12.3 Radiation Protection Design Features

This section describes specific U.S. EPR design features for maintaining personnel exposures as low as reasonably achievable (ALARA). These features include facility-specific features, shielding, ventilation, radioactivity monitoring systems, and contamination control. Also presented in this section is a projected collective dose assessment for the U.S. EPR.

12.3.1 Facility Design Features

Occupational and offsite external radiation exposures are maintained ALARA, in compliance with 10 CFR 20.1201 and 40 CFR 190. The U.S. EPR facility layout uses a compartmental approach to maintain personnel exposures ALARA. The general guidance on which this design is based is presented in Section 12.1.2. Specific design features and illustrative examples follow and show how this guidance is employed in the design.

12.3.1.1 Reactor Building

The Reactor Building is a cylindrical building that is in the middle of the common basemat in the Nuclear Island (NI). The Reactor Building is an integrated structure consisting of an inner Containment Building, an outer Shield Building, and an annular space between the two buildings that separates them. The Containment Building is divided into two compartments: an inner equipment compartment and an outer service compartment. The inner compartment contains the steam generators (SG), reactor coolant pumps (RCP), and primary loop piping. The outer compartment houses support equipment. Shielding is provided within each room or compartment to shield components from one another. The reactor vessel is shielded to reduce streaming from the reactor vessel annulus and the biological shield (see Figures 12.3-7 and 12.3-8).

The Reactor Building compartmental configuration reduces the dose rate to operators entering the service compartment. While the operators do not routinely enter the building at power, the compartmental design provides shielding to personnel staging equipment for an outage during the last few days of power operation. This shielding reduces the cumulative dose to workers during the outage.

Construction of the Containment Building and the Shield Building is described in Section 6.2. The annulus between the Containment Building and the Shield Building captures effluent leakage from the Containment Building for filtration prior to its release to the environment. This annular space and the associated ESF filters mitigate the radiological consequences from a design basis accident.

Components may enter the radiological controlled area through an equipment hatch in the Reactor Building. The CVCS high-pressure cooler rooms have removable shielding to enable replacement of the letdown high-pressure coolers and are provided sufficient laydown space for special tools around the component for ease of maintenance.

12.3.1.2 Safeguard Building

The Safeguard Building surrounds the west, north, and east quadrants of the Reactor Building and is on the common NI basemat. The Safeguard Building consists of four separate and independent divisions, each containing a complete train of safeguard equipment needed to mitigate an accident. If one safeguard train is out of commission, three other safeguard trains remain active, thus eliminating the need for immediate repair during accident conditions.

Each division is further divided into two areas: the radiological controlled area (consisting of safety injection and vent and drain systems), and the uncontrolled area (containing instrumentation, control equipment, and switchgear). The systems that contain radiation sources are placed closest to the Reactor Building in the bottom two floors. For instance, the most significant source is the low head safety injection heat exchanger in each division, at elevation -16 feet. This arrangement minimizes the piping to the reactor coolant loop and also provides additional shielding between the radiological sources and the outside environment. Each of the areas is also served by a separate ventilation system. The ventilation system is divided into two trains of equipment to separately serve the controlled and uncontrolled areas. An additional shield wall surrounding the Safeguard Building results in dose rates outside of the external walls of the building below 1 mrem/hr.

12.3.1.3 Fuel Building

The Fuel Building surrounds the south quadrant of the Reactor Building and is on the common NI basemat. The Fuel Building contains the spent fuel pool, fuel handling equipment, portions of the CVCS, portions of the boric acid recovery system, the fuel pool cooling and purification system, and a dedicated room for decontaminating RCPs for maintenance. The Fuel Building is divided into cells for ventilation purposes and to isolate components. Additional shielding is provided by segregating components into separate rooms. A loading hall is located in the Fuel Building; vehicles carrying new fuel enter into the radiological controlled area through this hall.

The roof of the Fuel Building is constructed of approximately 6-feet-thick concrete, and the walls that face the environment are constructed of approximately 8.5-feet-thick concrete. The bottom three levels of this building are below grade and no high radiation source components are adjacent to outside walls; this arrangement results in dose rates outside of the external walls of the building below 1 mrem/hr.

12.3.1.4 Nuclear Auxiliary Building

The Nuclear Auxiliary Building is adjacent to the east side of the Fuel Building and adjacent to the south side of Safeguard Building Division 4. This building is constructed on an individual basemat. The Nuclear Auxiliary Building is divided into three ventilation cells, with components segregated into separate rooms.

The Nuclear Auxiliary Building contains two radiation sources adjacent to outside walls: the coolant storage tanks and the gaseous waste processing system delay beds. These two radiation sources are the strongest sources in the building. The coolant storage tanks are shielded by roughly 2.5-feet-thick concrete walls to the outside environment and the delay beds by over 3-feet-thick concrete walls to the outside environment. These walls are sufficient to maintain the dose rate on the outside of the walls below 1 mrem/hr.

Radiochemistry Laboratory

The radiochemistry laboratory facilities are centrally located at elevation -31 feet of the Nuclear Auxiliary Building for receiving, storing, preparing, analyzing, and disposing of solid, liquid, and gaseous sample media. The laboratory and counting room facility and instrumentation are sufficiently shielded to maintain low background radiation levels to permit analysis of samples during routine and accident conditions.

The radiochemistry laboratory contains floor drains, a sink, a fume hood, a cabinet with worktop, a storage locker, and an emergency shower and eyewash system. Drains are piped to the liquid waste management system, and the fume hood exhausts to a monitored building ventilation exhaust system. This facility is shown in Figure 12.3-80—Radiochemistry Laboratory Facilities -31 Ft Elevation Nuclear Auxiliary Building.

Machine Shop for Activated or Contaminated Components and Equipment

A hot workshop, shown on Figure 12.3-50—Nuclear Auxiliary Building +64 Ft Elevation Radiation Zones, is provided for receiving, disassembling, repairing, and machining activated or contaminated components and equipment to control the spread of contamination and provide a low dose rate area for servicing. A tool store adjacent to the hot workshop is provided for the control, storage, issuance, and receipt of contaminated tools and equipment. This tool store helps to minimize the generation of radioactive waste and to control the spread of contamination.

12.3.1.5 Radioactive Waste Processing Building

The Radioactive Waste Processing Building is adjacent to the east side of the Nuclear Auxiliary Building. This building is constructed on an individual basemat. The

Radioactive Waste Processing Building is divided into three ventilation cells, with components segregated into separate rooms.

The Radioactive Waste Processing Building houses portions of the coolant purification system, the liquid waste management system, and the solid waste management system. The bottom three levels of this building are below grade. High radiation sources (resin waste tanks, the Group I liquid storage tanks, evaporator bottoms, centrifuge sludge, purification filters, and drum storage) are shielded to result in dose rates less than 1 mrem/hr immediately adjacent to the Radioactive Waste Processing Building.

12.3.1.6 Access Building

Access control facilities control the entrance and exit of personnel and materials into and from the radiologically controlled area (RCA) of the plant. [[Separate change areas for male and female personnel are located at the access control facility. These facilities are located at elevations -13 feet and 0 feet of the Access Building. The change areas are sufficiently sized to support routine operations, maintenance, and typical refueling outage conditions.

Radiation protection offices sufficient to support staff oversight of the radiological control program are located at elevation +39 feet of the Access Building. Space is provided for storage and issuance of radiation protection equipment, instrumentation, dosimetry, and supplies.

Access control facilities are shown in Figure 12.3-14—[[Access Building at -31 Ft Elevation Radiation Zones]] through Figure 12.3-20—[[Access Building at +54 Ft Elevation Radiation Zones]]

Personnel Decontamination Area

[[Once a worker has entered the RCA within the Access Building, entrance to the portions of the connecting buildings in the RCA is at elevation 0 feet, where the worker enters Safeguard Building Division 4. From there, the worker can follow a passageway around the Reactor Building and enter the Fuel Building and Nuclear Auxiliary Building or access other divisions of the Safeguard Building.

Personnel decontamination areas are located near the exit side of the primary access control facility at elevation 0 feet of the Access Building near the control point. The personnel decontamination area is supplied with sinks and showers with drains that are routed to the liquid waste management system.]]

Portable Instrument Calibration Facility

[[A portable instrument calibration facility is located at elevation 0 feet of the Access Building and is designed so that radiation fields created during calibrations do not unnecessarily expose personnel and do not interfere with low-level monitoring or counting systems. This facility is in a low-background radiation area so that ambient radiation fields from plant operation do not interfere with low-range instrument calibrations.]]

Respiratory Facility

[[A respirator facility is located with the laundry and consumables storage area at elevation 0 feet in the Access Building. Room is provided for respirator inspection, maintenance, repair, storage, inventory, control, and issuance.]]

Equipment Decontamination Facility

[[Decontamination and cleaning of personnel protective equipment, instrumentation, and small items are performed in a facility set up for that specific purpose at elevation 0 feet of the Access Building. The washdown area and sink drains are routed to the liquid waste management system, and positive air flow is maintained into the decontamination facility and exhausted into a monitored building ventilation system. The facility is provided with coated walls and floors to ease cleanup and decontamination.]]

Radioactive Materials Storage Area

[[A radioactive materials storage area is located at elevation 0 feet of the Access Building and provides for secure storage of calibration sources.]]

Facility for Dosimetry Processing and Bioassay

[[A bioassay room is located at elevation 0 feet of the Access Building outside of the radiological controlled area for dosimetry processing and bioassays collection. The facility is sufficiently shielded to maintain low background radiation levels.]]

12.3.1.7 Layout Design Features for ALARA

12.3.1.7.1 Isolation and Decontamination

Serviceable systems and components that constitute substantial radiation sources are designed with features that permit isolation and decontamination. The radioactive piping and associated equipment are isolated and drained for routine maintenance. As part of this design capability, the valves in radioactive system pipes that are 2.5 inches and larger are provided with leakoff connections piped directly to individual collection systems. A typical configuration is the CVCS letdown line. In this line, vent connections are provided above the piping centerline, the piping slopes between the connection from the RCS to the VCT, and drains are provided from the low points.

Certain systems have provisions for chemical and mechanical cleaning prior to maintenance. Flushing and recirculation features are provided so that workers can decontaminate portions of systems for maintenance. For example, the letdown portion of the CVCS includes quick-disconnect connections for flushing, draining, venting, and recirculation. These connections are provided before and after each high-pressure cooler that is located in a high radiation area on the elevation -8 feet of the Reactor Building.

Some plant conditions, such as loss of off-site power, require the cooldown to be performed with the SG main steam relief train (MSRT). The release points of the MSRTs are designed to minimize the potential contamination of drains or supply fan intakes, and minimize the contamination of neighboring structures.

12.3.1.7.2 Contamination Control

In addition to the compartmental design features incorporated into the U.S. EPR, a fuel pool cooling and purification system also provides contamination control. This system removes contaminants from the fuel pool water. Water is taken from the bottom of the pool, and the purified water is returned to the top of the pool. The system has skimmers on the pool to prevent corrosion products from settling on surfaces in the pool. This system is further described in Section 9.1.3.

