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).

Radiation protection doors inside the Reactor Building are designed based on the size, location and intensity of sources, and the type of radiation emitted, as well as the associated energy of the ionizing radiation. Two types of radiation protection door designs are used based on whether the door is protecting against a gamma source or a mixed neutron/gamma source. The material chosen for the gamma shield is stainless steel. High density polyethylene (PE) is applied for neutron shielding. The radiation protection doors used for a mixed neutron/gamma source (this includes most radiation protection doors inside the Reactor Building) are typically designed with one layer of steel, one layer of PE, and one layer of steel. The last layer of steel is necessary because the neutrons penetrating the PE create an additional non-negligible gamma source that must be attenuated by a thickness of steel. Gamma and neutron scattering and streaming are considered in the design of the radiation protection doors. As a general rule, radiation protection doors are embedded or overlap the concrete opening. If the optimized and preferred configuration is not achievable due to space limitation or layout constraints, an alternative solution (e.g., such as a maze with many steps) is applied to the radiation protection door. The lower and upper parts of the doors are also typically embedded in the concrete opening to provide a good shielding efficiency.

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 access room (UJA15 024) is not a radiological vital area and does not need to be accessed during normal fuel moves or after a fuel handling accident (FHA). The room outside the access room (UJA15 018) is a relatively large area containing the spray lines. To maintain this as an accessible area during refueling operations, there are two 16 cm thick steel radiation protection doors, one between room UJA15 024 and the reactor cavity and one between rooms UJA15 024 and UJA15 018, in addition to the 5 cm thick steel cavity sealing door between UJA15 024 and the reactor cavity (See Figure 12.3-2). These shielding doors are designed to reduce radiation dose rates to 1.7 rem/hr in room UJA 15024 and 32 mrem/hr in UJA15 018 at one meter from the wall in the event of a dropped fuel assembly.

- 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 access area leading to the transfer pit compartment, 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.

- The annulus area, 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.

To control access during fuel transfers, access doors or gates are used to limit entry to the above areas. The design features include double locks, local and remote alarms, and video surveillance in compliance with 10 CFR 20.1602. Design features include provision for emergency egress of personnel in the affected areas.

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.

Radiation protection doors that separate the Reactor Building equipment and service compartments consist of two types, as noted in Section 3.8.3.1.13. These two types are as follows:

- Radiation protection doors, with a pressure relief function.
- Radiation protection doors, without a pressure relief function.

The design features of these radiation protection doors which meet 10 CFR 20.1601 and 10 CFR 20.1602 are described in Section 3.8.3.1.13, including the provision for emergency egress from all accessible areas that fall within the requirements of 10 CFR 20.1601 and 10 CFR 20.1602.

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 (UFA16 023), 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 transfer pit access room (UFA15 096) is sealed from the transfer pit by a water tight access door in compliance with 10 CFR 20.1602.

- 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. Accessible 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).

To control access during fuel transfers, access doors or gates are used to limit entry to the above areas. The design features include double locks, local and remote alarms, and video surveillance in compliance with 10 CFR 20.1602.

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 (CSSS) 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 programs MicroShield® (References 4 and 14) and DIDOS-V (an internally-developed AREVA computer code) 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® and DIDOS-V 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 component shielding input parameters, Tables 12.3-15, 12.3-16, 12.3-17, and 12.3-18). The activity inventories and source spectra are included in Tables 12.2-7, 12.2-8, 12.2-10 through 12.2-18, and 12.2-20 through 12.2-70. 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.

DIDOS-V computer code is a point kernel shielding code used to determine dose rates from cylindrical sources.

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 for MCR 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.

Radiation Zones During Off-Normal Conditions

The radiation zone drawings reflect the maximum radiation zone designations in rooms containing radioactive sources. These radiation zone designations consider off-normal operations such as maintenance activities.

In addition, several areas have been identified with off-normal conditions that occur during specialized activities like spent fuel transfer. These activities may create short-term high radiation areas. Controlled access to these areas is provided in accordance with the ALARA program that is provided by the COL applicant as described in Section 12.1.3.

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.
- Drains from ESF filter system moisture separators are routed to the nuclear island drain and vent system (NIDVS) which handles potentially contaminated liquids.
- 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 in Section 7.1.1.5.5. Setpoint information is provided in Section 7.1.3.4.9.

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 area radiation monitoring instrumentation.

Refer to Section 7.5 for information regarding specific post-accident 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.
- 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 Stage 1 Containment Isolation on high range radiation monitor signal inside the Reactor Building.

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 applicable requirements of IEEE Std 497-2002 (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 NABVS and exhaust the fuel handling area by SBVS when a high exhaust activity setpoint is reached or instrument failure is detected (refer to Table 12.3-4 and Table 11.5-1, Monitor R-19).
- Initiates automatic closure of the CREF bypass inlet isolation dampers, opens CREF inlet isolation dampers, and initiates CREF fans when a high activity setpoint (refer to Table 11.5-1, Monitors R-29 and R-30 for automatic control functions) 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 12.3-4—Airborne Radioactivity Detector Parameters 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 airborne radioactivity monitoring instrumentation.

Refer to Section 7.5 for information regarding post-accident 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 9.4.1-1—Control Room Air Intake and CREF (Iodine Filtration) Train Subsystem, Figure 9.4.1-2—Control Room Air Conditioning and Recirculation Air Handling Subsystem, and Figure 9.4.1-3—Control Room Envelope Air Supply and Recirculation Subsystem.

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 access area outside the containment heat removal system pump rooms.
  - Safeguard Building access area outside the 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.

- The penetrations between the post-LOCA mechanical rooms and the access corridors are pressure tight and shielded.

- 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 containment and 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 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.

Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access presents the highest airborne concentrations in each of the radiological vital areas requiring post-accident access. These concentrations are based on post-accident Containment Building leakage and are consistent with entry into each of the radiological vital areas at the earliest expected time or later. The concentrations are time-dependent, and the maximum values presented correspond to the point in time when the inhalation dose is the greatest if no credit is taken for respiratory protection. Therefore, these concentrations constitute the maximum challenge to the respiratory protection credit.

The maximum external exposure doses for immersion for collection of post-accident ventilation air samples and sample analysis (included in the doses presented in Table 12.3-12) correspond to different airborne activity concentrations from those presented in Table 12.3-13. This is because the greatest potential for inhalation dose (averted by the use of respiratory protection) for those missions does not occur at the same time as the maximum external exposure dose. Other areas of the Fuel Building and Safeguard Buildings (e.g., Shield Building penetration areas) may exhibit higher activity concentrations, but those areas do not require post-accident access.

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.

To reduce the potential to contaminate the facility, the CSSS is designed so that leakage is detected and quantified, and the location of the leak is identified by the leakage detection system. As described in Section 5.1.1, the reactor coolant system pressure boundary provides the first barrier against reactor coolant contaminating the facility, and provides the second barrier against the release of radioactivity from the fission products of the fuel into the facility. The CSSS is designed to provide a secure envelope for the retention of reactor coolant and associated gases. This system uses vessels and welded piping (including the local sampling system) to provide a barrier against leakage, and is equipped with manual valves to provide system isolation from non-contaminated support systems such as the demineralized water distribution system, As described in Section 6.1.1.1, the piping and equipment exposed to reactor coolant are austenitic steel to avoid corrosion issues.

12.3.6.1.5 Cross-Contamination Prevention

To prevent cross-contamination of interfacing support systems, level measurements in vessels of the CSSS will prevent high liquid levels by automatically isolating inlet supplies to the vessels. In the event of a leak, these level measurements identify the vessel that is leaking by a low-level measurement. The compartment where the coolant storage tanks are installed is designed with a leak retention capability equivalent to the complete drainage of one tank. A steam generator tube leak will be detected by continuous process radiation monitors or radiochemical grab sampling on the secondary side of the steam generator as described in Section 5.4.2.5.2.5.

12.3.6.1.6 Maintenance Activities - Reactor Coolant Pumps

To reduce facility contamination due to maintenance activities involving the reactor coolant pumps, design features such as a removable shaft and permanently installed decontamination equipment are provided (see Sections 12.3.1.3 and 12.3.1.9.2). In addition, a dedicated room is provided for reactor coolant pump maintenance. For portions of the piping system potentially requiring inspections, maintenance, or repairs, bolted flanges are provided. For the tanks in the CSSS, inspection and maintenance can be conducted during plant operations as any one tank may be isolated, drained, purged, and opened independently of other tanks while maintaining the normal functions of the system. Prior to performing maintenance activities, the coolant storage tanks can be decontaminated. The monitoring instruments of the CSSS are located in accessible rooms for ease of inspection and maintenance.

12.3.6.1.7 Access Building - Eyewash and Shower Waste-Water

Eyewash stations and shower wastewater in the Access Building are routed to a tank in the nuclear island drain and vent system. Liquid effluent from the decontamination facilities (e.g., showers, floor washing) will be collected and stored in the storage tanks of the nuclear island drain and vent system.

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.

To prevent contaminated liquid release to the environment and to mitigate the airborne consequences of a leak to the environment, the nuclear island drain and vent system provides leak detection and isolation measures. Level instrumentation and other leak detection measures detect leaks that could result in internal flooding. These leak detection systems provide a signal to automatically isolate the affected system or to provide indication to the MCR to initiate operator action from within the MCR or locally (see Section 3.4.1).

To prevent potential contamination of the environment through the release of a normally non-contaminated liquid, the nuclear island drain and vent system is designed to prevent the inadvertent transfer of contaminated fluids to non-contaminated drainage systems (see Table 11.5-1). Portions of this system that are located in areas that may contain radioactive effluents are physically separated from the plant areas that do not contain radioactive effluents.

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 vent stack, on the roof of the Fuel Building, allows the vent 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.

12.3.6.5 Contamination Control for Systems

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 following principles are embodied in this philosophy:

- Prevention of an unintended release.
- Early detection if there is an unintended release.
- Prompt and aggressive cleanup, should there be an unintended release.

