frequency and extent of the surveys will depend upon several factors, such as location, actual or potential radiation levels, plant operational status and work in progress, and accessibility/occupancy. The frequency of surveys may be weekly, monthly, quarterly, semiannually, annually, or as directed by the Radiation Protection Manager. Surveys are performed more frequently in accessible areas subject to changes in radiological conditions. Site specific procedures will define the survey frequencies and extent.

Survey results are recorded and maintained in accordance with the requirements in 10 CFR Part 20. Survey results for accessible areas are posted or otherwise made available to provide adequate notice to workers of radiological conditions.

Radiation surveys are routinely performed for detection of beta and gamma radiation. Surveys for neutron radiation are performed in accessible areas where such radiation may be present.

Area contamination surveys are routinely performed for the detection of removable and fixed beta-gamma contamination. Surveys for alpha contamination are performed where alpha contamination is anticipated. Alpha contamination surveys will also be performed periodically as a check to verify that alpha contamination is not present. Personnel will monitor themselves for contamination after exiting from contaminated areas and at exit points from the RCA or other Restricted Areas with a potential for contamination. Materials and equipment are monitored for contamination after removal from contaminated areas and prior to being released from the RCA or other Restricted Areas with a potential for contamination.

Surveys to assess airborne radioactivity levels are performed with continuous air monitors (CAMs) and by taking grab samples (using portable low or high volume air samplers) with appropriate media for collecting particulate, iodine, gas, or tritium samples. In order to warn personnel of changing airborne conditions, CAM alarm set points are set at a fraction of the concentration values given in 10 CFR Part 20, Appendix B, Table 1, Column 3, for radionuclides expected to be encountered. Air monitoring and sampling are sufficient to identify the potential hazard(s), determine the need for and verify the effectiveness of process and engineering controls, permit proper selection of respiratory protection equipment, and estimate doses from intakes.

Emergency operating procedures include provisions for use of a portable monitoring system, consistent with the criteria in NUREG-0737, Item III.D.3.3, to sample and analyze for radioiodine in areas of the plant during and following an accident. Procedures include methods for taking and analyzing samples in the field, as well as for analyzing samples in the count room facility, accounting for techniques to reduce counting system saturation from a high-activity sample.

Instrumentation and equipment used to perform surveys are calibrated prior to initial use, after performance of maintenance or repairs that might affect the calibration, and at least annually. Operational checks to test function or response are made daily for continuously operating instrumentation and equipment (e.g., friskers, portal monitors, and continuous air monitors) and prior

to use or daily, whichever is less frequent, for other instrumentation and equipment. Operational checks are performed for emergency and special use instrumentation and equipment on a regular schedule as specified in written procedures.

Survey records and records of calibration and maintenance of instrumentation and equipment used for surveys are documented and maintained in accordance with applicable requirements in 10 CFR 20.2101-20.2110.

12AA.5.4.2 Methods to Maintain Exposures ALARA

Methods to maintain exposures ALARA in accordance with Regulatory Guides 8.8 and 8.10 are included in radiation protection procedures, as well as applicable operating and maintenance procedures. Key ALARA operational policies and considerations are described in FSAR Section 12.1. Some examples of the types of methods that will be used to maintain exposures ALARA are discussed below for the following operational categories.

Refueling

After the reactor coolant system is depressurized, it is degassed as needed and sampled to verify that the gaseous radioactivity is low, prior to removing the reactor head. The Radiation Work Permit (RWP) system is used to maintain positive radiological control over work in progress. Prior to and during refueling, the refueling pool water is continually purified in order to maintain exposures from activity in the water ALARA. During refueling operations, irradiated fuel assemblies are maintained underwater at all times. By following these procedures, exposures from refueling operations are maintained ALARA.

Inservice Inspection

Prior to entry into radiation areas to perform inspections, personnel should study, as appropriate: blueprints, drawings, photographs, videotapes, previous inspection reports, previous radiation and contamination surveys, and/or previous RWPs appropriate to the particular inspection/job to be performed. This will acquaint personnel with the inspection location, room layout and equipment configuration, the work to be done, and radiation and contamination levels previously experienced at the location. Surveys are performed to the extent required to determine current contamination and/or radiation levels. From this data, previous data, and past work experience of personnel for similar jobs/inspections performed, an RWP (paragraph 12.5.4.5) is issued. Equipment is checked and/or calibrated to verify it is operating properly prior to entry into the radiation area. Temporary shielding will be used, where practicable, to reduce personnel radiation exposures.