Coatings such as sealers or special paint permit easy decontamination and are used on walls, ceilings, and floors where the potential for surface contamination exists. For example, the floors of the coolant purification valve rooms at elevation -21 feet of the Nuclear Auxiliary Building are coated. These rooms are shown in Figure 12.3-43—Nuclear Auxiliary Building -21 Ft Elevation Radiation Zones.

The contamination control features comply with 10 CFR 20.1406 (see Section 12.3.6.1). Additional information on administrative controls to prevent the spread of contamination is provided in the Radiation Protection Program (see Section 12.5).

12.3.1.7.3 Control of Airborne Contaminants and Gaseous Radiation Sources

The U.S. EPR design provides features in process, containment, and ventilation systems to protect workers from airborne radioactive material. Air pressure gradients direct air from low potential airborne contamination areas to areas of higher potential airborne contamination and then exhaust the air through filters. A typical example of this configuration occurs in the Nuclear Auxiliary Building. Supply air enters the building through the air intake shown in Figure 12.3-48—Nuclear Auxiliary Building +34 Ft Elevation Radiation Zones. Supply air is drawn into the building by the supply air fans shown in the bottom center of the drawing. The air is distributed to the various levels via the supply air shaft on each elevation, as shown in the bottom center of the figure.

On individual levels, the air is ducted to the service corridor or anterooms, where the air is introduced into a low potential airborne contaminated area. For example, at elevation +34 feet of the Nuclear Auxiliary Building, the supply air enters the maintenance floor, flows through the two delay bed rooms, and then flows through the gel drier room. The air is finally exhausted through the gaseous waste processing systems, through HEPA filters to the vent stack, and out to the environment. Thus, the air flows through the rooms from the maintenance floor (with a low potential for airborne contamination) to rooms containing gaseous waste processing system components (with a higher potential for airborne contamination) and is then exhausted.

These design features comply with 10 CFR 20.1406 (see Section 12.3.6.1).

12.3.1.7.4 Piping

Nonradioactive piping is segregated from radioactive piping to reduce workers' exposure to radiation during maintenance. This design feature is illustrated by the arrangement of piping ducts at elevation -21 feet of the Nuclear Auxiliary Building (Figure 12.3-43). The nonradioactive piping ducts, located in green zones, are located on the north side of the building. The radioactive piping ducts run along the west wall, around the south area of the building that is north of the coolant storage tanks, then up the east wall, and finally around the valve rooms on the north side.

The U.S. EPR routes radioactive piping through designated pipe ducts to the compartments where the components that the piping serves are located. Pipe ducts that serve radiation source components are located on the opposite side of the room from the accessible areas. For example, as shown on Figure 12.3-43, the pipe duct along the west wall serves the rooms containing the gas drier, gas cooler, and two rooms containing coolant treatment circulating pumps. The access corridor is located on the east side of these rooms; thus, the pipe penetrations do not penetrate the access corridor.

The piping compartments have labyrinth shielding and, in some cases, shielded doors to eliminate the streaming that a high radiation source emits. Potentially radioactive piping is located in appropriately zoned and restricted areas, and process piping is monitored to control access and limit exposure.

Piping configurations avoid stagnant legs by locating connections above piping centerlines, by using sloping rather than horizontal piping runs, and by providing drains at low points in the system. Thermal expansion loops are raised rather than dropped, where possible. In radioactive systems, nonremovable backing rings in the piping joints are prohibited, eliminating a potential crud trap for radioactive materials. Butt welds are used in lieu of socket welds for resin slurry and evaporator bottoms piping. Whenever possible, branch lines with little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

Pipes embedded in concrete structures are minimized to the extent practical. Concrete embedment is not relied upon as a shielding option because pipes embedded in concrete impede inspections, impede repairs, and increase dose and waste during decommissioning. Also, floor drain pipes at the lowest elevation that are embedded in concrete include a concentric guard pipe fitted with an alarm moisture detection monitor.

12.3.1.8 Access to Radiologically Restricted Areas

The U.S. EPR provides lockable doors to high-radiation areas, in compliance with 10 CFR 20.1601. Very high radiation areas are designed to be generally inaccessible, in compliance with 10 CFR 20.1602. Additional information on access to radiologically restricted areas is provided in the Radiation Protection Program (see Section 12.5).

12.3.1.8.1 Reactor Building

The very high radiation areas in the Reactor Building equipment compartment during normal and refueling operations include:

- The spreading area, which is designed to contain the molten mass from the reactor vessel in the event of a severe accident. This area is inaccessible.
- The reactor cavity, which is the location for the refueling pool during fuel handling and is a very high radiation area during refueling and during operation. An access room is provided in the design to enable workers to access the reactor vessel head. Double doors to the reactor cavity prevent workers from entering the reactor cavity.
- The core internals storage area, which is a very high radiation area during that portion of the refueling evolution in which the internals are removed from the reactor vessel and stored. This area is flooded with refueling water during this period and is inaccessible.
- The instrument lance storage, which is a very high radiation area during that portion of the refueling evolution in which the instrument lances are removed from the reactor vessel and stored. This area is flooded with refueling water during this period and is inaccessible.
- The transfer pit, which is a very high radiation area during that portion of the refueling evolution in which spent fuel is being moved between the Reactor Building and the Fuel Building. This area is flooded with refueling water during this period and is inaccessible. When not refueling, access is provided through the access room.

The aeroball measurement room is inside the Reactor Building and controlled as a high-radiation area. The room has interlocks which prevent access while the system is in operation. The aeroball system normal operation does not require local operator action and is not considered a radiological vital area.

Radiation sources in the Reactor Building include the reactor vessel, RCS, CVCS, safety injection system, pressurizer relief tank, in-containment refueling water storage tank, refueling system, aeroball system, and the reactor drain system. Doors separate Reactor Building equipment and service compartments.

The following figures illustrate the Reactor Building and are based on the general arrangement drawings provided in Section 1.2:

- Figure 12.3-1—Spreading Area at the -20 Ft Elevation of the Reactor Building.
- Figure 12.3-2—Reactor Cavity at the +17 Ft Elevation of the Reactor Building.
- Figure 12.3-3—Core Internals Storage Area and Instrument Lance Storage Areas at the +17 Ft Elevation in the Reactor Building.
- Figure 12.3-4—Transfer Pit at the +17 Ft Elevation in the Reactor Building.
- Figure 12.3-7—Reactor Cavity Section.
- Figure 12.3-8—Containment Building Section Looking Plant-West at the Reactor Cavity, Core Internals Storage, Instrument Lance Storage, and Spreading Area.
- Figure 12.3-9—Containment Building Section Looking Plant-East at the Reactor Cavity, Core Internals Storage, Transfer Pit, and Spreading Area.

12.3.1.8.2 Fuel Building

The very high radiation areas in the Fuel Building during normal and refueling operations are:

- The transfer pit, which is a very high radiation area only during that portion of the refueling evolution in which fuel is being moved between the Reactor Building and the Fuel Building. This area is flooded with refueling water during this period and is inaccessible.
- The spent fuel pool, which is flooded with water and is inaccessible.
- The cask loading pit, which is flooded with water and is inaccessible.

The water in the spent fuel pool and shielding in the walls maintain occupational doses ALARA. Occupied areas adjacent to the fuel transfer tube are shielded so that dose rates are less than 100 rads per hour during fuel movement operations, in accordance with Section 12 of the NUREG-0800 SRP (Reference 1).

The following figures illustrate the Fuel Building and are based on the general arrangement drawings provided in Section 3.8.4:

- Figure 12.3-5—Transfer Pit at the +12 Ft Elevation in the Fuel Building.
- Figure 12.3-6—Loading Pit, Spent Fuel Pool, and Transfer Pit at the +24 Ft Elevation of the Fuel Building.
- Figure 12.3-10—Loading Pit Section Looking Plant-West in the Fuel Building.
- Figure 12.3-11—Transfer Pit Looking Plant-West in the Fuel Building.
- Figure 12.3-12—Spent Fuel Pool Section Looking Plant-North in the Fuel Building.

12.3.1.9 Equipment Design Features and Shielding for ALARA

12.3.1.9.1 Activated Corrosion Product Control

The selection of materials and chemistry control minimize production of activated corrosion products. The distribution of corrosion products is limited by cleanup systems, by providing laminar flow, by providing smooth surfaces inside piping and components, and by minimizing corrosion product traps in the RCS. These systems and controls are described further in Section 12.1.2.3.

12.3.1.9.2 Equipment Design Features

The equipment described below is common to many of the U.S. EPR plant systems and contains features that result in the reduction of personnel radiation exposure.

Liquid Filters

The filter handling process uses several approaches to minimize exposure to personnel and the possibility of inadvertent radioactive release to the environment, including:

- Compartmentalization. The filters are located within a shielded compartment equipped with ventilation supply and exhaust, as well as drainage.
- Remote handling system. The filter changeout process is generally automated, allowing personnel to monitor and control the operation from a shielded control panel. The filters and filter housings are standardized so that a single filter changeout machine can access and change out the filters.
- Layout design features. Space is provided for filter removal, filter placement into a shielded cask, and transportation of the cask to the Radioactive Waste Processing Building.

Demineralizers

Demineralizer units are located in shielded compartments equipped with removable top shield plugs. Spent demineralizer resins are remotely flushed and hydraulically transferred to spent resin tanks; this process eliminates the need to remove the top shield plug. The compartments are equipped with individual ventilation supply and exhaust, as well as drainage.

Adsorber Beds

Adsorber beds are used in the gaseous waste processing system to hold up or delay gaseous fission products to permit decay before the gases are released through the vent stack.

Particulate Filters

High efficiency particulate air (HEPA) filters are installed in the ventilation system trains that exhaust spaces potentially containing airborne contamination. The ventilation system is designed to minimize dose resulting from service, testing, inspection, decontamination, and replacement of components. The components have sufficient space around them to provide ready access and to expedite work on these units. This arrangement is shown in the HEPA filter room in Figure 12.3-49—Nuclear Auxiliary Building +50 Ft Elevation Radiation Zones.

Recombiners

A recombiner is provided in the gaseous waste processing system to convert the free hydrogen and free oxygen in the gas mixture to water through a catalytically enhanced chemical reaction. This process reduces the levels of hydrogen and oxygen in the downstream flow in order to prevent explosive mixtures. The recombiner is located at elevation -11 feet of the Nuclear Auxiliary Building (shown as the KPL filter room on Figure 12.3-44).

Tanks

Tanks containing radiological material, fabricated with stainless steel materials to minimize corrosion, are sloped towards the process outlet nozzle unless special features such as agitators, stirrers, or decontamination provisions are provided and are oriented in the vertical position, if deposits are possible. Tank sampling stations are designed to minimize leakage to the floor and include leakage collection capability in the event of a leakage. Tanks are instrumented with both local and remote level indications and alarms.