In accordance with these principles, tanks that potentially contain radioactive liquids are located inside the Nuclear Island structures, are located above floor level, and can be inspected and repaired in the event of a leak. Interfaces between radiological systems and non-radiological systems have been minimized. Where these interfaces do exist, at least two barriers are included in the design to minimize the potential for cross-contamination, and instrumentation is provided for prompt detection of potential cross-contamination.

The following design features have been incorporated into the U.S. EPR to minimize facility contamination:

- Extensive compartmentalization to minimize the spreading of contamination through the facility by containing potential contamination within a compartment and providing ventilation in these compartments. Ventilation to these compartments promotes air flow towards compartments with the greater potential for contamination minimizing the spread of airborne contamination in the facility.
- A smooth epoxy or, in high temperature exposure areas, an inorganic coating on surfaces that could potentially become contaminated to facilitate decontamination. A maintenance program maintains control and qualification of these applied coatings.
- Personnel decontamination areas located near the exit side of the primary access control facility of the Access Building and supplied with sinks and showers with drains that are routed to the liquid waste management system.
- A facility in the Access Building for decontamination and cleaning of personnel protective equipment, instrumentation, and small items. This facility provides a washdown area and sink that drains to the liquid waste management system. A positive air flow is maintained into the decontamination facility and exhausted into a monitored building ventilation system. Walls and floors in this area are coated to facilitate cleanup and decontamination.
- A core melt stabilization system to stabilize molten core debris resulting from the most severe category of reactor accidents by providing temporary retention and

conditioning of molten core debris, an area for corium to spread, features that limit potential energetic fuel-coolant interactions, and immobilizing and containing radionuclides.

The following design features have been incorporated into the U.S. EPR to minimize contamination of the environment:

- The nuclear island foundation basemat acting together with the reactor coolant boundary to maintain an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.
- Reinforcing steel bars in the concrete containment walls and dome for crack control and strength to accommodate seismic and other loads (e.g., thermal cycling) to minimize potential leak paths to the environment.
- Containment penetrations that are paths for potential bypass leakage terminate in areas of the surrounding buildings that are filtered during a postulated accident. The U.S. EPR design has no primary containment penetrations or seals that terminate outside the secondary containment to the general environment that create a bypass path. The three U.S. EPR penetration categories are described in Figure 6.2.3.2.3.
- Sufficient space in the facility layout for tools and equipment staging and to easily disassemble and reassemble components.

12.3.6.5.1 Fuel Storage and Handling System

The fuel handling and storage systems are designed to minimize contamination of the facility and the environment as described by the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2.

Design Provisions for Minimizing Contamination of the Facility

To minimize the potential to contaminate the facility, the spent fuel pool is designed so that no postulated event could cause excessive loss-of-pool water inventory, including designing in accordance with the criteria for Seismic Category I structures and locating piping connections near the top of the spent fuel pool. The spent fuel pool is a reinforced concrete structure with a stainless steel liner plate. Leak detection channels are provided behind seams in the liner plate for collection and monitoring of potential pool leaks. Any water collected is directed to the floor and equipment drain system and transferred to the liquid radwaste system for processing.

Based on the historical plant issues dealing with clogged pool liner channels and the eventual seepage of contaminated liquid releases into the surrounding concrete structures and groundwater, the U.S. EPR pool liner leakage monitoring system is an alternate design in accordance with NEI 07-07, “Industry Groundwater Protection Initiative – Final Guidance Document” (Reference 11). The pool liner collection channels (both vertical and horizontal) are accessible for monitoring, maintenance, testing, and inspections other than terminal end (discharge) at the pool liner leakage monitoring drain collection points, prior to disposal into the plant drain system.

The U.S. EPR spent fuel pool does not interface with the environment. Table 3.8-50—Fuel Building Plan Section A-A, Table 3.8-51—Fuel Building Plan Section B-B, and Table 3.8-52—Fuel Building Plan Section C-C, show that the U.S. EPR spent fuel pool is located above grade with several floors of accessible rooms located beneath it. Table 3.8-42—Fuel Building Plan Elevation +12 Feet, Table 3.8-43—Fuel Building Plan Elevation +24 Feet, Table 3.8-44—Fuel Building Plan Elevation +36 Feet, and Table 3.8-45—Fuel Building Plan Elevation +49 Feet show that the spent fuel pool is located in the interior of the Fuel Building (FB) and has no walls interfacing with external walls that interface with the environment. A leakage from the spent fuel pool will not directly reach the environment and is collected in a tank or sump located above the lowest building elevations. The FB sits on the common basemat providing additional protection for the environment.

Sumps for drain lines that may collect potentially contaminated liquids are lined with stainless steel over the potentially wetted surface. Concrete surfaces are protected by a smooth epoxy coated surface where there is potential for contamination.

Instrumentation is provided to detect and to alarm in the main control room (MCR) when low water level occurs in the spent fuel pool. Area radiation monitors are located in the fuel storage area for personnel and facility contamination protection. These area monitors alarm locally and in the MCR.

The concrete structure for the fuel transfer canal (and the spent fuel pool) is designed in accordance with the criteria for Seismic Category I structures. As such, it is designed to maintain leak-tight integrity to prevent the loss of cooling water from the pool. The fuel transfer canal is lined with stainless steel liner plates. Expansion joints are provided for the fuel transfer tube on the Reactor Building (RB) and FB side to accommodate the differential movement and provide leak-tight sealing. These expansion joints are equipped with a sensor for detecting leaks and providing an alarm in the MCR. In addition, to minimize potential facility contamination due to an event in the Containment Building, the fuel transfer tube between the fuel transfer canal and the spent fuel pool is equipped with a blind flange that provides containment isolation during power operations and a manual gate valve that allows isolation when the blind flange is removed during refueling operations. This fuel transfer tube consists of a stainless steel pipe installed inside a large sleeve that is anchored to the concrete of the Containment Building and welded to the containment liner plate. Bellows and water-tight seals are provided around the fuel transfer tube where it passes through the RB internal structures refueling canal concrete and the Reactor Shield Building and FB concrete. Where the potential exists for contamination, concrete surfaces are protected by a smooth epoxy.

To minimize facility contamination associated with the maintenance and replacement of contaminated fuel handling equipment, this equipment is designed for the life of the plant. In addition, the materials of construction, surface finish (for contamination prevention), and lubricant use are designed in accordance with the recommendations prescribed in ANS 57.1-1992 (Reference 12) for the fuel handling equipment.

Loading of spent fuel into casks is performed in the loading hall. The equipment and structures associated with the loading hall operation are defined as the Spent Fuel Cask Transfer Facility (SFCTF). The SFCTF has the following design features to preclude contamination of the facility. The overall design is consistent with U.S. EPR contaminant management philosophy in compliance with 10 CFR 20.1406.

- The fluid circuits have provisions to detect a leak via an abnormal pressure or level drop. The SFCTF is provided with an emergency stop pushbutton which can be actuated in case of leak detection.

- The penetration is equipped with double-seals for the upper cover of the penetration, for the double-walled bellow flange, and for the leak-tightness flange. The space between the two seals is monitored for leak tightness, as well as the space between the two walls of the bellows. The water leak sensor, connected to the plant main control room, monitors for potential leakage caused by failure of the seal at the upper cover of the penetration.

- The leak-tightness flange is connected at the upper end to the docking flange and the double walled bellows of the penetration. The lower end of the leak-tightness flange contacts the mating surface of the cask when the cask is docked to the penetration. When the transfer machine is not placed under the penetration, the leak-tightness flange is bolted with the lower cover of the penetration. The leak-tightness flange is equipped with two seals, each at the upper and lower end and an arrangement to monitor the space between the seals for leak-tightness. The leak-tightness flange is identified in Figure 9.1.4-7—Spent Fuel Cask Transfer Facility and Figure 9.1.4-8—Cask Loading Pit Penetration Assembly.

- An interlock permits opening of the upper cover of the penetration only after the correct docking has been confirmed, the anti-seismic locking of the SFCTM has been correctly engaged, and the correct water level has been attained. An accidental travel motion of the SFCTM when the penetration is docked is avoided using the interlock.

- Water level in the cask is checked before lowering the lid, before undocking the penetration, and before moving the SFCTM from the penetration.

- The geometry and surface finish of the immersed parts are selected to prevent the formation of radioactive-particle retention areas and to facilitate decontamination. Piping is designed to maintain minimum flow velocities and is installed with slopes to facilitate complete draining. The immersed parts, in particular the moving parts, are designed so that they can be easily and efficiently rinsed.

- The penetration is designed to remain leak-tight during and after a safe shutdown earthquake (SSE), except that a brief unseating of the normally leak-tight connection at the mating surface of the cask may occur resulting in some seepage around the seals, but does not result in any significant loss of water inventory from the cask loading pit or SFP.
- Effluents created by a postulated leakage of the operational fluid circuits on the SFCTM are collected by the trolley platform from where they can be drained to the NIDVS or loading hall sumps.
- The sumps in the loading hall are connected to the NIDVS, which prevents flooding of the loading hall.
- The SFCTM and the cask are checked for contamination before leaving the loading hall.

Design Provisions for Minimizing Contamination of the Environment

To minimize airborne contamination of the environment, the fuel building ventilation system provides appropriate ventilation and filtration to limit potential release of airborne radioactivity to the environment from the fuel storage facility under normal operation and in the event of a fuel handling accident in the spent fuel pool area. This ventilation system is continuously monitored by gaseous, particulate, and radio-iodine radiation monitors, which alarm locally and in the MCR. Isolation dampers isolate the ventilation system for specific rooms within the FB to mitigate the consequences of a fuel handling accident. Further information on the fuel building ventilation system is provided in Section 9.4.2.

The structural design and leak detection system features of the spent fuel pool and the fuel transfer tube also help protect the environment from contamination. During fuel handling operations, a controlled and monitored ventilation system removes gaseous radioactivity from the atmosphere above the spent fuel pool and processes it through high efficiency particulate air filters and charcoal adsorber units to the unit vent.