Routine Maintenance

Routine maintenance is comprised of the categories of preventive maintenance (planned and scheduled maintenance such as lubrication, adjustments, and tests) and corrective maintenance (unscheduled maintenance such as valve packing, pump seal replacement, and stopping leaks). Procedures are usually written for preventive maintenance jobs and for some recurring corrective maintenance jobs. These procedures specify the precautions to be taken to minimize personnel exposure while performing the maintenance. The procedures list the required lubricants, special tools and equipment, and the acceptance standards. This serves to minimize the time spent in the radiation area and thereby minimize personnel dose.

In addition, the preventive maintenance procedure normally states whether an RWP is required. When the RWP is issued, the radiation and/or contamination levels are listed, shielding is specified, if appropriate, and additional specific instructions are given to personnel. For corrective maintenance jobs in radiation areas, a similar approach is used.

Extension tools are used when practical to minimize personnel dose when working on radioactive components/equipment. Detailed surveys are performed and the RWP is issued (if required) with specific instructions. The individuals performing the work may be required to read procedure manuals or may be shown pictures or sketches of the work area to aid in understanding what is to be accomplished, how it is to be accomplished as safely and quickly as possible, and what the acceptance criteria are for completing the job. At the discretion of health physics personnel, additional requirements may be imposed to reduce personnel exposures.

After the job is completed, debriefings may be conducted to obtain input from personnel actually performing the work, as well as from supervisory and support personnel. This will assist in revising procedures for ALARA considerations.

Calibration

Calibration of most ranges of the portable gamma detection instruments is performed inside a shielded calibrator, thereby eliminating a large portion of the exposure received from calibration of portable instruments. Portable sources used to calibrate fixed instruments (such as the area radiation monitoring system) are transported in shielded containers to minimize personnel exposure.

Where possible, fixed instruments requiring routine calibration are situated so that the necessary test signals needed for calibration can be inserted from a low radiation area with the instruments in place.

12AA.5.4.3 Posting and Labeling

Procedures for posting and labeling will assure compliance with 10 CFR 20.1901, 20.1902, 20.1903, 20.1904, and 20.1905.

Based on current survey results, Radiation Areas, High Radiation Areas, Very High Radiation Areas, Airborne Radioactivity Areas, and Radioactive Materials Areas are posted in accordance with the requirements in 10 CFR 20.1901, 20.1902, and 20.1903. Containers of licensed radioactive materials are labeled in accordance with 10 CFR 20.1904 and 20.1905.

Criteria and procedures are established for posting areas and marking items (e.g., tools and equipment) to indicate the presence of fixed or removable surface contamination. Areas posted to indicate the presence of removable contamination, are referred to hereafter as “Contamination Areas.”

“Posted areas”, as used in Section 12.5, refers to Radiation Areas, High Radiation Areas, Very High Radiation Areas, Airborne Radioactivity Areas, Contamination Areas, and Radioactive Materials Areas.

12AA.5.4.4 Access Control

Procedures for access control will assure compliance with 10 CFR 20.1902, 20.1903, 20.1601, and 20.1602 and are consistent with the guidance in Regulatory Guide 8.38.

Access to posted areas is restricted and controlled, at a minimum, through the use of instructions to workers, radiation work permits, caution signs, and barriers. Access to High and Very High Radiation Areas is controlled consistent with the guidance in Regulatory Guide 8.38, including the use of alternative methods for access control as described in the regulatory guide and specified in plant technical specifications.

Table 12AA-201 identifies plant areas designated as Very High Radiation Areas (VHRAs), lists corresponding plant layout drawings showing the VHRA in Section 12.3, specifies the condition under which the area is designated VHRA, identifies the primary source of the VHRA, and

summarizes the frequency of access and reason for access. VHRAs are listed as Radiation Zone IX, which corresponds to a dose rate greater than 500 rad/hr.

In each of the VHRAs, with the exception of the Reactor Vessel Cavity and Delay-Bed / Guard-Bed Compartment, the primary radioactive source is transient (such as fuel passing through the transfer tube), removable (such as resin in the demineralizers), or can be relocated. When the primary source is removed, the dose rate in each of these areas will be less than Zone IX and, in effect, the area will no longer be a VHRA. With planning, the need for human entrance to a VHRA when the primary source is present can be largely or entirely avoided.