Tank vents are designed to transfer any overflow to a receiving tank. Designed liquid leak-offs that have potentially not been degassed are collected in tanks that are purged to the gaseous waste processing system. Designed liquid leak-offs that have been degassed are collected in tanks that are purged to the plant ventilation system.

Tanks that collect liquids from the vent and drain system are recessed, where possible, and well shielded. An example of a recessed sump drain tank is the drain tank in the room labeled “KTA tank room” in Safeguard Building Division 1, as shown on Figure 12.3-21—Safeguard Building 1 -31 Ft Elevation Radiation Zones.

Evaporators

Evaporators are provided with chemical addition connections to allow chemicals to be used for descaling operations. Sufficient space is provided in the area to allow the removal of heating tube bundles. Shield walls separate more radioactive components from less radioactive components. Instruments and controls are located on the accessible side of the shield wall.

Pumps

Wherever practical, pumps have mechanical seals to reduce seal servicing time. Pumps in the radioactive waste systems are provided with flanged connections for ease of removal. Piping or pump casing drain connections are provided for draining the pump for maintenance.

Steam Generators

The U.S. EPR SGs incorporate numerous features to improve reliability and minimize maintenance worker occupancy times when special maintenance is necessary. An example of one such improvement is the material selection of low cobalt content alloys, such as Alloy 690 for the SG tubing (see Section 12.1.2.3.1).

Valve Operating Systems

Valves for radioactive systems are located in separate, shielded subcompartments (or “galleries”) rather than in high radiation areas. For valves located in the radiation areas, the design allows drainage of the adjacent radioactive components when maintenance is required.

The valve galleries are divided into subcompartments that service only two or three components. The subcompartments are further subdivided by walls and have shielded entrances so that the personnel are only exposed to the valves and piping associated with one component at a given location. An example of this configuration is shown in Figure 12.3-42. The two coolant supply and storage system valve rooms are located in an adjacent radiation area rather than in the high radiation area compartment with the coolant storage tanks on the level above.

For infrequently operated equipment, manual valves associated with safe operation, shutdown, and draining of the equipment are provided with remote manual operators or reach rods where feasible. To the maximum extent practical, simple, straight-reach rods have been used to allow the operators to feel whether the valves are tightly closed or not. Valves with reach rods are installed horizontally or vertically. Reach rods that are installed horizontally are installed with the valve stem and rod located above the heads of personnel to allow ready access.

Full-ported valves are used in systems that are expected to contain radioactive solids. Valve designs with minimum internal crevices are used where crud trapping could become a problem, especially for piping carrying spent resin or evaporator bottoms.

Sample Stations

Sample stations for routine sampling of process fluids are located in accessible areas. Shielding is provided at the local sample stations as required to maintain radiation zoning in proximate areas and to minimize personnel exposure during sampling. The counting room and laboratory facilities are described in Section 12.3.1.4.

Instruments

The U.S. EPR compartmental design reduces the dose to workers who maintain instrumentation and control equipment. Instrumentation and control equipment in the Reactor Building is located in the service compartment, with instrument and sensing line connections located to avoid corrosion product and radioactive gas buildup. Similarly, the Safeguard Building houses the switchgear and instrumentation and control equipment on the upper levels, which are low-dose-rate zones. Backup instrumentation eliminates the need for immediate repair.

Radiation Monitoring Systems

The radiation monitoring system indicates when a component has failed. The system signals are fed into the process information and control system (refer to Section 7.1.1.3.2). This electronic system permits an operator to access radiation monitor readouts from the main control room (MCR), health physics office, or the access control point. The radiation monitoring instrumentation is further described in Section 12.3.4.

Floor Drains

Floor and equipment drains and sumps are provided throughout the facility to collect and route contaminated liquids to the Radioactive Waste Processing Building for processing. Sumps in the facility are constructed of nonporous material. The inner surfaces of sumps that are in contact with the radioactive fluid are lined with an impermeable coating to reduce corrosion. Sumps that are at the lowest building elevations are double lined and fitted with alarmed leakage detection instrumentation. Sumps are recessed into concrete floors for shielding. Neither local gas traps nor porous seals are used on radioactive waste floor drains. Gas traps are provided at the common sump or tank.

Curbs and drain catch trays provide drainage control. For example, a curb is provided at elevation +5 feet of the Reactor Building. This curb surrounds the floor openings to prevent leaks or spills from flowing down to elevation -8 feet. Drain trays are located under the coolant treatment pumps in the Nuclear Auxiliary Building at elevation -21 feet, shown on the upper left of Figure 12.3-43.

Reactor Coolant Pumps

RCPs are located in areas that experience high radiation fields and contain primary coolant. Thus, these components can become significantly contaminated. In the event that special maintenance is required, design features such as a removable shaft and permanently installed decontamination equipment reduce occupancy times.

Reactor Coolant System and Reactor Vessel Insulation

Reflective metal insulation is installed on the RCS and reactor vessel. This insulation is fabricated out of individual pieces, each individually identified, that connect together with quick-disconnect clasps for easy removal and installation.

12.3.2 Shielding

This section provides the design bases for the U.S. EPR shielding and an explanation of the radiation zones.

12.3.2.1 Design Objectives

The U.S. EPR design objectives for shielding are:

- To provide shielding between individual components to avoid exposure to workers from adjacent components in compliance with GDC 61. This arrangement also provides work space and as much distance between components as is reasonably possible.

- To provide labyrinths and doors where the potential exists for streaming or scattered radiation. These labyrinths and doors provide the same shielding value as the adjoining walls. An example of this configuration is in the Safeguard Building, as shown in Figure 12.3-21. The access, “JNG pump room,” and “KTA tank room” shown on this figure together have labyrinth shielding and double doors with sufficient space for equipment to be removed and replaced.

- To avoid radiation streaming through penetrations, or to shield pipe penetrations to reduce streaming.

- To design concrete radiation shields that conform to RG 1.69 and ANSI/ANS-6.4-1997 (Reference 2) in accordance with Section 12 of the NUREG-0800 SRP (Reference 1).

- To design shielding that allows access to vital areas following an accident in compliance with 10 CFR 34(f)(2)(vii) and the criteria in Item II.B.2 of NUREG-0737 (Reference 3) as described in Section 12.3.5.2.

- To provide external wall thickness to reduce exposure to members of the public (see Section 12.3.5.3).

- To provide appropriate removable shielding for components that may need to be replaced in high-radiation areas.

- To design shield walls surrounding the Reactor Building and the Safeguard Building to reduce the dose rates immediately adjacent to the buildings to below 1 mrem/hr.

12.3.2.2 Shielding Calculation Methods

The photon spectra of the various sources used as input to the computer program MicroShield® (Reference 4) are provided in Section 12.2. For each source modeled, the dose rate was calculated based on:

- A concrete density of 2.35 g/cc and a steel density of 7.85 g/cc.

- Water phase density of 1 g/cc, gaseous phase of 0.00122 g/cc.

- A cylindrical geometry, with the diameter fixed for the component and the height adjusted to accommodate the full volume of the component.

- Preservation of the shell thickness, with no credit taken for internal structures such as heat exchangers.

- Selection of buildup factors based on the last significant shield material the radiation passes through.

- No credit taken for any reinforcing bar contained in the concrete.

- Integration set to 100 in the radial, circumferential, and axial directions.

The MicroShield® models for most components are a cylinder with side shields and a cylinder with end shields. The model dimension inputs are the radius, height, and thickness of the component (see Table 12.3-1—MicroShield® Input Parameters). The source terms are included in Tables 12.2-5, 12.2-7, 12.2-8, and Tables 12.2-10 through 12.2-18. Shielding thicknesses are taken from the concrete wall thicknesses of the adjacent compartment walls, ceilings, and floors, as shown on the Radiation Zones in Figures 12.3-13 through 12.3-59.

The RANKERN computer code (Reference 5) was used to determine the dose rates within the lower elevations of Safeguard Building Divisions 1 and 2 for post-LOCA conditions. The RANKERN code uses a Monte Carlo treatment of the source and is in common usage in Europe.

The gamma and neutron dose rates in the Reactor Building were calculated using MCNP, Version 4c (Reference 10). Within the Reactor Building, all areas and components making up two primary loops (approximately half of the entire building) were modeled. The dose rate in the personnel access areas was calculated based on:

- Concrete density of 2.35 g/cc and a polyethylene density of 0.95 g/cc.
- Neutron sources streaming through the primary piping openings in the bulk shielding.
- Nitrogen-17 sources in the reactor coolant loop.

The radiation sources listed below identify the location of major equipment and how doses to personnel are minimized.

Reactor Vessel

The reactor vessel is located low in the center of the Reactor Building and is well shielded. The shielding arrangement is shown in Figures 12.3-7 and 12.3-8.

Reactor Coolant System

Each RCP and each SG are located in individual compartments in the equipment compartment of the Reactor Building, providing shielding from each other as well as the service compartment of the Reactor Building (see Section 12.3.1.1). The shielding that separates the equipment compartment from the service compartment provides sufficient shielding to enable personnel to enter the Reactor Building during power operation (see Figure 12.3-13).

Chemical and Volume Control System

The CVCS components and piping are located in the Reactor Building and the Fuel Building. On the letdown portion of the system, the regenerative heat exchanger (elevation +5 feet) and the high pressure (HP) coolers (elevation -8 feet) are each located in separate well-shielded compartments of the Reactor Building. The compartments for these components are designed with removable shield walls.

Shielded pipe ducts are provided for the letdown piping from the Reactor Building to the Fuel Building as well as within the Fuel Building. The volume control tank, which is located in a compartment by itself, spans elevations +0 feet and +12 feet of the Fuel Building. A shielded anteroom provides access to the tank room at elevation +12 feet while providing further shielding protection to minimize personnel exposure. Charging pumps are located in separate shielded compartments in the Fuel Building starting at elevation -11 feet (see Figures 12.3-32, 12.3-33, and 12.3-34).

Primary Coolant Degasification System

The major components of the coolant degasification system are located in the Nuclear Auxiliary Building. The degasification column is located in a separate shielded compartment at elevation +34 feet of the Nuclear Auxiliary Building. Shielded pipe ducts are used to route reactor coolant to and from the letdown stream in the Fuel Building (see Figure 12.3-48).

Fuel Pool Cooling and Purification System

The components of fuel pool cooling portion of the system are comprised of pumps, heat exchangers, valves and piping located in the Fuel Building, an ion exchanger located in the Nuclear Auxiliary Building, and interconnecting valves and piping located in shielded valve rooms and shielded pipe ducts in the Reactor, Fuel, and Nuclear Auxiliary Buildings. The pumps are located in separate shielded compartments at elevation -31 feet of the Fuel Building. The heat exchangers are located in separate shielded compartments at elevation -20 feet of the Fuel Building. The ion exchanger is located in a shielded compartment which spans elevations -11 feet to +12 feet of the Nuclear Auxiliary Building (see Figures 12.3-30, 12.3-31, 12.3-44 through 12.3-46).