There are no portions of the spent fuel pool system handling potentially contaminated material that are buried or routed through exterior boundaries. Leak detection under the spent fuel pool provides full coverage in case of a leak, and leak detection equipment in channels aid in identifying the location of the leak. Sumps that collect potential spent fuel pool leakage are double lined with non-porous material. In addition, walls and curbs are used in areas with potential leaks of contaminated fluids to prevent the spread of these fluids.

12.3.6.5.2 Process Sampling System

The process sampling systems of the U.S. EPR are designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2. Nuclear sampling, sample activity monitoring, and radiation monitoring comprise the process sampling system.

The process sampling system is described in Section 9.3.2.

Design Provisions for Minimizing Contamination of the Facility

To minimize potential contamination of the facility, the process sampling systems are designed to:

- Monitor for potential higher than normal levels of radiation in the facility, and thereby, provide a means to mitigate it from spreading to other parts of the facility.
- Monitor variables and systems over their anticipated ranges to assure adequate safety, including those variables and systems that can affect the integrity of the reactor core and the reactor coolant pressure boundary.
- Provide a confinement boundary against any releases from the sampling system.
- Confirm that contaminated fluids are not transferred to non-contaminated fluids.

The process sampling systems monitor radioactivity levels in plant process streams and atmospheres, indicate and alarm excessive radioactivity levels, and in some cases automatically initiate protective isolation actions via radiation monitors (refer to Table 12.3-4 and Table 11.5-1) to minimize potential contamination of the facility. The systems consist of permanently installed, continuous monitoring devices together with a program of, and provisions for, specific sample collections and laboratory analyses. For example, area radiation monitors located in the Safeguard and Radioactive Waste Processing Buildings are provided to continually monitor radiation levels in the spaces which contain components for recirculation of loss of coolant accident (LOCA) fluids and components for processing radioactive wastes. In case of high levels of radiation, local alarms and signals to the MCR are provided as well as automatic control functions (refer to Table 12.3-4 and Table 11.5-1). Additional process monitoring functions are detailed in Section 11.5.4. The process sampling systems also provide information regarding the release of radioactive materials during normal operations, anticipated operational occurrences, and postulated accidents to provide an early indication of the need to initiate other protective actions to minimize potential facility contamination. For example, under accident conditions samples of the containment atmosphere can be taken via the sampling activity monitoring system to provide data on airborne radioactive concentrations within the containment.

The process sampling system obtains and analyzes key chemistry parameters such as chloride, hydrogen, and oxygen concentrations in the reactor coolant. The control of corrosive chemicals decreases the potential for facility contamination by decreasing the probability that the reactor coolant pressure boundary or fuel cladding are compromised due to degradation from corrosive chemical attack.

To minimize the potential for facility contamination due to a leak from the process sampling systems, sample lines penetrating the containment are capable of isolation upon receipt of a containment isolation signal from the reactor protection system. In addition, the portion of the process sampling system that includes the reactor coolant pressure boundary is designed, fabricated, erected, and tested to have a low probability of abnormal leakage, rapidly propagating failure and gross rupture. Motor-operated isolation valves in the three nuclear sampling lines connected to the reactor coolant system (RCS) maintain the reactor coolant pressure boundary integrity. Sample (glove) boxes are used to collect active liquid grab samples to confine any spills. Safety-related portions of the process sampling systems are designed to withstand the effects of natural phenomena. Non-safety-related portions of the process sampling systems are designed to have provisions for a leakage detection and control program to minimize the leakage from those portions of the process sampling systems outside of the containment that contain or may contain radioactive material following an accident.

The design of the process sampling system prevents the inadvertent transfer of contaminated fluids to non-contaminated drainage systems. This is accomplished by transferring contaminated fluids either back to the system being sampled or to an appropriate radwaste system.

The components are designed to permit periodic testing and in-service inspections during plant operation and are designed for the life of the plant. The piping connections and joints in these systems are welded except where flanged or screwed connections are required to facilitate equipment removal for inspection, maintenance, or pressure testing.The pipes inside containment are routed with a continuous slope without low points and each sample line is equipped with an inner and outer containment isolation valve. In addition, there is a sampling isolation valve in each line that belongs to the RCS. Sample lines are flushed for a sufficient period of time prior to sample extraction to remove sediment deposits and air and gas pockets. In addition, samples from tanks are taken from the bulk volume to avoid low points and sediment traps. Decontamination fluid can be injected via dedicated nozzles to vessels in the process sampling systems.

Design Provisions for Minimizing Contamination of the Environment

The process sampling systems minimize contamination of the environment by: (1) monitoring the atmosphere in various locations of the facility and taking actions to minimize potential releases from the facility; (2) monitoring the effluents discharged from the various ventilation systems addressed in Section 9.4 and taking actions to minimize potential releases from the facility; (3) controlling and potentially reducing the concentration and quality of fission products potentially released following postulated accidents; and (4) providing protection against leaks from sampling equipment.

The process sampling systems provide radiation and airborne monitors at various plant locations to assist in the detection of abnormal operational conditions. Upon detection of contamination, these monitors provide indication and alarms in the MCR and health physics office to initiate actions to minimize environmental contamination. For example, the containment atmosphere is monitored during normal and transient operations by containment gaseous radiation monitors and under accident conditions, can initiate RB containment isolation, and thereby, minimize any releases to the environment. In addition, sampling points are located on the process radiological monitoring and sampling systems to permit representative sampling for radiochemical analysis to indicate the existence of and, to the extent possible, the magnitude of reactor coolant and reactor auxiliary system leakage to the containment atmosphere, cooling water systems, and the secondary side of the steam generators. Process monitors also provide an alarm and gross indication of the extent of failed fuel. They also monitor radioactive waste systems and associated handling areas to detect and alarm under conditions that may result in a loss of residual heat removal capability and excessive radiation levels. In each of these cases, the process sampling system has monitors that provide indications and alarms to the MCR to initiate actions to minimize potential environmental contamination.

The process sampling systems also continuously monitor facility radioactivity levels in the effluent discharge paths during normal and accident conditions. The gaseous effluent monitoring and sampling system monitors the Reactor Containment Building, the FB, the Nuclear Auxiliary Building, the mechanical area of the Safeguard Buildings, the controlled area of the Access and Radioactive Waste Processing Buildings and the vent stack. Sampling points are located on effluent radiological monitoring and sampling systems to permit representative sampling for radiochemical analysis. The gaseous effluent radiological monitoring and sampling systems alarm but perform various automatic control functions (refer to Table 12.3-4 and Table 11.5-1), when radionuclide concentrations exceed the specified limits. As stated in Section 11.5.2, a COL applicant will describe in the Offsite Dose Calculation Manual (ODCM) how a gaseous radiological release will be controlled.

The liquid effluent radioactive waste monitoring and sampling system measures the concentration of radioactive materials in liquids to be discharged to the environment in batches from waste monitoring tanks. Prior to release of a liquid radioactive waste from a monitoring tank, the system obtains a representative sample which is radiochemically analyzed. If the sample is acceptable, flow from these tanks to the environment is permitted. The flow is monitored and if radionuclide concentrations exceed the specified limits, the discharge to the environment is automatically isolated upstream prior to any unacceptable release to the environment (refer to Table 11.5-1, Monitor R-32).

The non-safety-related portions of the process sampling systems are designed to control fission products, chloride, hydrogen, oxygen, and other substances that may be released into the reactor containment and also, reduce the concentration and quality of fission products released to the environment following postulated accidents. The control of corrosive chemicals decreases the potential for a release to the environment by decreasing the probability that the reactor coolant pressure boundary is compromised due to degradation from corrosive chemical attack. Non-safety-related portions of the process sampling systems include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

There are no buried pipes in the process sampling systems that handle potentially contaminated liquids, and hence, no means to contaminate the environment from a leaking pipe. There are also no by-pass lines around the radiation monitors for the liquid effluents released from the waste monitoring tanks. Gases that may potentially leak from these process sampling systems are collected by one of the HVAC systems described in Section 9.4 and subsequently filtered prior to a release from the vent stack.

12.3.6.5.3 Coolant Supply and Storage System

The coolant supply and storage system is designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Section 12.3.6.1 and 12.3.6.2.

Design Provisions for Minimizing Contamination of the Facility

To minimize the potential to contaminate the facility, the coolant supply and storage system is designed so that any leakage is detected, quantified, and the location of the leak determined by the leakage detection system. The RCS pressure boundary provides the first barrier against reactor coolant contaminating the facility and the second barrier against the release of radioactivity from the fission products of the fuel into the facility. The coolant supply and storage system is designed to provide a secure envelope for the retention of reactor coolant and associated gases. This system uses vessels and welded piping (including for the local sampling system) to provide a barrier against leakage and is equipped with manual valves to provide system isolation from non-contaminated support systems such as the demineralized water distribution system. The piping and equipment exposed to coolant are austenitic stainless steel to avoid corrosion issues. In addition, level measurements in the vessels of this system prevent high levels by automatically isolating inlet supplies to the vessels and preventing cross contamination of interfacing support systems. In case of a leak, these level measurements also identify the vessel that is leaking by a low level measurement. The compartment where the coolant storage tanks are installed is designed with a leak retention capability equivalent to the complete drainage of one tank. If the leak is due to a steam generator tube leak, it is detected by continuous process radiation monitors or radiochemical grab sampling.

The leakage detection systems, in combination with instrumentation from other interconnected systems, detect, quantify, and determine the location of leakage from the reactor coolant pressure boundary. These systems provide a method of collecting and quantifying reactor coolant pressure boundary leakage. These leakage detection systems include diverse measurement methods such as sump level and discharge flow monitors, containment atmosphere radiation monitors, containment air cooler condensate flow monitors, containment humidity monitors, temperature monitors of the reactor vessel closure joint, and reactor coolant inventory monitors at the pressurizer, volume control tank, and coolant drain collection tank. Provisions are also incorporated into the U.S. EPR to isolate, capture, and quantify leakage from known potential sources, such as flanges and relief valves, so that such leakage may be monitored separately from unidentified leakage. Each of these monitoring systems provide indications of leak rates and leak locations to the plant operators in the MCR.