In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in one hour at one meter from a radiation source or any surface through which the radiation penetrates:

- Sign(s) conspicuously posted stating GRAVE DANGER, VERY HIGH RADIATION AREA.
- Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the functional manager in charge of radiation protection as described in Section 13.1.
- Plant Manager's (or designee) approval required for entry.
- Radiation Protection personnel shall accompany person(s) making the entry. Radiation Protection personnel shall assess the radiation exposure conditions at the time of the entry.

A verification walk down will be performed with the purpose of verifying barriers to the Very High Radiation Areas in the final design of the facility are consistent with Regulatory Guide 8.38 guidance as part of the implementation of the Radiation Protection and ALARA programs on the schedule identified in Table 13.4-201.

Unescorted access to Radiation Areas or Radioactive Materials Areas will require, at a minimum, authorization by Radiation Protection, the use of an RWP, and instruction of individuals gaining unescorted access in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guide 8.13. In addition to the foregoing, unescorted access to Contamination, High Radiation, Very High Radiation, or Airborne Radioactivity Areas will require, at a minimum, training of individuals gaining unescorted access consistent with Regulatory Guides 8.27 and 8.29.

Posted areas will generally be contained within the plant Security Area, i.e., an area to which access is controlled in accordance with 10 CFR Part 73. Unescorted access to the plant Security Area will require instruction of individuals gaining unescorted access in accordance with 10 CFR 19.12.

Areas where significant doses could be received (e.g., High Radiation, Very High Radiation, and Airborne Radioactivity Areas), are generally contained within the plant building complex. A Radiological Controlled Area (RCA) is established to encompass the plant building complex to enhance control over access to such areas. Access to the RCA is through a primary access control point or alternate access control points as established by Radiation Protection. Unescorted access to the RCA will require authorization by Radiation Protection, the use of an RWP and instruction and training of individuals gaining unrestricted access in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 8.13, 8.27, and 8.29.

Radiation Protection may authorize access to the Security Area, RCA, or a Radiation or Radioactive Materials Area for individuals without instruction or training where such individuals are continuously under the control of a designated escort. The designated escort shall be instructed and trained in accordance with the requirements of 10 CFR 19.12 and the guidance in Regulatory Guides 8.13, 8.27, and 8.29, and shall be instructed on the duties and responsibilities associated with being an escort.

Access by a worker who is a minor (i.e., under the age of 18 years) or a declared pregnant worker to posted areas with a potential for significant exposure, e.g., High Radiation, Very High Radiation, and Airborne Radioactivity Areas is restricted unless otherwise authorized by Radiation Protection.

12AA.5.4.5 Radiation Work Permits

Procedures covering the use of a radiation work permit (RWP) are consistent with the guidance in Regulatory Guide 8.8.

RWPs are issued by Radiation Protection to help ensure adequate protection of personnel for access to and work within areas with a potential for significant exposure. Access to any posted area will require an RWP. An RWP may control access to multiple areas or to a set of related jobs or tasks.

At a minimum, each RWP will include the following information:

- Description of the area(s) to be accessed and work to be performed;
- Designation of personnel or groups covered by the RWP;
- Radiological conditions existing within the area(s) to be accessed, based on current radiological surveys, and anticipated radiological conditions for the time span over which the work is performed (including location of hot spots, radiation gradients, and low dose “waiting areas”);
- Requirements for use of personnel monitoring devices, protective clothing, and respiratory protection equipment;
- Special instructions and a description of special tools, shielding, other equipment utilized to perform work, and any process and engineering controls being employed to minimize exposures; and
- Extent and type of radiation protection monitoring and surveillance to be provided.

As described in Section 12.1, for access to and work within High Radiation and Very High Radiation Areas, the applicable RWP will specify a limitation on staytime or a means for limiting dose received while in the area (e.g., via an alarm set point for an electronic dosimeter).

12AA.5.4.6 Personnel Monitoring

Personnel monitoring procedures are sufficient to assure compliance with 10 CFR Parts 19 and 20 and are consistent with the guidance in Regulatory Guides 8.2, 8.7, 8.9, 8.13, 8.34, 8.35, and 8.36.