Liquid Waste Management System

The components of the liquid waste management system are located in the Radioactive Waste Processing Building and consist of the components described in Section 11.2. The liquid waste storage tanks and concentrate tanks are located in shielded compartments in the Radioactive Waste Processing Building. The sludge tank is located in separate shielded compartment. The monitoring tanks are both located in a single shielded compartment. These tanks are expected to contain water that meets discharge requirements.

The recirculation, sludge, centrifuge feed pump, decanter feed pump, and concentrate pumps are located in separate shielded compartments at elevation -31 feet in the building. The recirculation and discharge pumps are located in a single shielded compartment at elevation -31 feet of the building. The evaporator feed pump and the forced recirculation pump are located in separate shielded compartment at elevation 0 feet of the building.

The pre-heater, evaporator column, evaporator, and vent gas cooler are all part of the evaporator system and are located in a shielded compartment that spans elevations +12 feet to +36 feet in the building. The electrical heater is located in separate shielded compartment on the +23 feet elevation of the building. The decanter and separator are located in a single compartment at elevation -21 feet of the building (refer to Figures 12.3-52 through 12.3-59).

Gaseous Waste Processing System

The primary components of the gaseous waste processing system are located in the Nuclear Auxiliary Building and consist of the components described in Section 11.3. The components are generally located in separate, shielded compartments.

The three delay beds are located in two shielded compartments (two in one compartment and one in an adjacent compartment) that span elevations +34 feet to +50 feet of the Nuclear Auxiliary Building. The gas filter and the reducing station are located in a shielded compartment at elevation +64 feet of the Nuclear Auxiliary Building. The gel drier is located in a separate shielded compartment at elevation +34 feet of the Nuclear Auxiliary Building. Gaseous waste processing system piping is routed through shielded pipe ducts (refer to Figures 12.3-48 through 12.3-50).

The connected components purged by the gaseous waste processing system are located in separate shielded compartments in the Reactor Building, Safeguard Building, and Fuel Building. The NI vent and drain system primary effluent drain tanks, which are swept by purge gas, are located in separate shielded compartments at elevation -31 feet of the Fuel Building and at elevation -31 feet of the Safeguard Building, and are located adjacent to the Reactor Building to minimize piping runs and afford the most shielding.

The pressurizer relief tank, which is swept by purge gas, is located in separate shielded compartments at elevation +5 feet of the Reactor Building.

The shielding design provides reasonable assurance that the service corridors remain nonradiation areas during operation, thus permitting operators and maintenance technician access while maintaining dose ALARA. Because of the reduced self-shielding of the radioactive waste gases as compared to liquid radioactive wastes and the energetic photon spectra of these gases, the design employs additional shielding for these components.

Solid Waste Management System

The components of the solid waste management system are located in the Radioactive Waste Processing Building. The drum drying units and handling equipment are located in separate shielded compartments at elevation -31 feet of the Radioactive Waste Processing Building. The drum storage area is located in two shielded compartments that span elevations -31 feet to -11 feet in the Radioactive Waste Processing Building. The shielding design provides reasonable assurance that the service corridors remain nonradiation areas during operation, thus permitting operators and maintenance technician access while maintaining dose ALARA.

ESF Filters Post-LOCA

ESF filters for the annulus ventilation and safeguard controlled area ventilation systems are located in the Fuel Building at elevations +24 feet and +36 feet, and the MCR ESF filters are located in the Safeguard Building Division 2 and 3 at elevation +69 feet. During a post-LOCA event, these filters become loaded with radioactive material, creating high and very high radiation zones in the immediate surrounding areas. These radiation zones (for the annulus and safeguard filters, a maximum of 28 rem/h at floors above and below the filters and 3 rem/h in adjacent rooms to the filters) are in areas that do not need to be immediately accessed following a LOCA event. The filter loading will decay prior to personnel entry to the area. Access to these areas is addressed as part of the Radiation Protection Program (see Section 12.5). See Table 12.3-12—U.S. EPR Estimated Accident Mission Dose personnel doses because of direct shine from the MCR filters.

12.3.2.3 Radiation Zoning

Radiation zones for each area are defined by the dose rate in the areas, taking into account sources within each area as well as contributing dose rate from sources in adjacent areas and intervening shielding. Radiation zone categories employed and their descriptions are provided in Table 12.3-2—U.S. EPR Radiation Zone Designation.

Frequently accessed areas, such as corridors, are shielded for Zone 3. Buildings that contain radioactive material are shielded so that the dose rate outside of the external walls of the building are below 1 mrem/hr. The radiation zone maps are included in Figure 12.3-13—Reactor Building Cross-Section Radiation Zones through Figure 12.3-59—Radioactive Waste Building +53 Ft Elevation Radiation Zones. Personnel access paths are indicated on the radiation zone maps.

Additional personnel access paths for upper levels of the Safeguard Building (electrical areas) are included in Figure 12.3-60—Safeguard Buildings 2 and 3 +15 Ft Elevation Access Paths though Figure 12.3-63—Safeguard Buildings 2 and 3 +53 Ft Elevation Access Paths. These figures show the additional routes to access the MCR. The MCR can be accessed from the Access Building at elevation 0 feet by going through Division 4 and into Division 3 of the Safeguard Building. A staircase or elevator leads to elevation +53 feet of Division 3 and into the MCR.

The postaccident radiation zone maps are included in Figure 12.3-64—Safeguard Building 1 -31 Ft Elevation Postaccident Radiation Zones through Figure 12.3-71—Reactor Building Cross-Section Postaccident Radiation Zones.

12.3.3 Ventilation

12.3.3.1 Design Objectives

The U.S. EPR heating, ventilation, air conditioning (HVAC) system design criteria include the following:

- Design features for controlling the intake of radioactive material and maintaining personnel exposures ALARA in accordance with 10 CFR 20.
- Features for maintaining airborne radioactivity concentrations in unrestricted areas in accordance with 10 CFR 20.
- Features to maintain the dose to MCR personnel below the limit specified in 10 CFR 50, Appendix A, GDC 19.

12.3.3.2 HVAC System Description

The HVAC system for each of the following buildings is described in detail in Section 9.4:

- Containment Building (refer to Section 9.4.7).
- Nuclear Auxiliary Building (refer to Section 9.4.3).
- Fuel Building (refer to Section 9.4.2).
- Radioactive Waste Processing Building (refer to Section 9.4.8).
- Access Building (refer to Section 9.4.14).
- Safeguard Building (refer to Section 9.4.5, 9.4.6).

Although the control room envelope is considered to be a nonradioactive area, radiation protection is provided to maintain radiological habitability during design basis accidents (refer to Sections 9.4.1 and 6.4).

12.3.3.3 Protective Design Features

The following protective design features are used to accomplish the HVAC design objectives.

- For radiological areas, airflow within the area is from areas of low potential radioactivity to those of higher potential radioactivity.
- HVAC systems serving potentially contaminated areas maintain the area under negative pressure with respect to adjacent cleaner areas. Infiltration and leakage into the area is considered when sizing the system.

- Positive pressure is maintained in the MCR to prevent uncontrolled in-leakage of airborne radioactivity.

- Ventilating air is recirculated in the clean (uncontaminated) areas only.

- Removal of airborne radioactive iodine and radioactive particulates from the air stream prior to release to the environment, or means are provided to isolate these areas upon indication of contamination to minimize the discharge of these types of contaminants to the environment.

- Suitable containment isolation valves are installed in accordance with 10 CFR 50, Appendix A, GDC 54 and 56, including valve controls, to make certain that the containment integrity is maintained (refer to the description in Section 6.2.4).

- The NI vent and drain systems are connected directly to the ventilation systems rather than being vented to containment spaces.

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

- Maintenance for carbon filters is performed by special machines that remove any charcoal dust during recharging of the filters.

- The air cleaning system design, maintenance, and testing criteria are designed in accordance with the regulatory criteria contained in RG 1.52 (postaccident engineered safety feature atmospheric cleanup system) and RG 1.140 (normal atmospheric cleanup systems).

12.3.3.4 Air Filtration System Design

The facility layout provides dedicated rooms for HVAC filter housings and provides sufficient space for conducting HVAC maintenance activities, such as filter change-outs and bagging of filters. Filter change-outs are conducted within these separate compartments, preventing the spread of contamination. These separate HVAC compartments provide shielding to adjacent areas and corridors, minimizing dose to workers from nearby components. These filter room design features, including a representative layout of the filter housings, are shown in Figure 12.3-35.

Provisions for testing, isolation, and decontamination are described in detail in Section 9.4. Filters are monitored for pressure drops so that filter elements can be replaced before radiation levels become an ALARA concern or personnel hazard. Filters with a radioactivity level (because of a postulated accident) so high that a change of filter elements constitutes a personnel hazard can be removed intact. Filters with a buildup of short-lived radioisotopes are allowed to decay prior to being changed.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation and airborne radioactivity monitoring instrumentation is designed to:

- Assess radiation and airborne radioactivity levels at various plant locations to assist in the detection of abnormal operational conditions.
- Assess the magnitude of radionuclide releases to the environment.
- Assess accessibility to radiological vital areas during accident conditions.
- Provide a local readout, an audible alarm, and visual alarms in each monitored area to alert operating personnel. Visual alarms are provided in high noise areas as well as outside of each monitored area so that they are visible to operating personnel prior to entry.
- Display data from these monitors in the MCR using the process information and control system.
- Provide indication and alarms in the MCR and health physics office.

The instrumentation complies with the requirements of 10 CFR 20.1501, 10 CFR 50.34, and GDC 63, and conforms to applicable portions of RG 8.8 and RG 8.25. Additional information on instrument calibration is provided as part of the Radiation Protection Program (see Section 12.5).

12.3.4.1 Area Radiation Monitoring Instrumentation

12.3.4.1.1 Normal Operations

The area radiation monitoring instrumentation for use during normal operation and AOOs is provided to:

- Measure the radiation levels in specific areas of the plant.
- Provide a continuous record of radiation levels at key locations throughout the plant.
- Annunciate and warn of possible equipment malfunctions and leaks in specific areas.
- Furnish information for radiation surveys.

The area radiation monitoring instruments provide onscale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located as well as the maximum dose rate for AOOs and accidents. This instrument category is designated only for routine monitoring and is powered by the non-1E power supply (refer to Section 8.3.1), which has no auxiliary power.

Instrumentation placement follows the criteria for selection and placement of the area radiation monitoring instrumentation of ANSI/ANS-HPSSC-6.8.1-1981 (Reference 6) and includes:

- Location in areas that are normally occupied with and without restricted access and that have a potential for radiation fields in excess of the radiation zone designations.
- Location to best measure the exposure rates within a specific area, while avoiding shielding of the detector by equipment or structural materials.
- Consideration of the environmental conditions under which the monitor operates.
- Provision of access for field alignment, calibration, and maintenance.

Table 12.3-3—Radiation Monitor Detector Parameters includes the normal operation area radiation monitoring instrumentation.