Leakage of reactor coolant into the component cooling water system (CCWS) from a residual heat removal heat exchanger tube, reactor coolant pressure seal thermal barrier, or other source is identified by increased activity in the CCWS fluid as detected by a continuous monitor or routine sampling, and is also indicated by an unexpected increasing level in the surge tank. The dedicated CCWS surge tank is charged by nitrogen over-pressurization, resulting in potential component coolant leakage into, rather than out of, most interfaces with contaminated fluids (e.g., the severe accident heat removal system). For potentially contaminated systems operating at pressures greater than the CCWS, radiation and flow monitors in the CCWS detect and allow actions to be taken to limit the leakage into the system. For example, the chemical volume and control system (CVCS) high pressure coolers operate at pressures greater than the CCWS, and upon a high pressure cooler tube rupture results in a leak of reactor coolant into the CCWS. This leakage into the CCWS is detected by the CCWS flow meters (increased flow) or radiation monitors (increased radioactivity) and the high activity measurement generates a signal to automatically close the cooler isolation valves isolating the CVCS HP cooler from the CCWS to minimize the leakage into CCWS (refer to Table 11.5-1, Monitors R-51 through R-54).

In addition to maintaining the confinement barriers of the reactor coolant, facility contamination is minimized by the compartmentalization of buildings that contain portions of the reactor coolant pressure boundary. The Containment Building is divided into two compartments: an inner equipment compartment and an outer service compartment. The inner compartment contains the steam generators, reactor coolant pumps, and primary loop piping. The outer compartment contains support equipment. In the event of a reactor coolant leak, facility contamination is minimized by containing the majority of the contamination in the inner compartment. Similarly, the portion of the Safeguard Buildings that coolant passes through is in the radiological controlled areas of these buildings which are separated from the non-radiological areas (i.e., uncontrolled areas) that contain items such as instrumentation, control equipment, and switchgear. To minimize the spread of contamination, these two areas of the Safeguard Buildings are served by separate ventilation systems with the radiological controlled area ventilated by the safeguards building controlled-area ventilation system described in Section 9.4.5. The reactor coolant storage tanks that reside in the Nuclear Auxiliary Building are located in a similarly compartmentalized area within this building. This potentially contaminated area is ventilated by the nuclear auxiliary building ventilation system described in Section 9.4.3.

To minimize facility contamination caused by leaks of contaminated fluids, coatings such as sealers or special paint are used on walls, ceilings, and floors potentially exposed to these leaks to permit easy decontamination. The coating-sealed first level of the concrete compartments housing the coolant storage tanks contain elevated access doors, sealed floor penetrations, and floor drains that are normally closed so that each compartment has the capability of retaining the complete drainage of one coolant storage tank. This feature minimizes the spread of contamination in the unlikely event of such a leak.

To minimize facility contamination due to maintenance activities involving the reactor coolant pumps, design features such as a removable shaft and permanently installed decontamination equipment are provided. A dedicated room for maintenance of the reactor coolant pumps is provided in the U.S. EPR. In addition, bolted flanges are provided on the piping system only where removal is required for inspection, maintenance, or repair. For the tanks in the coolant supply and storage system, system inspection, and maintenance can be conducted during plant operations as any one tank may be isolated, drained, purged, and opened independently of other tanks while maintaining the normal functions of the system. Prior to performing maintenance activities, the coolant storage tanks can be decontaminated. The monitoring instruments of the coolant supply and storage system are located in accessible rooms for ease of inspection and maintenance.

Design Provisions for Minimizing Contamination of the Environment

The coolant supply and storage system is designed to minimize contamination of the environment by providing multiple barriers against radiological material reaching the environment. The coolant supply and storage system is designed to provide a secure envelope to retain reactor coolant and associated gases through the use of welded vessels and piping to provide a barrier against leakage of radiological material from this system. In case of a leak from this barrier, leakage detection and collection are provided to allow for identification of the location of the leak and to collect the leakage within the facility and prevent leakage from reaching the environment. Ventilation systems collect and filter releases from leaks to minimize the potential contamination released to the environment.

To minimize contamination of the environment from the reactor coolant, the Nuclear Island foundation basemat acts together with the reactor coolant pressure boundary to maintain an essentially leak-tight barrier. The concrete radiological shielding and the leak-tight steel liner plate within the containment limit the uncontrolled release of radioactivity to the environment. Additionally, the Reactor Shield Building completely encloses the Reactor Containment Building and provides a second containment barrier to the release of airborne radioactive material from containment. The space between these two buildings forms an annulus, which is maintained at a sub-atmospheric pressure and is filtered by the annulus ventilation system (refer to Section 6.2.3 for a description).

There are no portions of the coolant supply and storage system handling potentially contaminated material that are buried or routed through exterior boundaries. However, sections of the safety injection system (SIS) / residual heat removal system (RHRS) are outside containment; hence, these systems are designed to control and detect leakage outside containment following an accident. Upon detection of leakage, the section of these systems located outside confinement can be isolated.

12.3.6.5.4 Radioactive Waste Management Systems

The U.S. EPR radioactive waste management systems include the liquid waste, gaseous waste, and solid waste systems.

Design Provisions for Minimizing Contamination of the Facility

As described in Section 12.3.6.1, the design of the U.S. EPR minimizes facility contamination through the use of compartmentalization, heating, ventilation, air conditioning (HVAC) systems to control airborne dispersion, spill prevention features, and leak detection and mitigation features.

Design features are provided to control and collect radioactive material spills from liquid vessels and pipes. Tanks are designed with level measurements and overflows to prevent uncontrolled overflow paths to the environment; they are contained in rooms with drains to collect any spills, and to prevent uncontrolled releases to the environment. These rooms have no doors leading directly to the outside environment. The radioactive resins of the purification system are stored in two tanks located in separate, dedicated rooms on the bottom floor of the Radioactive Waste Processing Building. Each of these rooms is designed to contain leakage from these tanks using curbs located at the entrance doorway to each room. Other rooms drain to sumps equipped with leak detection systems that provide a signal to automatically isolate the affected system or provide an indication to the MCR to initiate operator action from within the MCR or locally.

The Radioactive Waste Processing Building is sized to provide space and support services for optional site-specific mobile or vendor-supplied processing equipment.

Flexible hose or pipe used with site-specific mobile or vendor-supplied solid waste processing systems is subject to the hydrostatic test requirements in accordance with requirements specified in Section 11.4.1.2.5. A mobile or vendor-supplied system is a site-specific design feature that is outside the scope of the design certification.

The U.S. EPR liquid waste management system receives degasified liquids in the storage tanks. These tanks are continuously vented to the radioactive waste processing building ventilation system so that any generation of gaseous activity is continually removed. The primary design functions of the gaseous waste processing system are to collect radioactive waste gases from the various systems in which they are released, to process these waste gases and provide sufficient holdup time for radioactive decay to reduce the activity present, and to control the subsequent release of processed waste gases to the atmosphere in compliance with regulatory limits. To continuously vent the tanks, the system maintains a negative pressure to prevent the escape of radioactive gases from components connected to the building air.

Releases from the gaseous waste processing system are continuously monitored by radiation sensors in the delay system discharge line. The system also provides grab sample collections for analysis from several different points on the process stream, and from each of the delay beds along the discharge line. Gaseous waste processing system releases are routed through the filtration system of the nuclear auxiliary building ventilation system. The gaseous waste processing system operates at a negative pressure relative to its surroundings, preventing radioactive gases from leaking and contaminating the facility, or flowing back towards the hydrogen, nitrogen, and oxygen gas supply systems. Multiple mechanical barriers are also used to prevent facility contamination through the gas supply systems.

In the drum drying station of the solid waste treatment unit, a vacuum seal is established on the drum and heaters are energized to evaporate the water from the drum. The vacuum in the drum allows the water to boil off at a lower required heating temperature. The water vapor is condensed, collected, and the volume measured before it is drained to the condensate collection tank. The air and radioactive non-condensable gases are routed to the radioactive waste processing building ventilation system for processing. Process monitors installed on the drum drying system detect in-process radiation levels to keep the operator informed.

The solid waste management system is equipped with a sorting box that is used to sort the various dry actives wastes produced in the controlled areas of the plant. The sorting box contains hand holes with rubber gloves for sorting the wastes. The sorting box is connected to the radioactive waste processing building ventilation system through a filling hood. Any airborne contaminants created during the sorting, shredding, or compaction processes are captured by the filling hood and subsequently treated in the radioactive waste building ventilation system.

The sampling box serves as the sampling point for the concentrate buffer tank. The box enclosure is equipped with gloves and a gate for inserting and removing the sample bottles. Inside the box are the sample valve and a demineralized water valve used to flush the inside of the box and the sample bottles. A ventilation connection is provided to maintain a negative pressure within the sampling box.

Area radiation monitors throughout the Radioactive Waste Processing Building detect excessive radiation levels and alert the operators to this condition. Any released gases that escape from the radioactive waste management systems are collected by the building ventilation HVAC systems, as described in Section 9.4. The piping and equipment for these components are constructed of stainless steel to avoid corrosion caused by wastewater, demineralized water, chemicals, and decontamination wastes. In addition, the liquid waste processing system is designed to allow the addition of a chelating agent to help remove any encrusted solids in the process (e.g., evaporator column) to prevent their buildup. Contamination of the facility potentially caused during filter change outs is minimized by the design of the U.S. EPR's filter changing equipment, which uses a filter changing machine to automatically and remotely perform filter change outs. The spent filters to be disposed from the filter changing unit are placed in a waste drum which is contained within a shielding cask to reduce exposure to personnel and mitigate potential contamination of the facility due to a spill during transport from the Nuclear Auxiliary Building to the Radioactive Waste Building. The filter changing machine is also equipped with seals to prevent leakage of contaminated gases into the room, and contain any leakage from the filters so that it drains from the bottom of the machine.