Each individual accessing the RCA or a posted area on an unescorted basis, or for whom occupational dose monitoring of external dose is required in accordance with 10 CFR 20, is monitored using an individual monitoring device that is appropriate for monitoring the types of external radiation to which the individual is exposed. For individuals who are required to be monitored in accordance with 10 CFR Part 20, if the individual monitoring device does not provide real-time

dose information (i.e., the capability for the individual to track his or her own dose as it occurs), then an additional means of monitoring is provided for the individual that fulfills that function.

Individuals accessing the RCA or a posted area on an escorted basis, for whom occupational dose monitoring of external dose is not required in accordance with 10 CFR Part 20, are monitored either with an individual monitoring device worn by the individual or via an individual monitoring device worn by the escort.

Individual monitoring devices that require processing, except for those devices excluded by 10 CFR 20.1501(c), are processed and evaluated by a NVLAP-accredited processor, as appropriate, for the type(s) and ranges of radiation being monitored with the device. Each individual whose internal dose is required to be monitored in accordance with 10 CFR Part 20, or who wears a respirator for radiation protection purposes, or who accesses an Airborne Radioactivity Area, is monitored by means sufficient to identify and quantify intakes in order to be able to estimate his or her committed effective dose equivalent (CEDE) and, as applicable, his or her committed dose equivalent (CDE).

Situations that may result in a person receiving an abnormal or inadvertent intake are evaluated on a case-by-case basis to determine the need for monitoring by means sufficient to identify and quantify intakes in order to be able to estimate the CEDE or CDE, as applicable.

Individuals suspected of having received an intake are evaluated to quantify the intake, if any, in order to estimate the CEDE or CDE, as applicable. In demonstrating compliance with regulatory requirements, effective dose equivalent may be used in lieu of deep dose equivalent consistent with the guidance in Regulatory Issue Summary (RIS) 2003-04 and other related guidance.

Individual monitoring results are reported annually to the individual, and at the request of an individual who is terminating employment or who is requesting this information from a previous employer, in accordance with the requirements in 10 CFR 19.13.

Personnel monitoring records, as well as records associated with testing, calibration, processing, and maintaining instrumentation and equipment used for personnel monitoring, are documented and maintained in accordance with applicable requirements in 10 CFR 20-2101-20.2110.

12AA.5.4.7 Dose Control

Compliance is maintained with the requirements in 10 CFR 20.1201, 20.1202, 20.1203, and 20.1204, as they relate to demonstrating compliance with internal and external occupational dose limits contained in 10 CFR 20, Subpart C. Doses to adult workers are kept below the occupational dose limits in 10 CFR 20.1201. Doses to workers who are minors and declared pregnant workers are kept below the respective occupational dose limits in 10 CFR 20.1207 and 10 CFR 20.1208. Doses to members of the public are kept below public dose limits in 10 CFR 20.1301, which is demonstrated by complying with the requirements of 10 CFR 20.1302.

To the extent practical, procedures and engineered controls based on sound radiation protection principles are used to keep occupational doses and doses to members of the public as low as is reasonably achievable (ALARA). A description of facility design features and engineered controls intended to maintain occupational exposures ALARA is included in Sections 12.3-12.4. A description of systems and facility design features intended to maintain public exposures ALARA is included in Chapter 11.

As described in Sections 12.1, Appendix 12AA and 13.1, management policy is established, and organizational responsibilities and authorities are assigned to implement an effective program for maintaining occupational radiation exposures ALARA. Procedures are established and implemented

that are in accordance with 10 CFR 20.1101 and consistent with the guidance in Regulatory Guides 8.8 and 8.10. Examples of such procedures include the following:

I. During the construction, pre-operational and operational phases, Radiation Protection will assure that new or modified designs and the selection of equipment are reviewed to assure that measures are considered to minimize occupational and public radiation exposures during operation, refueling, and decommissioning of the plant.

II. Radiation Protection will assure that procedures and methods for operation, maintenance, repair, surveillance, refueling, and other activities that may involve significant exposures are reviewed prior to initial use and periodically thereafter that assure measures are considered to minimize occupational and public radiation exposures. For example, "significant exposures" may include activities that are estimated to involve greater than 1 person-rem of collective dose.

III. For activities involving significant exposures, pre-job briefings are conducted for personnel who will receive the exposures. The briefings are intended to assure that personnel understand the radiological conditions expected to be present and the measures being employed to control and minimize dose. Post-job reviews are performed to evaluate the effectiveness of measures employed to control and minimize dose and to identify and implement improvements to minimize occupational and/or public radiation exposures for future similar activities.