12.3.4.1.2 Accident Monitoring

Area radiation monitoring equipment used during postulated accidents is provided to:

- Provide long-term accident monitoring using both safety-related and non-safety-related monitors.
- Provide a local readout, an audible alarm, and visual alarms outside of the room in which the detector is located and are visible to operating personnel prior to entry.

The accident area radiation monitors have usable ranges that include the maximum calculated accident levels and are designed to operate effectively under the environmental conditions caused by an accident. These monitors follow the guidance of RG 1.97 (refer to Section 7.5). This instrumentation is powered by the Class 1E uninterruptible power supply (EUPS), described in Section 8.3.1, which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

Table 12.3-3 includes the accident area radiation monitoring instrumentation.

12.3.4.1.3 In-containment High-Range Monitoring

The in-containment monitoring instrumentation used during postulated accidents is provided to:

- Measure gamma radiation, primarily from airborne gaseous radioactivity, in order to detect a breach of fuel cladding and the primary coolant boundary.
- Deliver a signal to the MCR to alert operators when predetermined setpoints are reached.

- Record data from the monitors to maintain a record of the gamma radiation after an accident as a function of time so that the inventory of radioactive materials in the containment volume can be estimated.

- Initiate Reactor Building containment ventilation system isolation and exhaust filtering on high radiation inside the Reactor Building.

- Initiate Safeguard Building controlled-area ventilation system isolation and exhaust filtering on high radiation inside the Reactor Building.

- Initiate Fuel Building ventilation system exhaust filtering on high radiation in the Reactor Building, concurrent with Safeguard Building controlled area ventilation system alignment.

These safety-related instruments and the associated network are environmentally qualified (refer to Section 3.11) to survive an accident and perform their design functions. The instruments are designed to respond to gamma radiation over the energy range of at least 60 keV to 3 MeV, with a dose rate response accuracy within a factor of two over the entire range. These monitors conform to the criteria set forth in 10 CFR 50.34(f)(2)(xvii), NUREG-0737, II.F.1 (Reference 3), and RG 1.97 (refer to Section 7.5). These monitors also meet the requirements of IEEE Std 497-2002 for a Type C instrument and a Type E instrument (Reference 7). This instrumentation is powered by the EUPS (refer to Section 8.3.1), which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

The in-containment high-range monitoring instrumentation consists of four independent high-range monitors located in widely separated areas in the service compartment of the containment. The locations are chosen to allow the detectors to be exposed to a significant volume of the containment atmosphere without obstruction so that the readouts are representative of the containment atmosphere, yet permitting easy access for calibration and maintenance activities.

Table 12.3-3 includes the high-range monitoring instrumentation.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

12.3.4.2.1 Normal Operations

The airborne radioactivity monitoring instrumentation for use during normal operation and AOOs is provided to:

- Continuously monitor for the presence of airborne radioactivity at selected locations of the plant that are normally occupied and may contain airborne radioactivity.

- Detect derived air concentrations in air (DAC) of the most restrictive particulate and iodine radionuclides in any compartment or room served by lowest ventilation rate within 10 hours (i.e., 10 DAC-hours) in accordance with Section 12.3-12.4 in NUREG-0800 (Reference 1).

- Verify the integrity of systems that contain radioactive material.
- Warn of unexpected releases of airborne radioactive material.
- Initiate automatic air isolation of the fuel handling area when a high exhaust activity setpoint is reached or instrument failure is detected.
- Initiate automatic isolation of the MCR intakes when a high activity setpoint is reached or instrument failure is detected. This is an ESF system (refer to Section 7.3).

This instrument category, other than ESF system detectors, is designated only for routine monitoring and is powered by the non-1E power supply (refer to Section 8.3.1), which has no auxiliary power. The ESF system detectors are powered by the EUPS (refer to Section 8.3.1) which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication.

Representative air concentrations are measured at the detectors, which are located as close to the sampler intakes as possible and upstream of HEPA filters, in accordance with ANSI/HPS N13.1-1999 (Reference 8).

Airborne monitors are located as shown on the following figures:

- Figure 12.3-73—Reactor and Fuel Buildings Airborne Monitoring,
- Figure 12.3-74—Access, Nuclear Auxiliary, and Radioactive Waste Buildings Airborne Monitoring.

Table 12.3-2—U.S. EPR Radiation Zone Designation includes the normal operations airborne radioactivity monitoring instrumentation.

12.3.4.2.2 Accident Conditions

Airborne radioactivity monitoring instrumentation is used during postulated accidents to provide indication and alarm to the MCR operator to indicate a potential or actual breach of the fuel cladding, primary coolant boundary, or containment by detecting the release of fission products. These instruments provide information that permits the MCR operator to assess the magnitude of the release in the event of an accident and to assess the release while in progress.

Emergency power is supplied to installed accident monitoring systems via the 1E power supply (refer to Section 8.3.1), which has diesel generators as the auxiliary power to provide power in the event of loss of normal power.

Table 12.3-4 includes the accident airborne radioactivity monitoring instrumentation.

12.3.4.2.3 Control Room Airborne Radioactivity Monitoring System

The MCR envelope (MCR, technical support center, and MCR HVAC room) is normally supplied with fresh unfiltered air. Airborne radioactivity monitoring instrumentation is provided for the MCR to:

- Monitor for airborne radioactivity so that the control room envelope remains habitable following a radioactive release.
- Provide a signal to initiate the supplemental air filtration system, isolate the MCR complex air intake and exhaust ducts, and activate the emergency habitability system when predetermined setpoints are exceeded.

The control room airborne radioactivity monitoring system is an ESF system (refer to Section 7.3). This instrumentation is powered by the EUPS (refer to Section 8.3.1), which is served by a two-hour battery backup with diesel generators as the auxiliary power to provide continuous indication. The system is illustrated in Figure 12.3-72—Main Control Room Airborne Monitoring.

12.3.4.3 Portable Airborne Monitoring Instrumentation

The use and location of portable instruments, associated training and procedures, the methods to determine airborne concentration, and surveys and procedures for locating suspected high-activity areas are part of the Radiation Protection Program (see Section 12.5).

12.3.4.4 Criticality Accident Monitoring

In lieu of the installation of a criticality monitoring system, design and analysis requirements specified in 10 CFR 50.68(b) are followed to prevent criticality. Refer to Section 9.1.1 for a description of how the U.S. EPR complies with 10 CFR 50.68(b) in the fuel handling and storage areas.

12.3.4.5 Implementation of Regulatory Guidance

A COL applicant that references the U.S. EPR design certification will describe the use of portable instruments, and the associated training and procedures, to accurately determine the airborne iodine concentration within the facility where plant personnel may be present during an accident, in accordance with requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737 (Reference 3). The procedures for locating suspected high-activity areas will be described.

A COL applicant that references the U.S. EPR design certification will provide site-specific information on the extent to which the guidance provided by RG 1.21, 1.97, 8.2, 8.8, and ANSI/HPS-N13.1-1999 (Reference 8) is employed in sampling, recording, and reporting airborne releases of radioactivity.

12.3.5 Dose Assessment

This section provides a dose assessment that includes the projected collective radiation doses from normal operations, AOOs, expected maintenance, and inspections in the various areas of the U.S. EPR and to members of the public in accordance with Section 12 of the SRP in NUREG-0800 (Reference 1). In addition, an assessment is performed of the dose to personnel required to perform tasks in radiological vital areas postaccident. Dose assessment of postulated radiological release events and accidents are presented in Section 15.0.3.

Radiation exposures to personnel occur within the radiologically controlled areas (RCAs) of the plant and within the site boundary. Within the RCAs, radiation exposures primarily result from direct gamma radiation from components and equipment containing radioactive material. In a few RCAs, internal and external radiation exposures can occur because of airborne radionuclides. However, experience at operating light water reactors, as described in NUREG-0713, Volume 19 (Reference 9), demonstrates that the dose from airborne radioactivity is not a significant contribution to the total dose. The U.S. EPR shielding is designed so that the direct radiation exposure outside the containment and outside other buildings containing radioactive materials is less than the radiation received within the RCAs.

12.3.5.1 Overall Plant Doses

The estimated total occupational radiation exposure for the U.S. EPR is 50 person-rem per year based on the ALARA improvements that result in a lower estimated annual dose compared to previous plant designs.

A combined license (COL) applicant that references the U.S. EPR design certification will provide site-specific information on estimated annual doses to construction workers in a new unit construction area as a result of radiation from onsite radiation sources from the existing operating plant(s). This information will include bases, models, assumptions, and input parameters associated with these annual doses.

12.3.5.1.1 Dose Calculation Method

The occupational radiation dose for the U.S. EPR was determined in accordance with the methods described in RG 8.19. This estimate is based on data from U.S., French, and German operating plants, adjusted for differences because of power level. The French and German operating plant data is the most representative data available because of the similarity in design features to the U.S. EPR. The occupational dose rate for activities outside of the Reactor Building (mainly because of deposition of activated corrosion products) was conservatively increased because of the increase in power and cycle length between the U.S. EPR and available data.

The following groupings of activities are used to determine the estimated annual occupational radiation exposure at the U.S. EPR:

- Reactor operations and surveillance.
- Routine maintenance.
- Inservice inspection.
- Special maintenance.
- Waste processing.
- Refueling.

The estimated annual personnel doses are summarized in Table 12.3-5—Estimated Annual Personnel Doses. Additional information for each activity is presented below.

12.3.5.1.2 Reactor Operations and Surveillance

This category consists of recurring activities performed by operations, health physics, maintenance, instrument and controls, and chemistry personnel. For these personnel, occupational dose is primarily accumulated while performing activities within the Reactor Building and Nuclear Auxiliary Building. Examples of activities in this category include:

- Operator rounds (inspections).
- Valve repositioning.
- Logging of data obtained from instrumentation throughout the plant.
- Surveillance testing on equipment located throughout the plant in conformance with technical specifications or other regulatory requirements.
- Calibration of instrumentation found out of calibration during surveillance testing.
- Health physics surveys.
- Chemistry sampling and analysis of radioactive fluids.

The Reactor Building shielding design allows containment entries to the service compartment at any power level (see Section 12.3.1.1). Entry into containment during operation is normally scheduled during the last few days of power operation. This scheduled entry is for staging outage-related maintenance equipment to allow a more efficient outage period and reduce down time.

Table 12.3-6—Dose Estimate for Reactor Operations and Surveillance provides a breakdown of the individual and collective doses for reactor operations and surveillance.

12.3.5.1.3 Routine Maintenance

Routine maintenance consists of the following types of tasks:

- Decontamination of various portions of the radiologically controlled area.
- Valve maintenance, including repacking.
- Relamping.
- RCP oil changeout.
- Filter replacement, including high efficiency particulate air filter and charcoal adsorber replacement.
- Snubber inspection and repair.

The U.S. EPR incorporates design features that reduce occupational exposures during routine maintenance, such as compartmentalization of radiological components, use of installed platforms, and reduction in the number of components that require maintenance activities. See Section 12.1.2 for additional ALARA design considerations.