Similarly, contamination of the facility potentially caused during the removal of spent resins from the fuel pool purification mixed bed ion exchanger or the demineralizers is minimized by remotely removing these resins. These resins are remotely flushed and hydraulically transferred to the spent resin waste tanks and subsequently to the liquid waste treatment unit. Each of the radioactive waste management systems have been designed to allow maintenance during operations. For example, the filter changing unit is designed to confirm that the spent filter cartridges are always found in shielded equipment (i.e., the filter changing machine or one of the two shielding casks) and, in case the equipment gets contaminated, can be decontaminated by a mobile decontamination system. In addition, a dose rate monitor is also included in this area to provide maintenance workers notification of higher than normal exhaust rates.

The radioactive waste management systems have a design life of 60 years and, therefore, large component removal and its potential for facility contamination are minimized. The shielding casks of the filter changing unit are steel castings and the majority of the components of the filter changing equipment in this unit are stainless steel.

Design Provisions for Minimizing Contamination of the Environment

The radioactive waste management system is designed to minimize contamination of the environment by providing multiple barriers against radiological material reaching the environment and by meeting regulatory requirements for liquid effluent discharged to the environment. The U.S. EPR is designed to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, which includes anticipated operational occurrences. The radioactive waste management systems are designed to minimize inadvertent releases of radioactivity from the facility and to maintain permitted radioactive waste discharges below the regulatory limits of 10 CFR 50, Appendix I, during normal operation. Most of the operations for these units occur within the Radioactive Waste Building with the exception of portions of the liquid waste system (transferring releasable waste water to the environment) and portions of the solid waste system (transferring disposable wastes into disposal containers).

For the U.S. EPR, releases of radioactive effluent via the liquid pathway occur only via discharges from the monitoring tanks in the liquid waste storage system. In the monitoring tanks of the liquid waste system, the treated wastewater is chemically adjusted to an optimum pH and checked for activity prior to its discharge from the plant. The pH adjustment of wastewater in the liquid waste storage tanks and of the treated wastewater in the monitoring tanks also significantly reduces or eliminates the discharge of boric acid to the environment. The line releasing these effluents to the environment contains an administratively controlled, locked-closed upstream isolation valve. Personnel in the MCR maintain custody of the key to this valve and only issue the key upon receipt of a completed analysis demonstrating that the treated wastewater in a monitoring tank is within limits for release. When this valve is opened, the treated wastewater enters the activity measurement tank in the release line. Radiation sensors in this tank continuously measure and record the activity as the treated wastewater is released. Flow sensors downstream of the activity measurement tank continuously measure and record the volume and flow rate as the treated wastewater is released. If the total activity indicated by sensors exceeds predetermined limits, control signals are generated automatically to close two redundant downstream isolation valves, close the upstream isolation valve, and shut down the operating recirculation and discharge pump(s) (refer to Table 12.3-4 and Table 11.5-1, Monitor R-32). The content of the monitoring tanks is then sent back to the processing system for further treatment.

For gases, the U.S. EPR liquid waste management system receives degasified liquids in the storage tanks. These tanks are continuously vented to the radioactive waste processing building ventilation system (refer to Section 9.4.8) so that any generation of gaseous activity is continually removed. The gaseous waste processing system is primarily designed to collect radioactive waste gases from the various systems in which they are released, to process these waste gases and provide sufficient holdup time for radioactive decay to reduce the activity present, and to control the subsequent release of processed waste gases to the environment in compliance with regulatory limits. This system maintains a negative system pressure to prevent the escape of radioactive gases from the components connected to it.

Releases from the gaseous waste processing system are continuously monitored by radiation sensors in the delay system discharge line. The system also provides grab sample collections for analysis from several different points on the process stream, and from each of the delay beds along the discharge line. Gaseous waste processing system releases are routed through the filtration system of the nuclear auxiliary building ventilation system (refer to Section 9.4.3 for information on this HVAC system).

For spills from liquid tanks outside of containment, the U.S. EPR provides design features to control and collect radioactive material spills. The tanks for these systems are contained in rooms with drains to collect any spills and to prevent any uncontrolled release to the environment (refer to Section 9.3.3). If a leak escapes into the room containing a waste system vessel, then the room contains the leak or drains the leak to a nearby sump. The floor drain from a room can be opened to drain the leakage into a sump. From the sump, the liquid is pumped into a storage tank in the liquid waste storage system.

For the gaseous release associated with spills from these systems, the U.S. EPR provides the radioactive waste building ventilation system which is addressed in Section 9.4.8. Other portions of the solid waste treatment system contain sorting boxes used to sort the various dry active wastes produced in the controlled areas of the plant. These sorting boxes contain hand holes with rubber gloves for sorting the wastes and are connected to the radioactive waste processing building ventilation system through a filling hood. Any airborne contaminants created during the sorting, shredding, or compaction processes are captured by the filling hood and subsequently treated in the radioactive waste building ventilation system.

These radioactive waste treatment systems are contained in rooms that have no doors leading directly to the outside environment, which further prevents an environmental release. The piping and equipment for these systems are constructed of stainless steel to avoid corrosion caused by wastewater, demineralized water, chemicals, and decontamination wastes.

12.3.6.5.5 Equipment, Floor, Chemical, and Detergent Drain Systems

The equipment, floor, chemical, and detergent drain systems of the U.S. EPR are designed to minimize contamination of the facility and the environment as described by the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2.

Design Provisions for Minimizing Contamination of the Facility

To minimize the potential to contaminate the facility, the nuclear island drain and vent system (NIDVS) is designed to collect, temporarily store, and discharge in a controlled manner any leakage from equipment located on the Nuclear Island. In addition, this system is provided with leak detection equipment used to mitigate consequences associated with postulated leaks. Section 3.4 provides an assessment of the potential causes for internal flooding and how the NIDVS is designed to prevent such an event and how this system prevents backflow into areas of the plant that contain safety-related equipment through the use of check valves.

To minimize the spread of contamination of the facility created by a leak, the NIDVS is designed to include curbs and drain catch trays to provide drainage control. The NIDVS also includes leak detection and isolation measures to mitigate the consequences of leaks. Liquid leakages or discharges drain by gravity to the sumps as shown in Figure 9.3.3-1. Sump pumps automatically or manually transfer their contents to storage tanks. Mobile pumps are used only where drainage is impractical and are connected to the permanent piping using temporary flexible hoses. The water level instrumentation within sumps and storage tanks and other leak detection measures detect leaks. These leak detection systems provide a signal to automatically isolate the affected system or to provide indication to the MCR to initiate operator action from within the MCR or locally. For example, the sump pumps inside the Safeguard Buildings and FB are equipped with a double level measurement device for detection of leakage with an indication provided to the MCR.

The NIDVS is also designed with containment isolation valves to provide isolation of containment in case of a radiological release within containment, thereby, removing a potential leak path from containment to the rest of the facility. Additionally, the RB floor drains are designed to collect leakage from contaminated spaces in the RB and from process drains that cannot be recycled. The reactor coolant pressure boundary leakage drains to the floor drains system and ultimately to the sump where it is identified and quantified by the sump instrumentation. The NIDVS pumps, tanks, and sumps are sized to process the maximum expected rate of influx and total volume of expected leakage.

The NIDVS is designed and equipped with provisions to permit testing for operability and calibration. The storage tanks contained in this system are located in vessel pits which are equipped with alarming level detectors to detect their failure. These storage tanks can also be decontaminated via temporary connections. Components of the NIDVS are designed to operate for 60 years, thereby minimizing the potential generation of waste associated with operating and maintaining this system. The materials used in this system are compatible with the services required. Most components of this system are constructed of 304 stainless steel for corrosion resistance.

Eyewash stations and shower wastewater in the Access Building are routed to a tank in the NIDVS. Liquid effluent from the decontamination facilities (showers, floor washing) are also collected and stored in the storage tanks of the NIDVS.

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.

Design Provisions for Minimizing Contamination of the Environment

The NIDVS is designed to minimize contamination of the environment by providing multiple barriers to prevent radiological material from reaching the environment. The floor drain pipes at the lowest elevation that are embedded in concrete include a concentric guard pipe fitted with an alarm moisture detection monitor. 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.

To prevent a contaminated liquid release to the environment and to mitigate the airborne consequences of a leak to the environment, the NIDVS provides leak detection and isolation measures. Level instrumentation and other leak detection measures detect leaks that could result in internal flooding. These leak detection systems provide a signal to automatically isolate the affected system or to provide indication to the MCR to initiate operator action from within the MCR or locally.

To prevent the potential of contaminating the environment through the release of a normally non-contaminated liquid, the NIDVS is designed to prevent the inadvertent transfer of contaminated fluids to non-contaminated drainage systems. Portions of this system that are located in areas that may contain radioactive effluents are physically separated from the plant areas that do not contain radioactive effluents.

In addition to the NIDVS, the CVCS has provisions for a leakage detection and control program to minimize the leakage from those portions of the CVCS outside of the containment that contain or may contain radioactive material following an accident. The CVCS drains are piped to the NIDVS for storage and processing of the discharged liquids.

An oil collection system is also provided to collect and drain the motor lube oil (upper and lower bearing lube oil systems) in the event of leakage from the motor lubrication system to prevent any leakage to the environment.

No access openings or tunnels penetrate the exterior walls of the nuclear island below grade. In addition, pipes embedded in concrete structures are minimized to the extent practical as the concrete impedes inspections, impedes repairs, and increases dose and waste during decommissioning.

12.3.6.5.6 Building HVAC Systems

The building HVAC systems are designed to minimize contamination of the facility and the environment as described in the general protective design features and the air filtration system design listed in Sections 12.3.6.1 and 12.3.6.2.