Planned special exposures, as described in 10 CFR 20.1206, if used, will be conducted in accordance with the requirements in 10 CFR 20.2104 and consistent with the guidance in Regulatory Guide 8.35.

12AA.5.4.8 Contamination Control

Contamination control procedures are established to help assure compliance with 10 CFR Parts 20.1406 and 20.1701 and to prevent the unauthorized release of radioactive materials to unrestricted areas.

Areas, items, and personnel are routinely surveyed and monitored for contamination to protect personnel, ensure that contamination control methods are effective and to prevent licensed materials from being released from an RCA or Controlled Area in an unauthorized manner. Areas and items with fixed or removable contamination are posted, labeled, or marked in a conspicuous manner to indicate the presence of contamination.

Personnel accessing Contamination or Airborne Radioactivity Areas are required to use protective clothing and equipment appropriate to the circumstances to prevent personal contamination.

Personnel found with external contamination are decontaminated promptly. Contaminated items are decontaminated or disposed of as radioactive waste or are marked and controlled. Areas that become contaminated are decontaminated as soon and as thoroughly as practical, taking into account factors such as the nature of operations in the area and the potential for exposure associated with the decontamination. The number of accessible contaminated areas within the plant are kept to a minimum.

Facility design and operational procedures are reviewed to identify nonradioactive systems that could possibly become radioactive through interfaces with radioactive systems. Routine sampling and monitoring of these systems is described in the plant radiation monitoring program, and overall guidance is consistent with Bulletin 80-10.

Practical measures are implemented to prevent the spread of contamination, including:

- Air pressure gradients and airflows are maintained from areas of low potential contamination to areas of higher potential contamination and then to installed filters and/or ventilation systems;
- Leaks and spills are contained promptly and repaired or cleaned up as soon as practical;
- Potentially contaminated systems, equipment, and components are surveyed for the presence of contamination when opened or prior to removal;
- Containments, caches and enclosures are used during maintenance, repairs, and testing, when practical, to contain spills or releases;
- Engineering controls, such as portable ventilation or filtration units to reduce concentrations of radioactivity in air or fluids, are used where practical;
- Criteria for selecting tools, materials, and equipment for use in contaminated areas will include minimizing the use of porous or other materials that are difficult to decontaminate;
- The use of disposable materials that are likely to become contaminated and necessitate disposal as radioactive waste are minimized;
- Areas, surfaces, and tools that are prone to contamination are designed and coated (e.g., using agents to “fix” contamination, such as strippable coatings), as practical, to facilitate decontamination;
- Contaminated tools and equipment are segregated from clean tools and equipment.

This subsection adopts NEI 08-08A (Reference 201), for a description of the operational and programmatic elements and controls that minimize contamination of the facility, site, and the environment, to meet the requirements of 10 CFR 20.1406.

12AA.5.4.9 Respiratory Protection

Respiratory protection procedures will assure compliance with 10 CFR Part 20, Subpart H, and are consistent with the guidance in Regulatory Guide 8.15.

A written policy statement established by the plant management covers the use of process and engineering controls in lieu of respirator use to limit intakes and to limit the routine, non-routine, and emergency use of respirators.

Written procedures are established and implemented that cover the following:

- Monitoring, including air sampling and bioassays;
- Supervision and training of respirator users;
- Fit-testing;
- Respirator selection;
- Breathing air quality;

- Inventory, control, storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment;
- Recordkeeping; and
- Limitations on periods of use and relief from respirator use.

An assessment is performed to assure that the total effective dose equivalent (TEDE) is maintained ALARA, when respiratory protection equipment is used to limit intakes of radioactive materials.

Airborne radioactivity is minimized by the design and configuration of the plant's heating, ventilation and air conditioning systems (HVAC), the use of enclosures and containments, and good housekeeping practices. Portable air movers and vacuums equipped with HEPA filters to minimize concentrations of radioactivity in air or on surfaces are vented to monitored, filtered discharge pathways.

When it is not practical to apply process and engineering controls to control the concentrations of radioactive materials in the air and maintain the TEDE ALARA, intakes are limited by controlling access to and limiting stay times in Airborne Radioactivity Areas and by using respiratory protection equipment or other controls.