Table 12.3-7—Dose Estimate for Routine Inspection and Maintenance provides a breakdown of the individual and collective doses for routine maintenance.

12.3.5.1.4 Inservice Inspection

Inservice inspection is performed on various plant components in accordance with ASME Code Section XI. Typically, some inservice inspection activities occur with every refueling outage, but certain activities are performed with the plant at power. Because many of the inservice inspected components are associated with significant radiation dose rates, design features are incorporated to minimize the occupancy times and personnel requirements for this activity.

Examples of these design features include:

- A reduced number of welds on the RCS require fewer inservice inspections.
- Insulation over areas to be inservice inspected is easily removed and reinstalled.
- Permanent work platforms are used so that scaffolding does not need to be erected or taken down.

Table 12.3-8—Dose Estimate for Inservice Inspection provides a breakdown of the individual and collective doses for inservice inspection.

12.3.5.1.5 Special Maintenance

Special maintenance consists of maintenance activities that go beyond routine scheduled maintenance, modifications of equipment to upgrade the plant, and repairs to failed components. Because these activities are generally unexpected and unscheduled, special radiation protection design features to provide dose reduction may not be in place. For the U.S. EPR, occupational dose reduction efforts related to special maintenance during the design phase consist of the following:

- Use operating experience to determine special maintenance activities that have resulted in significant occupational dose.
- Design components for high reliability as well as ease of maintenance or replacement (when components have required special maintenance with significant occupational dose).
- Provide sufficient space and structural support for the use of temporary shielding.
- Provide, as appropriate, maintenance support equipment that reduces occupancy times in radiation fields, such as permanently installed rigging or structural rigging points, readily available plant services such as compressed air and electrical power, and permanently installed platforms.
- Provide for ease of decontamination.
- Use design features such as a removable shafts and permanently installed decontamination equipment to reduce occupancy times.

Estimated annual doses from special maintenance are presented in Table 12.3-9—Dose Estimate for Special Maintenance.

12.3.5.1.6 Waste Processing

Waste processing activities include the processing, storage, and handling of liquid, gaseous, and solid wastes that result from plant operation. For liquid wastes, dose is kept to a minimum by segregating liquid wastes by category and processing these wastes remotely. In addition, the evaporator and centrifuge in the liquid waste processing system can be operated remotely.

Dose is reduced in the gaseous waste processing system by placing the components into well-shielded separate cubicles and locating the controls for the system remotely from the gaseous sources. The delay beds are highly radioactive, thus the shielding is designed to maintain the dose ALARA in surrounding areas to reduce dose to the operators whether working on this system or servicing adjacent system components.

The solid radioactive waste storage area is designed with strategically placed concrete columns to prevent radiation streams emanating from the individual drums from becoming additive. Permanently installed equipment provides remote handling of radioactive material.

Occupational doses for radwaste processing were estimated using an average of values reported for U.S. PWRs from NUREG-0713, Volume 19 (Reference 9). The estimated annual doses from waste processing are presented in Table 12.3-10—Dose Estimate for Waste Processing.

12.3.5.1.7 Refueling

The fuel handling system is designed so that the handling of irradiated fuel or equipment takes place under water to provide shielding and to maintain dose to personnel ALARA. The refueling machine in the Reactor Building and the spent fuel machine in the Fuel Building are equipped with a dose rate measurement device to halt the lifting of fuel assemblies if the allowable dose rates are exceeded.

Estimated annual doses from fuel handling are presented in Table 12.3-11—Dose Estimate for Refueling.

12.3.5.2 Postaccident Access to Radiological Vital Areas

The design of the U.S. EPR allows access to radiological vital areas with each mission task resulting in less than 5 rem total effective dose equivalent (TEDE), in accordance with 10 CFR 50.34(f)(2)(vii) and GDC 19, and in accordance with NUREG-0737 II.B.2 (Reference 3).

The following assumptions were used in determining mission doses under post-LOCA conditions:

- Radiological vital areas requiring postaccident accessibility include:
  - MCR, technical support center, and adjoining rooms.
  - Safeguard Building containment heat removal system pump rooms.
  - Safeguard Building residual heat removal system pump rooms.
  - Post-LOCA sampling room in the Fuel Building.
  - Post-LOCA ventilation air sampling room in the Fuel Building.
  - Radiological analysis laboratory in the Nuclear Auxiliary Building.
  - Diesel fuel oil delivery area.
- Missions considered are those to be carried out in the first 30 days postaccident.

- Figures 12.3-64 through 12.3-71 contain postaccident radiation zone maps that encompass the identified radiological vital areas. The radiation zones for Division 4 of the Safeguard Building are the same as those for Division 1 (symmetrical layout). These zones were determined in conformance with the source term assumptions of RG 1.183.

- For higher dose rate areas, actions such as flushing of pumps and lines and installation of local temporary shielding are used to reduce dose rate in area to 100 mrem/hr. Thus, a higher dose rate is used during preparatory work, and a lower dose rate is used after shielding installation and flushing operations are complete.

- Occupancy values used for the MCR, technical support center, and adjoining rooms are in accordance with RG 1.183.

Additional mission specific assumptions are as follows:

- MCR, technical support center, and adjoining rooms. The two sources of radiation are airborne radioactivity (because of ESF leakage resulting in both an immersion and inhalation dose) and direct radiation from the intake filters and from the recirculation filters located in the floor above the MCR. External shine from the Reactor Building and annulus structure is not a significant contributor to the dose rate because of the presence of substantial concrete shielding.

- Containment heat removal system or residual heat removal systems to clear blockage, back flushing of sump screens. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. Access begins no sooner than 20 hours post-LOCA.

- Post-LOCA grab samples. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. The sample lines within the sample room are the only significant sources of exposure. The first samples are drawn no sooner than 13 hours post-LOCA and are then transported to the laboratory in a shielded container.

- Post-LOCA ventilation air samples. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. The samples are transported in a shielded container to the laboratory.

- Process samples in the laboratory. The operator wears full protective clothing and respiratory protection, thus only direct dose is considered. Temporary shielding will be used as necessary so that the sampling box is the only significant source of exposure. The degassing vessel is the primary source of exposure within the sampling box.

- Diesel fuel delivery. The operator wears respiratory protection during delivery, thus only direct dose is considered.

Access routes for each radiological vital area within buildings are shown in Figures 12.3-75 through 12.3-79. For the diesel fuel delivery, trucks enter via the security access facility and proceed to the fill valve located on the outside of the

Emergency Power Generating Buildings 1 and 4. Table 12.3-12—U.S. EPR Estimated Accident Mission Dose summarizes each radiological vital area mission, including the dose rate, mission time (time to access the area, perform the task, and exit the area), and total mission dose.

12.3.5.3 Dose to the Public from Direct Radiation Exposure at the Exclusion Area Boundary

The annual radiation dose at the exclusion area boundary of 0.5 mi because of direct radiation from the Containment Building, Fuel Building, and other contained radioactive sources within the U.S. EPR plant site is less than 1 mrem and meets the limits of 10 CFR 20.1301(e) and 40 CFR 190. The dose rate from direct radiation at the site boundary does not exceed 2 mrem in any one hour in accordance with 10 CFR 20.1301(a)(2). The U.S. EPR design also provides storage for refueling water (IRWST) inside the Containment Building, instead of in an outside storage tank, thereby eliminating the refueling water storage tank as an offsite radiation source.

12.3.6 Minimization of Contamination

The U.S. EPR design complies with the requirements of 10 CFR 20.1406 by applying a contaminant management philosophy to the design of structures, systems, and components (SSC), which have the potential to contain radioactive materials. The principles embodied in this philosophy are prevention of unintended release, and early detection, if there is unintended release of radioactive contamination. The application of the contaminant management philosophy leads to design features that maintain occupational doses ALARA, minimize contamination, and facilitate the eventual decommissioning of the facility.

The following descriptions of the application of the contaminant management philosophy design demonstrate compliance with 10 CFR 20.1406.

12.3.6.1 Contamination Control for the Facility

12.3.6.1.1 Compartmentalization

Systems that are potentially radioactive are segregated from nonradioactive systems to minimize the migration of radioactive material across systems. The potable and sanitary water systems are designed to be separate from all other plant chemical and radiological systems to prevent the system from potentially being contaminated with chemical or radioactive materials. Potentially radioactive systems that interface with nonradioactive systems are designed to have a minimum of two barriers. For example, the essential cooling water (ECW) system supplies the water to the ultimate heat sink (UHS) cooling tower. The component cooling water system (CCW) is between the ECW system and the residual heat removal (RHR) system. The design provides a second barrier and the ECW water does not directly interface with the RHR water.

Two heat-exchangers have to simultaneously fail to directly transfer contaminated water to the UHS cooling tower. It is unlikely that two monitored systems can simultaneously fail and remain undetected.

The plant layout is designed so that personnel do not need to enter contaminated areas in order to reach noncontaminated areas. Similarly, the layout is designed so that personnel do not enter highly contaminated areas to reach moderately contaminated areas to perform required tasks.

12.3.6.1.2 Airborne

Air flow patterns route air from clean areas to progressively more contaminated areas and finally to filtered exhaust systems to prevent the spread of loose surface contamination. Air flow patterns through a room are designed to move the air away from the room entrance, toward the source of contamination, and then to a room exhaust. Air flow intakes are kept away from potentially leaking components.

The Reactor Building consists of two compartments that separate the equipment areas from the remaining volume of the building. The two-compartment design minimizes contamination levels in the Reactor Building. The air in the equipment compartment is continuously filtered with HEPA and charcoal filters to remove particulates and halogens, respectively. The removal of contaminants from the air in this compartment reduces the airborne concentrations, further minimizing the spread of contamination.

The radiological controlled areas are equipped with HEPA and charcoal filters. Radionuclide concentrations in air-conditioning coil condensation are reduced because of the filtration, resulting in a reduction in liquid contamination.

12.3.6.1.3 Spill Prevention

The U.S. EPR design also includes spill prevention and control measures. The components that are subject to leakage are placed on the lowest levels of buildings to assist in confining the contamination to as small an area as possible. Easily decontaminated, nonporous coatings are provided on floors and walls, where appropriate, in rooms subject to leakage of radioactive fluids. Equipment is mounted on pedestals to prevent the spill or leakage of one component from contaminating other components in the same room. Berms are provided for rooms in which components that are subject to leakage reside. Collection or drip pans are used under equipment, such as pumps, to limit contamination to a small area.

Decontamination rooms are provided in the design to better enable controlled decontamination of equipment to be repaired. The design also provides controlled access points, personnel contamination monitoring equipment, and protective equipment storage for additional contamination control.

12.3.6.1.4 Leak Detection

The Reactor Building is designed with a continuous steel liner and is equipped with leakage detection instrumentation. The liner protects the concrete and the environment below it from contamination. This design feature reduces the volume of contaminated material as well as dose during decommissioning and protects the environment.