Design Provisions for Minimizing Contamination of the Facility

The containment building ventilation system (CBVS), the fuel building ventilation system (FBVS), the safeguard building controlled-area ventilation system (SBVS), the nuclear auxiliary building ventilation system (NABVS), and the radioactive waste building ventilation system (RWBVS) are designed to minimize contamination of the facility. For each of these systems, this design objective is attained by maintaining a minimal air change rate and controlling the building pressurization. By maintaining a minimal air change rate, radiological material is less susceptible to becoming airborne and spreading to other portions of the building. Similarly, by maintaining pressures higher in areas of lower contamination relative to pressures in areas with the potential for higher contamination, the air flow travels from the lower to the higher areas of contamination, thereby minimizing the spread of contamination in the facility. For example, when the purge subsystem of CBVS is in operation, a differential pressure is maintained between the equipment compartment and the service compartment within the Containment Building, with the equipment compartment maintained at a slightly more negative pressure to allow any radiological activity to be contained in this compartment and prevent spreading contamination to the service compartment. For each of these systems, this design function is described in the following sections:

- For CBVS, Section 9.4.7.2.
- For FBVS, Section 9.4.2.2.
- For SBVS, Section 9.4.5.2.
- For NABVS, Section 9.4.3.2.
- For RWBVS, Section 9.4.8.2.

As described in Section 9.4.7.2.1, the CBVS also provides internal filtration to reduce radioactive contamination inside the equipment compartment of the Containment Building. This filtration system is designed to allow periodic inspection and maintenance.

To detect potential leaks in the facility, each of these ventilation systems, plus the annulus ventilation system (AVS), is equipped with airborne radioactivity monitors to allow monitoring of airborne radiation levels in the system. Area radiation monitors are also located throughout the facility to provide local readouts, audible alarms, and visual alarms to alert operating personnel. Section 12.3.4 provides additional details for these monitors and their actions. These monitors are located in accessible areas to allow maintenance, inspection, and replacement without significant personnel exposure occurring.

In addition to these radioactivity monitors, ventilation systems are designed to verify proper system behavior through the use of local instruments that measure properties such as differential pressures across filters, flows, temperatures, and pressures. Furthermore, indication of the operational status of the CBVS equipment, position of dampers, instrument indications, and alarms are provided in the MCR.

The design and installation of the components of these ventilation systems shall confirm that adequate clearance is provided for removal, maintenance, and inspections. However, to minimize the need to remove components from these ventilation systems, these components are designed to operate, with proper maintenance, for 60 years with the exception of some components such as motors and filters, which are monitored and replaced as necessary. Maintenance on the FBVS isolation dampers located in the fuel pool room can only be performed when no fuel handling activities are taking place in the FB. The containment purge subsystem of the CBVS is designed to clean up containment prior to an entry. The exhaust section of this portion of the CBVS is designed with redundant components allowing maintenance to be performed on a portion of this system during normal plant operation.

In general, these ventilation systems are not susceptible to streams that have the potential to encrust or crystallize in the ducting. However, as described in Section 9.4.2.1, the FBVS is equipped with electrical heaters in the boron rooms specifically designed to prevent crystallization in the borating system piping. Due to the continuous operation of these ventilation systems, blockages are not likely to occur in the ducting as airborne material collected by these systems are drawn towards the fans/exhausters and not allowed to settle and accumulate in the ducting. Airborne material may clog filters in these ventilation systems; therefore, these systems are designed with instruments to measure differential pressure across filters to avert clogging.

Facility contamination associated with potentially contaminated condensate from cooling coils in the CBVS, FBVS, NABVS, SBVS, and RWBVS is minimized by the inclusion of moisture separators and collection trays underneath the coils which collect and drain the condensate to the nuclear island drain and vent system.

The materials of the equipment used in these ventilation systems are compatible with the material in the process and facilitate decontamination. The exhaust and supply ductwork for these units are made of galvanized steel with the following exceptions:

- For CBVS, the exhaust ducts of the iodine filtration trains have airtight housings and all the ducts located outside the Containment Building are airtight and welded.
- For AVS, the accident train exhaust ducts are ferritic steel and the ducts of the normal ventilation train are concrete inside of the annulus and ferritic steel outside of the annulus.
- For FBVS, the main supply and exhaust duct chases are constructed of painted concrete and the ductwork for the fuel pool room is welded and constructed from stainless steel or from carbon steel with a coated surface suitable for decontamination.
- For SBVS, the main supply and exhaust air shafts are constructed of painted concrete and the surfaces of the metal and concrete exhaust ductwork which could be exposed to airborne contamination are painted with a special paint that allows easy decontamination.

The ductwork meets the design, testing, and construction requirements, per ASME AG-1 (Reference 13). Components for each of these ventilation systems are designed with consideration of minimizing deposits of material on component surfaces and ease of decontamination. For the CBVS and FBVS, all exhaust portions of ducts are capable of being decontaminated. Removal and transfer of contaminated filters are implemented under the Radiation Protection Program (see Section 12.5).

Design Provisions for Minimizing Contamination of the Environment

The filtered exhaust and the negative differential pressures with respect to the environment produced by the CBVS, AVS, FBVS, SBVS, NABVS, and RWBVS provide the primary protection against contamination of the environment. During normal operation, these ventilation systems produce a sub-atmospheric pressure in their ventilated zones and filter the air from these zones for removal of potential contaminants prior to release to the environment via the vent stack. The AVS provides isolation of the secondary containment and collects containment building leakage. Following a design basis accident, the AVS removes particulates from the contaminated air prior to release to the environment. The normal exhaust from the CBVS, FBVS, AVS, and SBVS are processed by the NABVS through a filtration train and the exhausted air is directed to the vent stack. The RWBVS draws air from locations in the Radioactive Waste Building where radioactivity is likely, and also collects air from activity-bearing systems, vented tanks, and work areas and machinery which may produce airborne releases.

Upon receipt of a containment isolation signal or a high radiation alarm in the mechanical areas of the Safeguard Buildings, the SBVS is isolated from its supply and the NABVS exhaust system, directing its exhaust air through the SBVS activated charcoal filtration beds located in the FB prior to release to the environment through the vent stack. Similarly, upon receipt of a containment isolation signal, the FBVS is isolated from the NABVS and the exhaust is processed through these same filtration trains of the SBVS. A containment isolation signal also isolates the normal operation trains from the NABVS and starts the AVS accident trains to draw a negative pressure in the annulus and filter the exhaust air through activated charcoal filtration beds located in the FB prior to release to the environment through the vent stack. The containment isolation signal also causes the CBVS to automatically isolate the containment atmosphere by quick closure of the system containment isolation valves to maintain the integrity of the containment boundary and to limit the potential release of radioactive material to the environment. The CBVS confines the containment volume and removes iodine released in the event of a fuel handling accident in the RB, The SBVS supply and exhaust duct network for the hot mechanical areas in the Safeguard Buildings are equipped with isolation dampers to isolate these areas during design bases accident conditions. SBVS also confines the volume of the fuel hall by maintaining negative pressure and removes iodine released in the event of a fuel handling accident in the FB. During fuel handling operations, a controlled and monitored ventilation system removes gaseous radioactivity from the atmosphere above the spent fuel pool and processes it through high efficiency particulate air (HEPA) filters and charcoal adsorber units of NABVS.

Details of the filter alignments for each ventilation system can be found in the following FSAR Sections: for CBVS Section 9.4.7, for AVS Section 6.2.3, for FBVS Section 9.4.2, for SBVS Section 9.4.5, for NABVS Section 9.4.3, and for RWBVS Section 9.4.8.

The filtration systems used by these ventilation units provides the final barrier against a release to the environment and consist of HEPA filters, pre-filters, adsorbers for iodine, heaters, fans, dampers, and ductwork that remove particulate and gaseous radioactive material from the atmosphere. Local instruments are provided to measure differential pressure across filters to confirm that they are effectively removing potential contaminants from the exhaust. The effectiveness of the filters is further confirmed by monitors on the air exhausted from these filtration trains. In the event of a high radioactivity level alarm, a system can be manually shut down and isolated.

These filtration systems are also designed to permit periodic inspection and periodic pressure and functional testing per ASME AG-1 (Reference 13). The filters are contained in housings and a dedicated, ventilated room to minimize the potential for facility and environmental contamination. For some units, lighting is also available inside filter banks between the rows of filters and inspection portholes in the filter housing doors to enable viewing while in operation.

All containment penetrations that introduce the potential for primary containment bypass leakage are equipped with filtered ventilation and monitoring prior to being discharged out the vent stack. These HVAC systems are located in one of the following areas:

1. Annulus.

2. Safeguard Building (mechanical areas).

3. Fuel Building.

For these ventilation systems, there are no buried pipes handling potentially contaminated exhaust gases and therefore, no means to contaminate the environment from a leaking pipe. Gases that may potentially leak from these ventilation systems upstream of the HEPA filters are collected and subsequently filtered by one of these ventilation systems, which are providing a sub-atmospheric pressure in the room where the leak may occur.

The registers of the ventilation systems are placed in each area to deliver the supply air high in the room or corridor, and to draw the air into the exhaust register high in the room or area served by the HVAC system. As a general design rule, the HVAC register placement is high above the flood plane.

The following design features are provided to prevent or mitigate contamination of the environment:

- HVAC air handling equipment is provided with drain connections at the bottom of the equipment for gravity drainage. The drainage fluid flows through the hard piping to collection tanks for processing. Drainage may also be collected in trays that have connections to the hard piping which ultimately goes to the nuclear island vent and drain system.

- Drainage from below-grade HVAC equipment is also collected in tanks and pumped to other collection tanks at a higher elevation for processing. Process piping is hard piping which ultimately goes to the nuclear island vent and drain system.

- HVAC systems are designed so that the supply air flow path is directed from clean areas to areas of increasingly higher potential contamination under normal and accident modes of operation. Potentially contaminated areas are maintained at a lower negative pressure with respect to the adjoining areas. The air from the contaminated areas is sampled and monitored for airborne radiation levels.

- Air intake connections for the HVAC systems are located on the roof or top floor of the building to prevent spread of contamination by flooding.

- Radioactive waste collection tanks and other tanks that have vent connections to HVAC system ducts are designed to prevent migration of contaminated fluid or resin to the HVAC system. The fluid or resin levels in the tanks are monitored with high level alarms. Air that is vented from the top of the tanks into the HVAC ducting system is exhausted through iodine filtration trains which are equipped with prefilter, moisture separator, electric heaters, HEPA filters and carbon adsorber. Moisture separators remove entrained moisture droplets. Heaters raise the temperature of the potentially saturated air to reduce the relative humidity to less than 70 percent in accordance with RG 1.52 or RG 1.140.