The Radiation Protection Manager will assign to a single individual, knowledgeable in the area of respiratory protection consistent with the guidance in Regulatory Guide 8.15, the overall responsibility to establish and maintain a respiratory protection program and procedures that include:

- air sampling and monitoring sufficient to identify hazards, select proper equipment, and determine doses from intakes;
- conducting surveys and bioassays as necessary to evaluate actual intakes; and
- testing respirators for operability immediately prior to each use.

Only respiratory protection equipment that is tested and certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) is used, unless otherwise authorized by the NRC.

Prior to being fit-tested for a face sealing respirator, and before the first field use of a non-face sealing respirator, individuals are certified as medically fit by a qualified medical practitioner. Recertification of medical fitness is made every twelve months or at a frequency specified by the medical practitioner.

Each respirator user is advised that he or she may leave the area at any time for relief from any conditions (such as equipment malfunction, physical or psychological distress, or communications failure) that might require such relief.

In selecting and using respiratory protection equipment, provisions are made for vision correction, adequate communications, extreme temperature conditions, and concurrent use of other safety or radiological protection equipment.

For circumstances when respiratory protection equipment is used from which an unaided individual would have difficulty extricating himself or herself, and therefore might be exposed to a potentially life-threatening situation, a standby rescue person is required. The standby rescue person shall be

equipped with respiratory protection equipment appropriate for the potential hazards and shall be immediately available to provide assistance.

12AA.5.4.10 Radioactive Material Control

Procedures are established, implemented and maintained that assure compliance with the requirements of 10 CFR 20.1801, 20.1802, 20.1902, 20.1904, 20.1905, 20.1906, 20.2001, 20.2005, 20.2006, 20.2007, 20.2201, and 10 CFR 71.5 to assure positive control over licensed radioactive material so that unnecessary or inadvertent exposures do not occur and such material is not released into uncontrolled areas in a manner that is not authorized by regulation or the license.

12AA.5.4.11 Radiation Protection Training

Procedures are developed, implemented, and maintained that assure that selection, qualification, training, and periodic retraining of radiation protection staff and radiation workers are conducted in accordance with the requirements in 10 CFR Parts 19, 20, and 10 CFR 50.120 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.15, 8.27, and 8.29.

12AA.5.4.12 Quality Assurance

The radiation protection program and procedures are established, implemented, maintained and reviewed consistent with the 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description described in Section 17.5.

Consistent with the requirements in 10 CFR 71.101(f), quality assurance requirements apply to the program, procedures and activities involving the transportation of radioactive material.

12AA.5.4.13 Reports

Procedures are established, implemented, and maintained to assure that reports and notifications are made in accordance with 10 CFR 20, Subpart M.

12AA.5.4.14 Groundwater Monitoring Program

A groundwater monitoring program beyond the normal radioactive effluent monitoring program is developed. If necessary to support this groundwater monitoring program, design features will be installed during the plant construction process. Areas of the site to be specifically considered in this groundwater monitoring program are (all directions based on plant standard):

- West of the auxiliary building in the area of the fuel transfer canal.
- West and south of the radwaste building.
- East of the auxiliary building rail bay and the radwaste building truck doors.

This subsection adopts NEI 08-08A (Reference 201), for the Groundwater Monitoring Program description.

12AA.5.4.15 Record of Operational Events of Interest for Decommissioning

This subsection adopts NEI 08-08A (Reference 201), for discussion of record keeping practices important to decomissioning.

12AA.5.5 References

1. 10 CFR Part 19, "Notices Instructions, and Reports to Workers: Inspections and Investigations."

2. 10 CFR Part 20, "Standards for Protection Against Radiation."

3. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

4. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

5. 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

6. 10 CFR Part 73, "Physical Protection of Plants and Materials"

7. Regulatory Guide 1.8, Revision 3, "Qualification and Training of Personnel for Nuclear Power Plants."

8. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

9. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

10. Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring."

11. Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters."