The spent fuel pool is above the lowest elevation in the Fuel Building and is equipped with leak collection and detection instrumentation. Any leakage from the spent fuel pool is automatically detected. The spent fuel pool design prevents a pool leak from migrating to other sections of the Fuel Building and to directly reach the environment. Spent fuel pool inspections and repairs are facilitated by the location of the spent fuel pool within the building. Refer to Section 9.1.2.2.2 for additional information on the design of the spent fuel pool and the leakage detection system.

The CCW and ECW water is monitored to detect heat exchanger leaks.

12.3.6.2 Contamination Control for the Environment

Tanks that contain potentially radioactive liquids are located inside the NI structures. These tanks are all above the floor level and can be inspected and repaired in the event of a leak. The liquid from potential tank leaks is contained by berms and collected by the plant drain system for processing in the liquid waste management system. The only tank-like structures that are below grade are the UHS cooling tower basins, which is not a radiological system.

NI floor drains, sumps and piping that transfer potentially radioactive liquids to the liquid waste management system are designed with barriers and leakage detection instrumentation. These barriers and detection instrumentation minimize the introduction of uncontrolled radioactive effluent into the environment.

The only pathway allowed for the discharge of a liquid effluent is after treatment from the liquid waste management systems. Liquid effluent activity and volumetric flow are recorded continuously in the Radioactive Waste Building during discharges to allow for immediate intervention in case any limit is exceeded. A vertical U-bend trap in the piping ahead of the Radioactive Waste Processing Building outlet serves to prevent any unintentional flow into the environment or backflow into the building. The piping outside the Radioactive Waste Processing Building is provided with a concentric guard pipe with the outer pipe fitted with an alarmed moisture detection monitor, which detects any leakages. The double pipe system extends to the discharge pipe outlet into the cooling water outfall. Samples can be taken from the outer pipe in case of any leakages.

The design of the Reactor Building prevents any leakage from the reactor pressure boundary from reaching the environment. As described in 12.3.6.1, the Reactor Building is designed with a continuous steel liner and the building is equipped with leakage detection instrumentation.

12.3.6.3 Decommissioning

12.3.6.3.1 Building Arrangement

The NI design includes a common basemat for the Reactor Building, the Safeguard Building, and the Fuel Building, and a separate basemat for the Nuclear Auxiliary Building. The Turbine Building is separated from these structures. This design facilitates the phased dismantling of the plant, since the demolition of the turbine island does not affect the stability of the Nuclear Buildings. The location of the gaseous waste evacuation stack, on the roof of the Fuel Building, allows the stack to be kept in service while the reactor is being dismantled.

12.3.6.3.2 Provisions for Removal of Components and Structures

The contaminant management philosophy is applied to design provisions to facilitate the removal of components and structures to reduce on-site work time and dose. The installation of some large components, particularly the SGs, RCPs, and the pressurizer, has been reviewed for future disassembly. Future disassembly includes reverse handling and transportation operations, to incorporate the removal of these components from the Reactor Building in one single piece, if appropriate. Feedback from the replacement of SGs in the U.S. PWR plants provides guidance and is taken into account in the design. For example, a protected area behind the equipment hatch is created in which an entire SG can be handled.

The measures adopted to enable maintenance during operation facilitate the removal of waste. These measures are associated with an approach to decommissioning which is based on starting from the access points. This approach provides the necessary areas for the deployment of machinery, the disassembly, the placement and processing (such as decontamination and cutting) of the components, and the implementation of waste measurement, packaging and characterization facilities.

12.3.6.3.3 Decommissioning

The U.S EPR occupational ALARA objective is in part accomplished by reducing the number of tasks that are in higher radiation areas and facilitating the evolution of all tasks in the radiological control areas. The same ALARA objective is applied during decommissioning by limiting the time near highly active components and increasing the speed with which these components are removed. The following summary provides an example of the main measures adopted in the U.S. EPR design:

- The design of many components (e.g., core instrumentation, SGs, RCPs, pressurizer, heat exchangers, evaporator-degasser) facilitates their decommissioning.
- The majority of the above components are located in inaccessible areas because of the level of radiation. The plant design allows for removal of the components in one piece. Also, the design implements handling capabilities, adequately-sized openings, and access which enable removal of components in a single piece and subsequent processing in a more suitable environment.
- The design of the reactor pit and the melt plug to the spreading compartment makes it possible to fill the reactor pit with water, thereby allowing the pressure vessel to be disassembled while submerged.
- The position of the in-containment water storage tank (IRWST) under the reactor vessel allows the collection of any water leaks during the dismantling of the reactor internal components.
- The thermal insulation on the main primary circuit is easy to remove from around the welds because of its modular assembly.

Several operations have been identified as significant aids to dismantling:

- Draining, filling, and filtering of the spent fuel pool.
- Draining and filling of the SGs.
- Transfers between the Reactor Building and the Fuel Building.
- Treatment of solid, liquid, and gaseous waste.
- Ventilation.
- Fire surveillance and protection.
- Radioactivity controls, and monitoring of the environment.
- Draining of cavities and floors.
- Power supply, compressed air, and raw water.

The measures implemented for the related circuits and systems mean that they can be kept in service and maintained after the permanent shutdown of the reactor. The design of the reactor in four separated trains allows phasing the dismantling works train by train, while keeping inservice the auxiliary systems housed in the Fuel Building and the Nuclear Auxiliary Building.

12.3.6.4 Minimization of Radioactive Wastes

Waste is minimized by reducing the source of waste through design features that limit contamination. This design philosophy minimizes waste activity and volume both during plant operation and ultimate decommissioning.

12.3.7 References

1. NUREG-0800, "U.S. NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007.

2. ANSI/ANS-6.4-1997, "Specification for Radiation Shielding Materials," American National Standards Institute, 1997.

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

4. "MicroShield® User's Manual," Version 7, Grove Software, Inc., Lynchburg, VA, October 2006.

5. "RANKERN Version 15a – A Point Kernel Integration Code for Complicated Geometry Problems," Serco Assurance, October 2005.

6. ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation. Monitoring Systems for Light Water Nuclear Reactors," American National Standards Institute/American Nuclear Society, May 1981.

7. IEEE Standard 497-2002, "Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc, 2002.

8. ANSI/HPS-N13.1-1999, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," American National Standards Institute, 1999.

9. NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," U.S. Nuclear Regulatory Commission, December 1997.

10. LA-13709-M, "MCNP–A General Monte Carlo N-Particle Transport Code," Version 4c, J.F. Briesmeister, Ed., Los Alamos National Laboratory, April 2000.

Table 12.3-1—MicroShield® Input Parameters

Component	Dimensions (cm)			Material	Self-Shielding
	Diameter	Height	Thickness		
Pressurizer	282	1074	14.7	stainless steel	water phase gaseous phase
Volume Control Tank	260	270	1.5	stainless steel	water phase gaseous phase
CPS Demineralizers	resin volume 21.2 ft^3	---	---	stainless steel	water phase
CPS Filters	filter volume 7 ft^3	---	---	stainless steel	water phase
Degasifier	180	294	2	stainless steel	water phase gaseous phase
FPCPS	123.2	590	2.8	stainless steel	water phase
Liquid Radioactive Waste Storage Tank	330	tank volume 70 m^3	1.3	stainless steel	water phase
Liquid Radioactive Waste Evaporator	175	500	0.8	stainless steel	water phase gaseous phase
Gaseous Radioactive Waste	120	volume 4.75 m^3	1.5	stainless steel	water phase
HEPA Filters					
HEPA Filters, rectangular configuration	**Length**	**Width**	**Height**	stainless steel	air
	122	122	10.2		

Table 12.3-2—U.S. EPR Radiation Zone Designation

Zone	Dose Rate Upper Limit	Color Designation
1	≤0.05 mrem/hr	Green Zone
2	≤0.25 mrem/hr	
3	≤2.5 mrem/hr	
3A	≤5 mrem/hr	Yellow Zone
4	≤25 mrem/hr	
5	≤100 mrem/hr	Magenta Zone
5A	≤200 mrem/hr	Red Zone
5B	≤1 rem/hr	
5C	≤3 rem/hr	
6	≤5 rem/hr	
6A	≤30 rem/hr	
6B	≤100 rad/hr	
7	≤500 rad/hr	
8	>500 rad/hr	

Table 12.3-3—Radiation Monitor Detector Parameters
Sheet 1 of 3

Monitor Location	Monitor Provisions		Range
	Continuous	ACF	
Reactor Building	1 monitor – in front of Personnel Air Lock (elevation +5’)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Transfer Tube Room (elevation +17’)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Instrumentation Measuring Table Room (elevation +45’)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Setdown Area (elevation +64’)	---	1E-4 – 1E+1 rem/hr
	1 monitor – in front of Equipment Hatch (+64’ elevation)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Refueling Bridge (fuel movement operations only elevation +64’)	---	1E-4 – 1E+4 rem/hr
Fuel Building	1 monitor – Fuel Pool Floor (elevation +64’)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Setdown Area (elevation + 64’)	---	1E-4 – 1E+1 rem/hr
	1 monitor – Loading Hall (elevation 0’)	---	1E-4 – 1E+1 rem/hr
	1 monitor – Access to Transfer Pit (elevation +12’)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Decontamination System Room for RCP (elevation +49’)	---	1E-4 – 1E+1 rem/hr
	1 monitor (spent fuel movement only) – spent fuel mast bridge (elevation +64’)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Setdown Area near Equipment hatch (elevation +64’)	---	1E-4 – 1E+1 rem/hr

Table 12.3-3—Radiation Monitor Detector Parameters
Sheet 2 of 3

Monitor Location	Monitor Provisions: Continuous	Monitor Provisions: ACF	Range
	2 monitors in air leaving containment – next to air duct (downstream KLA2 low flow purge exhaust)	---	1E-5 – 1E+0 rad/hr
	2 monitors in exhaust air from exhaust cell (downstream KLB accident exhaust filter)	---	1E-4 – 1E+4 rem/hr (Kr-85, Xe-133)
	2 monitors in exhaust air from exhaust cell (downstream KLC accident exhaust filter)	---	1E-4 – 1E+4 rem/hr (Kr-85, Xe-133)
Safeguard Building (Mechanical)	1 monitor – Service Corridor near Containment Heat Removal System (elevation -31’ Division 1)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Service Corridor near Safety Injection System (elevation -31’ Division 2)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Service Corridor near Safety Injection System (elevation -31’ Division 3)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Service Corridor near Containment Heat Removal System (elevation -31’ Division 4)	---	1E-4 – 1E+4 rem/hr
	1 monitor – Personnel Air Lock Area (elevation 0’ Division 2)	---	1E-4 – E+4 rem/hr
Safeguard Building (Electrical)	1 monitor – MCR (+53’ elevation Division 2/3)	---	1E-4 – 1E+4 rem/hr