- The HVAC system configuration is designed so that the supply air flow path is directed from clean areas to areas of increasingly higher potential contamination during switchover of the ventilation modes (e.g., switching from normal operation to accident mode operation, or switching between the air filtration trains). If the HVAC system design includes a bypass of the iodine filtration train, the radiation monitors are located upstream of the iodine filtration train. This allows the filtration trains to automatically swap from bypass to iodine filtration train to prevent escape of radioactive airborne particulate to the environment. In cases where the design does not include a bypass of the iodine filtration train, the radiation monitor is located downstream of the iodine filtration train to provide an alarm in the event that the iodine filter unit fails to operate correctly.

- HVAC filters, heaters and cooling coils are located inside a housing which is designed for a minimal amount of air in-leakage to control airborne contamination. During operation, a negative pressure with respect to the adjoining environment is created in the housing. This prevents radioactive material from reaching the environment as a result of filter element failure. The housing and filters are designed, fabricated and tested in accordance with ASME AG-1.

12.3.6.5.7 Essential Service Water System

The essential service water system (ESWS) is designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2.

The ESWS is free of radioactivity resulting from plant operation. The ESWS design is consistent with the U.S. EPR contaminant management philosophy in compliance with 10 CFR 20.1406. Migration of radioactive material from potentially radioactive systems is prevented with a minimum of two barriers. The ESWS supplies water to the CCWS heat exchangers (HX) and returns the water to the ultimate heat sink (UHS) cooling tower basins. The CCWS is between the ESWS and RHRS. In addition to the CCWS/ESWS HX, there is a second HX barrier between the CCWS and RHRS. Radiation monitors in the CCWS detect radioactive contaminants migrating through the system. In addition to the two barriers between the ESWS and RHRS, and the radiation monitoring provided in the CCWS, an additional radiation monitor and sampling point are provided downstream of the CCWS HX in ESWS Trains 10/20/30/40 and the dedicated train to detect potential leakage within the heat exchanger equipment. The location of the monitors represents the closest location to the point of potential contamination. Two valves in series upstream and downstream of the CCWS HX provide full isolation of the potentially contaminated ESWS loop. To prevent spreading of contamination, consistent with 10 CFR 20.1406, isolation of the ESWS is performed in the Safeguard Building before the potentially contaminated fluid exits the building.

12.3.6.5.8 Seal Water Supply System

The seal water supply system (SEWSS) is designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2.

The SEWSS is free of radioactivity, and its design is consistent with the requirements of 10 CFR 20.1406. Radioactive material release from potentially radioactive systems is prevented with a minimum of two barriers, which are provided by an arrangement of check valves between SEWSS users and the demineralized water distribution system.

12.3.6.5.9 Safety Chilled Water System

The safety chilled water system is designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2.

A process radiation monitor is provided in Trains 1 and 4 of the SCWS, downstream of the low head safety injection (LHSI) pump mechanical seal HX, to monitor for possible leakage of radioactive fluid from the HX. The LHSI pump mechanical seal is directly cooled by reactor coolant. This process radiation monitor satisfies the requirements of GDC 64, 10 CFR 52.47(a)(6), and 10 CFR 20.1406. Other than the previously mentioned potential for possible leakage, the SCWS is free of any radioactivity resulting from plant operation. The SCWS design is consistent with the U.S. EPR contaminant management philosophy to comply with the requirements of 10 CFR 20.1406. Migration of radioactive material from potentially radioactive systems is prevented with a minimum of two barriers. The CCWS is located between the SCWS and the RHRS. In addition to the CCWS/SCWS HX, there is a second HX barrier between the CCWS and RHRS. To directly transfer contaminated water to the SCWS, two HXs must fail simultaneously. It is unlikely that two monitored systems will fail simultaneously and remain undetected. Radiation monitors are located in the CCWS to detect radioactive contamination in the system.

12.3.6.5.10 Compressed Air System

The compressed air system is designed to minimize contamination of the facility and the environment as described in the general protective design features listed in Sections 12.3.6.1 and 12.3.6.2.

The CAS design is consistent with the U.S. EPR containment philosophy to comply with the requirements of 10 CFR 20.1406. This applies to both the instrument air and the service air portions of the CAS. The minimum operating pressure of the CAS at each system interface is higher than that of the interfacing system and the air is not recycled. A pressure instrument and an isolation valve that closes to prevent backflow are located at the entrance to each NI building in which CAS has a connection with a potentially contaminated system. There is also an isolation valve in the CAS piping as it enters the NI that can isolate the CAS piping to prevent contamination spread outside of the NI. The temporary hose connections are fitted with self-sealing quick disconnect connectors, and have normally-closed, branch manual isolation valves immediately upstream of the connections, which isolate individual connections from the remainder of the CAS. The CAS is designed to prevent backflow when compressor pressure is lost by "defense in depth" consisting of sufficient capacity in the air receiver, and multiple barriers such as isolation valves.

The CAS is designed for a single unit and is not shared with other units. The CAS does not provide any breathing air. The CAS air is normally pressurized and is not recycled. The overall design configuration of the CAS and its interfaces with contaminated systems includes sufficient barriers to prevent radioactive contamination of the CAS.

12.3.6.5.11 Demineralized Water Distribution System

The demineralized water distribution system (DWDS) storage tanks complex is west of the Turbine Island as shown in Figure 1.2-3. DWDS piping configuration is not buried where it interfaces with radioactive contaminated systems. This prevents direct uncontrolled releases of radioactivity to the environment in the event of pipe wall degradation and confirms the DWDS compliance with 10 CFR 20.1406. The DWDS supplies demineralized water to the following Nuclear Island contaminated consumers:

- Coolant degasification system.
- Coolant treatment system.
- Coolant purification system.
- Nuclear island drain and vent systems.
- Coolant supply and storage system.
- Nuclear sampling system.
- Fuel pool cooling and purification system.
- Reactor boron and water makeup system.
- Chemical control system.
- Pressure relief discharge system.

- Reactor coolant pump.
- In-containment refueling water storage tank system.
- Liquid waste processing/storage system.
- Solid waste system.
- Severe accident sampling system.

The DWDS interfaces with these contaminated systems in the RB, Safeguard Buildings, Nuclear Auxiliary Building, FB, and Radioactive Waste Processing Building. The DWDS is protected from contamination by system design and multiple interface barriers. The DWDS operating pressure is higher than the interfacing systems. The system pressure differential at the interface prevents contamination of the DWDS. Contamination of the DWDS when pressure is lost is prevented by defense in depth consisting of multiple barriers such as isolating valves, check valves, air gaps, and anti-siphoning features that isolate and prevent back flow. These mechanical barriers are part of the interfacing systems or DWDS to prevent contamination from reaching the DWDS. Additional barriers are in the DWDS to prevent upstream contamination such as isolating valves and check valves located at the NI Building entrances to further prevent upstream contamination outside of the NI. The overall design configuration of the DWDS and the contaminated interfacing systems contain sufficient barriers to prevent radioactive contamination of the DWDS.

12.3.6.5.12 Piping Design Requirements

With respect to piping that contains or can potentially contain radioactive fluids, the design requirements for minimizing contamination from pipe leakage include the following:

- Pipes embedded in concrete structures are to be avoided to the extent practical.
- Concrete embedment will not be relied upon as a shielding option because pipes embedded in concrete impede inspections and repairs, and increase dose and waste during decommissioning.
- Floor drain pipes at the lowest elevation are embedded in concrete and are provided with a concentric guard pipe fitted with an alarm moisture detection monitor.
- The only pathway allowed for the discharge of radioactive liquid effluent is subsequent to treatment by the liquid waste management system. Piping outside the Radioactive Waste Processing Building (RWB) will be provided with a concentric guard pipe. The outer pipe will be fitted with an alarmed leakage detection monitor, which detects any leakages. The double pipe system will extend to the discharge pipe outlet into the cooling water outfall. Samples can be taken from the outer pipe in the RWB in case of any leakage.

- To minimize the leakage of radioactive fluids to ground water, and the leakage of ground water into buildings, system and structural designs will avoid the use of below-grade conduit and piping penetrations through walls that form exterior boundaries. This is particularly applicable to penetrations at or through the floor level.
- Penetrations through outer walls of a building containing radioactive systems will be sealed to prevent leaks to the environment. The integrity of such seals will be periodically verified.

12.3.6.5.13 Condensate and Feedwater System, Main Steam Supply System, and Auxiliary Steam System

The condensate and feedwater system (CFS) is designed to provide feedwater to the steam generators (SGs) at the required temperature, pressure, and flow rate. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low pressure feedwater heaters and the deaerator-feedwater storage tank to the main feedwater pumps, and is then pumped through the high pressure feedwater heaters to the steam generators. Except for the blowdown heat exchanger cooling water, the condensate system is located in the Turbine Building. The CFS is described in Section 10.4.7.

The main steam supply system (MSSS) conveys steam from the SGs to the high pressure turbine. The MSSS also provides steam to the second stage reheaters, deaerator pegging steam, and backup auxiliary steam. The MSSS is described in Section 10.3.

The CFS and the MSSS/auxiliary steam system are not normally radioactive. The designs are consistent with the U.S. EPR contaminant management philosophy in compliance with 10 CFR 20.1406:

- Radiation monitors detect steam generator tube ruptures (SGTR), which are described in Sections 12.3.6.1.5 and 5.2.5.5.4.
- The radiation monitors on each main steam line are located just outside the containment as shown in Figure 10.3-1—Main Steam Supply System, Sheet 1.
- MSSS design feature reduce contamination in the event of a SGTR by closing the dedicated main steam isolation valve (MSIV) to isolate the affected SG.
- CFS design features reduce contamination in the event of a SGTR by closing the main feedwater isolation valve and the full-load isolation valve to isolate the water side of the affected SG.
- For the portions outside of containment, the piping and equipment in MSSS, auxiliary steam system, and CFS are accessible for inspection. The isolation valves outside the containment are located in dedicated UJE valve rooms. The dedicated UJE valve rooms are placed in the upper section (> +64 ft) of the UJK buildings.