12. Regulatory Guide 8.6, "Standard Test Procedures for G-M Counters."

13. Regulatory Guide 8.7, Revision 2, "Instructions for Recording and Reporting Occupational Radiation Exposure Data."

14. Regulatory Guide 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable."

15. Regulatory Guide 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

16. Regulatory Guide 8.10, Revision 1R, "Operational Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable."

17. Regulatory Guide 8.13, Revision 3, "Instruction Concerning Prenatal Radiation Exposure."

18. Regulatory Guide 8.15, Revision 1, "Acceptable Programs for Respiratory Protection."

19. Regulatory Guide 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants."

20. Regulatory Guide 8.28, "Audible Alarm Dosimeters."

21. Regulatory Guide 8.29, Revision 1, “Instruction Concerning Risks from Occupational Radiation Exposure.”

22. Regulatory Guide 8.34, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses.”

23. Regulatory Guide 8.35, “Planned Special Exposures.”

24. Regulatory Guide 8.36, “Radiation Doses to Embryo/Fetus.”

25. Regulatory Guide 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas of Nuclear Power Plants.”

26. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

27. NUREG-1736, “Consolidated Guidance: 10 CFR Part 20 Standards For Protection Against Radiation.”

28. Regulatory Issue Summary 2003-04, “Use of the Effective Dose Equivalent In Place of the Deep Dose Equivalent in Dose Assessments.”

29. SRP Branch Technical Position (BTP) 7-10, “Guidance on Application of Regulatory Guide 1.97,” NUREG-0800.

201. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

Table 12AA-201 (Sheet 1 of 2)
Very High Radiation Areas (VHRA)

Room Number	VHRA Location	Figure 12.3-1, Sheet No.	Primary Source(s)	VHRA Conditional Notes	Frequency of Access to VHRA Areas While VHRA Conditions Exist
11105	Reactor Vessel Cavity	3, 4, 5	Neutron activation of the material in and around the cavity during reactor operations, such as the concrete shield walls and the reactor insulation	Note 1	None Required
12151	Spent Fuel Pool Cooling System / Liquid Radwaste System Demineralizer/ Filter room (Inside Wall)	3	Resin in vessels	Notes 6, 8	None Required
12153	Delay-Bed/ Guard-Bed Compartment	3	Activated carbon holding radioactive gases	Note 10	None Required
12371	Filter-Storage Area	6, 7	Spent filter cartridges	Notes 4, 6, 7	None required
12372	Resin Transfer Pump/Valve Room	6	Spent resin in lines	Note 6	None required
12373	Spent-Resin Tank Room	6	Spent resin in tanks	Note 6	None Required
12374	Waste Disposal Container Area	6	Spent resin in vault	Note 6	None Required
12463	Cask Loading Pit	6	Spent fuel	Notes 2, 6	None Required
12563	Spent Fuel Pit	5, 6	Spent fuel	Note 6	None Required
Fuel Transfer Areas					
12564	Fuel Transfer Tube	6	Fuel in transit	Notes 2, 5, 9	None Required
11205	Reactor Vessel Nozzle Area	5	Fuel in transit	Notes 2, 3, 9	None Required
11504	Refueling Cavity	6	Fuel in transit	Notes 2, 3, 9	None Required

Table 12AA-201 (Sheet 2 of 2)
Very High Radiation Areas (VHRA)

Notes

1. VHRA during full power operation; less than 10 Rem/hr 24 hours after plant shutdown.
2. During underwater spent fuel transfer operations, this area can be as high as VHRA.
3. During underwater reactor internals transfers/ storage, this area can be as high as VHRA.
4. During spent resin waste disposal container transfer or loading, this area can be as high as VHRA. The contact dose rate of spent resin containers can be greater than 1000 Rem/hr.
5. Discussion about the Spent Fuel Transfer Canal and Tube Shielding is provided in Subsection 12.3.2.2.9.
6. Source is transient, removable, or can be relocated.
7. VHRA when hatch is removed during spent resin container handling operation.
8. In the event that the room does need to be accessed for maintenance or other reasons, temporary shielding is put in place and the resin is removed from the vessels. These measures reduce exposure rates in the room, such that this room is no longer a VHRA. Remote handling is used for any tasks that require the opening of the access hatch in the ceiling of this room when media is present.
9. These areas have no planned reasons for entry and are only classified as VHRAs during periods of fuel movement. In the event that these rooms do need to be accessed to repair the Fuel-Transfer System, Fuel Transfer Tube Gate Valve, or other components, it is done during a non-fuel movement time. This keeps the dose received by the worker as low as reasonably achievable.
10. Inspection of the equipment in this room, when required, is done using remote viewing equipment. Two plugs between Room 12153 and 12155 contain instruments and the plugs are expected to be removed every 12 to 18 months for performance of maintenance. Administrative procedures are implemented to protect workers pursuant to Regulatory Guide 8.38.