Table 12.3-3—Radiation Monitor Detector Parameters
Sheet 3 of 3

Monitor Location	Monitor Provisions		Range
	Continuous	ACF	
Nuclear Auxiliary Building	1 monitor – Filter Changing Equipment Room (elevation 0')	---	1E-4 – 1E+1 rem/hr
	1 monitor – In the primary sampling room (elevation -31')	---	1E5 – 1E+0 rem/hr
	1 monitor – In the active laboratory (elevation -31')	---	1E5 – 1E+0 rem/hr
	1 monitor – In the hot workshop (elevation +64')	---	1E5 – 1E+0 rem/hr
Post Accident Monitoring			
Reactor Building	4 monitors inside containment – Service Compartment	Signals Reactor Building air filtration isolation and RHR valve closure	1E-1 – 1E+7 rad/hr
Radioactive Waste Processing Building	1 monitor – In the drumming room next to conveyor	---	1E-4 – 1E+4 rem/hr
	1 monitor – In the decontamination room	---	1E-4 – 1E+4 rem/hr

Table 12.3-4—Airborne Radioactivity Detector Parameters
Sheet 1 of 4

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Reactor Building	1 noble gas monitor at refueling machine (used during spent fuel movement only)	---	1E-6 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 noble gas monitor in exhaust containment ventilation (upstream KLA05 filters)	---	3E-7 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 aerosol monitor in exhaust from containment ventilation (upstream KLA05 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	1 gaseous iodine monitors in exhaust from containment ventilation (upstream KLA05 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 5E-8 μCi/cc (I-131) Must be capable of detecting 10 DAC-hours
Fuel Building (Figure 12.3-73)	1 noble gas monitor at spent fuel mast bridge (used during spent fuel movement only)	---	1E-6 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 noble gas monitor in exhaust air of containment ventilation (upstream KLA2 filters)	---	3E-7 – 1E-2 μCi/cc (Kr-85, Xe-133)
	1 aerosol monitor in exhaust air of containment ventilation (upstream KLA2 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours

Table 12.3-4—Airborne Radioactivity Detector Parameters
Sheet 2 of 4

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
	1 gaseous iodine monitor in exhaust air of containment ventilation (upstream KLA2 filters)	---	5E-4 – 3E+0 μCi 3E-10 – 5E-8 μCi/cc (I-131) Must be capable of detecting 10 DAC-hours
	1 tritium monitor in exhaust air of containment ventilation (upstream KLA2 filters)	---	3E-9 – 3E-4 μCi/cc (H-3)
	2 monitors air leaving fuel handling area adjacent to monitored air duct (Fuel handling area)	Initiates automatic isolation of the Fuel Handling Area	1E-5 – 1E+0 rad/hr Must be capable of detecting 10 DAC-hours
Safeguard Building (Figure 12.3-72)	4 monitors intake air of the MCR	Initiates automatic isolation of the MCR ventilation intakes	1E-5 – 1E+1 rad/hr Must be capable of detecting 10 DAC-hours

Table 12.3-4—Airborne Radioactivity Detector Parameters
Sheet 3 of 4

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Nuclear Auxiliary Building (Figure 12.3-74)	6 aerosol monitors in exhaust air from exhaust cells of Safeguard Building and Nuclear Auxiliary Building (upstream KLE Filtration)	---	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours
	6 noble gas monitors in exhaust air from exhaust cells of Safeguard Building and Nuclear Auxiliary Building (upstream KLE Filtration)	---	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
	6 gaseous iodine monitors in exhaust air from exhaust cells of Safeguard Building and Nuclear Auxiliary Building (upstream KLE Filtration)	---	5E-4 – 3E+0 µCi 3E-10 – 5E-8 µCi/cc (I-131) Must be capable of detecting 10 DAC-hours
	1 aerosol monitor in laboratory exhaust air (KLE Laboratory Exhaust)	---	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours
	1 aerosol monitors in exhaust air of hot workshop (KLE Cell 3)	---	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours

Table 12.3-4—Airborne Radioactivity Detector Parameters
Sheet 4 of 4

Monitor Location	Monitor Provisions		Range[1]
	In-Process Continuous	ACF	
Radioactive Waste Processing Building (Figure 12.3-74)	1 aerosol monitor in exhaust air of decontamination room (KLF Rooms Cell 2)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	1 aerosol monitor in exhaust air of mechanical workshop (KLF System Rooms)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	2 aerosol monitors in exhaust air from exhaust cells (Upstream KLF Room Exhaust Filters)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc Must be capable of detecting 10 DAC-hours
	2 gaseous iodine monitors in exhaust air from exhaust cells (Upstream KLF Room Exhaust Filters)	---	5E-4 – 3E+0 μCi 3E-10 – 1E-6 μCi/cc (I-131) Must be capable of detecting 10 DAC-hours

Note:

1. Only particulate and iodine monitors are required to detect 10 DAC-hrs (see Section 12.3.4.2.1).

Table 12.3-5—Estimated Annual Personnel Doses

Category	Percent of Total	Estimated Annual Dose (person-rem)
Reactor operations and surveillance	13%	6.24
Routine maintenance	15%	7.50
Waste processing	10%	5.00
Refueling	16%	7.78
Inservice inspection	17%	8.52
Special maintenance	29%	14.42
Total:	100%	50

Table 12.3-6—Dose Estimate for Reactor Operations and Surveillance

Activity	Average Dose Rates (mrem/hr)	Worker Hours (person-hours)	Annual Dose (person-rem)
Chemical team	0.9	43	0.04
Cleaning and supplies management	0.9	4702	4.02
Health physics and safety team	0.9	1455	1.24
Operation team	0.9	954	0.82
Site management and coordination	0.9	140	0.12
Total:			6.24

Table 12.3-7—Dose Estimate for Routine Inspection and Maintenance

Activity	Average Dose Rates (mrem/hr)	Worker Hours (person-hours)	Annual Dose (person-rem)
All electrical work	0.6	715	0.44
All process control and I&C	0.6	1011	0.63
Driving and work on the polar crane and other locations	2.2	354	0.78
Filter replacement	8	48	0.38
Insulation and shielding	2.5	711	1.76
Miscellaneous work	0.5	693	0.33
Sludge lancing	2.9	497	1.45
Transfer pit maintenance	16	70	1.12
Work on the SG secondary side	2.9	195	0.57
Work on ventilation and filtration system	0.2	257	0.05
Total:			7.50

Table 12.3-8—Dose Estimate for Inservice Inspection

Activity	Average Dose Rates (mrem/hr)	Worker Hours (person-hours)	Annual Dose (person-rem)
Component/system inspection	16.2	451	7.31
Examinations for the sensitive welds	1.4	110	0.15
Nondestructive examinations	1.4	60	0.08
Pressure tests	0.9	200	0.18
Testing	1.4	591	0.80
Total:			8.52

Table 12.3-9—Dose Estimate for Special Maintenance

Activity	Average Dose Rates (mrem/ hr)	Worker Hours (person-hours)	Annual Dose (person-rem)
Maintenance on the Pressurizer	1.3	164	0.22
Opening and closure of the Pressurizer manway	1.3	69	0.09
Preparation work and opening of the SG primary-side manways	2.9	634	1.85
Primary-side repairs	2.9	219	0.64
Pump repairs	2.5	1999	4.95
Repairs to heat exchangers, vapor compressors, tanks, and separators	2.5	1167	2.89
Tube inspections, primary-side	2.9	493	1.44
Valve repairs	2.5	951	2.35
Total:			14.42

Table 12.3-10—Dose Estimate for Waste Processing

Activity	Average Dose Rates (mrem/hr)	Worker Hours (person-hours)	Annual Dose (person-rem)
Sampling and analysis of spent resins	69	19.5	1.3
Container inspection	100	26	2.6
Operation of waste processing and packing equipment	0.53	2000	1.1
Total:			5.0

Table 12.3-11—Dose Estimate for Refueling

Activity	Average Dose Rates (mrem/hr)	Worker Hours (person-hours)	Annual Dose (person-rem)
Core unloading and refueling	2.2	302	0.66
Inspections and work on threaded holes	2.2	148	0.33
Reactor pool decontamination	2.2	482	1.06
Reactor pressure vessel opening and closure	2.2	1473	3.24
Reactor pressure vessel work support between opening and closure	2.2	361	0.79
Work on in-core instrumentation	2.2	75	0.17
Work on the fuel	2.2	43	0.09
Work on the fuel manipulator crane	2.2	578	1.27
Work on the fuel transfer system	2.2	73	0.16
Total:			7.78

Table 12.3-12—U.S. EPR Estimated Accident Mission Dose

Mission	Dose per Person	Area Dose Rate	Occupancy Time
Staffing of MCR, TSC, nearby stations	4.0 rem (3.9 rem from immersion/inhalation 0.1 rem from direct shine from filters)	Varies	30 day
Access to SAHRS system	2.7 rem	200 mrem/h 700 mrem/h 100 mrem/h	0.5 h access/return 2 hr preparatory work 12 h repair
Access to RHR system	2.3 rem	300 mrem/h 400 mrem/h 100 mrem/h	1.0 h access/return 2 hr preparatory work 12 h repair
Postaccident sampling (RCS Liquid, Containment Atmosphere)	2.3 rem (1.05 rem extremity)	8.84 rem/h (63 rem/h extremity) 100 mrem/h	0.25 h in area, including 1 minute obtain sample 0.5 h transport route
Postaccident sampling (ventilation air samples)	383 mrem (obtain samples) < 1 mrem (extremity) < 1 mrem (transport)	2.3 rem/h (4.7 mrem/h extremity) 2.5 mrem/h	10 minutes in area to obtain sample 0.22 h transport route (access/return)
Sample counting lab (operator moves to lower dose rate area during processing)	1.0 rem	9.2 rem/h (adjacent to sampling box) 100 mrem/hr (low dose-rate area)	10 minutes in area (about 1/3rd of time in low dose-rate area)
Diesel Fuel Oil Delivery (per delivery)	0.5 rem	500 mrem/h	1 hour

Figure 12.3-1—Spreading Area at the -20 Ft Elevation of the Reactor Building

Figure 12.3-2—Reactor Cavity at the +17 Ft Elevation of the Reactor Building

Figure 12.3-3—Core Internals Storage Area and Instrument Lance Storage Areas at the +17 Ft Elevation in the Reactor Building

Figure 12.3-4—Transfer Pit at the +17 Ft Elevation in the Reactor Building

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Figure 12.3-5—Transfer Pit at the +12 Ft Elevation in the Fuel Building

Figure 12.3-6—Loading Pit, Spent Fuel Pool, and Transfer Pit at the +24 Ft Elevation of the Fuel Building

Figure 12.3-7—Reactor Cavity Section

Figure 12.3-8—Containment Building Section Looking Plant-West at the Reactor Cavity, Core Internals Storage, Instrument Lance Storage, and Spreading Area

Figure 12.3-9—Containment Building Section Looking Plant-East at the Reactor Cavity, Core Internals Storage, Transfer Pit, and Spreading Area

Figure 12.3-10—Loading Pit Section Looking Plant-West in the Fuel Building

Figure 12.3-11—Transfer Pit Looking Plant-West in the Fuel Building

Figure 12.3-12—Spent Fuel Pool Section Looking Plant-North in the Fuel Building

Next File