12.3.6.6 Operational Program to Minimize Contamination

The operational program to minimize contamination throughout the lifecycle of the facility is described in Section 12.5.

12.3.7 References

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

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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.

11. NEI 07-07, “Industry Ground Water Protection Initiative - Final Guidance Document,” Nuclear Energy Institute, August 2007.

12. ANS 57.1-1992, R1998, R2005 (R=Reaffirmed): “Design Requirements for Light Water Reactor Fuel Handling Systems,” American National Standards Institute/ American Nuclear Society, 2005.

13. ASME AG-1, "Code on Nuclear Air and Gas Treatment," The American Society of Mechanical Engineers, 1997 (including the AG-1a-2000, "Housings," Addenda).

14. MicroShield® User's Manual, Version 8, Grove Software, Inc., Lynchburg, VA, August 2009.

Table 12.3-1—Deleted

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		Range
	Continuous	ACF	
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 Severe Accident 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
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')	---	1E-5 – 1E+0 rem/hr
	1 monitor – In the active laboratory (elevation -31')	---	1E-5 – 1E+0 rem/hr
	1 monitor – In the hot workshop (elevation +64')	---	1E-5 – 1E+0 rem/hr
Radioactive Waste Processing Building	1 monitor – In the drumming room next to conveyor	---	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	
Post Accident Monitoring			
Reactor Building	4 monitors inside containment – Annual Service Space and Refueling Floor (1)	Initiates Stage 1 Containment Isolation on high radiation monitor signal inside the Reactor Building	1E-1 – 1E+7 rad/hr

Note:

1. These monitors meet the requirements of 10CFR50.34(f)(2)(xiv)(E) (TMI Action Item II.E.4.2) for containment high range radiation monitoring.

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 (R-10) in exhaust containment ventilation (upstream KLA05 filters)	---	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
	1 aerosol monitor (R-10) 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 (R-10) 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 (R-7) in exhaust air of containment ventilation (upstream KLA2 filters)	---	3E-7 – 1E-2 µCi/cc (Kr-85, Xe-133)
	1 aerosol monitor (R-7) 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	
Fuel Building (Figure 12.3-73) (continued)	1 gaseous iodine monitor (R-7) 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 (R-8) in exhaust air of containment ventilation (upstream KLA2 filters)	---	3E-9 – 3E-4 μCi/cc (H-3)
	2 noble gas monitors (R-19) air leaving fuel handling area adjacent to monitored air duct (Fuel handling area)	Refer to Table 11.5-1, Monitor R-19	1E-5 – 1E+0 rad/hr Must be capable of detecting 10 DAC-hours
	2 noble gas monitors (R-9) in air leaving containment (next to air duct) downstream KLA2 low flow purge exhaust		1E-5 – 1E+0 rad/hr
	2 noble gas accident monitors (R-27) in exhaust air from exhaust cell (downstream KLB accident exhaust filter)	Refer to Table 11.5-1, Monitor R-27	1E-4 – 1E+4 rad/hr
	2 noble gas accident monitors (R-26) in exhaust air from exhaust cell (downstream KLC accident exhaust filter)	Refer to Table 11.5-1, Monitor R-26	1E-4 – 1E+4 rad/hr
Safeguard Building (Figure 9.4.1-1)	4 monitors (R-29 and R-30) intake air of the MCR	Refer to Table 11.5-1, Footnote 19 for Monitors R-29 and R-30	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, Fuel Building (upstream KLE Filtration)	See Table 11.5-1, Monitors R-11, R-12, and R-13 (NABVS Cells 1-3), R-25 (SBVS Cell 6), R-17 and R-18 (FBVS Cells 4 and 5)	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, Fuel Building (upstream KLE Filtration)	See Table 11.5-1, Monitors R-11, R-12, and R-13 (NABVS Cells 1-3), R-25 (SBVS Cell 6), R-17 and R-18 (FBVS Cells 4 and 5)	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, Fuel Building (upstream KLE Filtration)	See Table 11.5-1, Monitors R-11, R-12, and R-13 (NABVS Cells 1-3), R-25 (SBVS Cell 6), R-17 and R-18 (FBVS Cells 4 and 5)	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
Nuclear Auxiliary Building (Figure 12.3-74) (continued)	4 radiation monitors (R-46 through R-49); 1 radiation monitor for each steam generator blowdown line	See Table 11.5-1, Footnote 7, Radiation Monitors R-46 through R-49	3E-6 – 1E-2 μCi/cc

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 (R-23) 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 (R-24) 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 (R-20 and R-22) in exhaust air from exhaust cells (Upstream KLF Room Exhaust Filters)	See Table 11.5-1, Monitor R-20 (Cell 1) and R-22 (Cell 2)	5E-4 – 3E+0 µCi 3E-10 – 1E-6 µCi/cc Must be capable of detecting 10 DAC-hours
	2 gaseous iodine monitors (R-20 and R-22) in exhaust air from exhaust cells (Upstream KLF Room Exhaust Filters)	See Table 11.5-1, Monitor R-20 (Cell 1) and R-22 (Cell 2)	5E-4 – 3E+0 µCi 3E-10 – 5E-8 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-hours (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 refueling machine	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
Sheet 1 of 2

Mission	Dose per Person	Area Dose Rate	Occupancy Time
Staffing of MCR, TSC, nearby stations			
Contained Sources (filter shine)	0.1 rem	Varies	30 day
Airborne Radioactivity (immersion/inhalation)	3.9 rem	Varies	30 day
Total	4.0 rem	N/A	30 day
Access to SAHRS System			
Contained Sources	2.7 rem	0.2 rem/hr 0.7 rem/hr 0.1 rem/hr	0.5 hr access/return 2 hr preparatory work 12 hr repair
Airborne Radioactivity (immersion)	0.7 rem	0.047 rem/hr	14.5 hr (0.5 + 2 + 12)
Total	3.4 rem	N/A	14.5 hr
Access to RHR System			
Contained Sources	2.3 rem	0.3 rem/hr 0.4 rem/hr 0.1 rem/hr	1 hr access/return 2 hr preparatory work 12 hr repair
Airborne Radioactivity (immersion)	0.7 rem	0.047 rem/hr	15 hr (1 + 2 + 12)
Total	3.0 rem	N/A	15 hr
Post-Accident Sampling (IRWST Liquid, Containment Atmosphere)			
Contained Sources	2.3 rem (1.05 rem extremity)	8.84 rem/hr (63 rem/hr extremity) 0.1 rem/hr	0.25 hr in area 1 min obtain sample 0.5 hr transport route
Airborne Radioactivity (immersion)	0.06 rem	0.08 rem/hr	0.77 hr (0.25 + 1/60 + 0.5)
Total	2.4 rem	N/A	0.77 hr
Post-Accident Sampling (Ventilation Air Sample)			
Contained Sources	0.38 rem (obtain samples) (<1 mrem extremity) <1 mrem (transport)	2.3 rem/hr (4.7 rem/hr extremity) 2.5 mrem/hr	10 min in area to obtain sample 0.22 hr transport route (access/return)
Airborne Radioactivity (immersion)	0.06 rem	0.16 rem/hr	0.39 hr (10/60 + 0.22)
Total	0.45 rem	N/A	0.39 hr

Table 12.3-12—U.S. EPR Estimated Accident Mission Dose
Sheet 2 of 2

Mission	Dose per Person	Area Dose Rate	Occupancy Time
Sample Counting Lab			
Contained Sources	1.0 rem	9.2 rem/hr (adjacent to sampling box) 100 mrem/hr (low dose-rate area)	10 min in area (about 1/3rd of time in low dose-rate area where operator moves during processing)
Airborne Radioactivity (immersion)	0.09 rem	0.52 rem/hr	10 min
Total	1.1 rem	N/A	10 min
Diesel Fuel Oil delivery (per delivery)			
Airborne Radioactivity (immersion/inhalation)	0.5 rem	0.5 rem/hr	1 hr

Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access
Sheet 1 of 3

Nuclide	Fuel Building at 1 hr post-LOCA (μCi/cc)	Safeguards Buildings at 20 hrs post-LOCA (μCi/cc)	Auxiliary Building at 1.8 hrs post-LOCA (μCi/cc)	Diesel Fuel Delivery Area at 4 days Post-LOCA (μCi/cc)
Kr-83m	7.43E-05	1.29E-05	1.82E-03	0.00E+00
Kr-85m	1.71E-04	2.45E-04	4.40E-03	1.33E-09
Kr-85	9.24E-06	2.51E-04	2.70E-04	1.74E-04
Kr-87	2.30E-04	1.99E-07	4.35E-03	0.00E+00
Kr-88	4.41E-04	1.16E-04	1.06E-02	0.00E+00
Kr-89	1.42E-09	0.00E+00	1.15E-12	0.00E+00
Xe-131m	6.77E-06	1.79E-04	1.98E-04	2.19E-04
Xe-133m	3.90E-05	8.93E-04	1.14E-03	5.84E-04
Xe-133	1.27E-03	3.22E-02	3.71E-02	2.20E-02
Xe-135m	1.25E-04	3.67E-04	5.46E-03	9.93E-05
Xe-135	4.35E-04	6.01E-03	1.27E-02	1.12E-04
Xe-137	2.13E-08	0.00E+00	1.05E-10	0.00E+00
Xe-138	5.73E-05	0.00E+00	1.60E-04	0.00E+00
Br-83	2.45E-05	9.49E-09	5.85E-06	0.00E+00
Br-84	1.64E-05	0.00E+00	1.73E-06	0.00E+00
Br-85	3.72E-11	0.00E+00	0.00E+00	0.00E+00
I-129	1.39E-11	1.33E-12	4.19E-12	1.93E-12
I-130	2.09E-05	6.88E-07	6.00E-06	1.41E-08
I-131	2.31E-04	2.07E-05	6.94E-05	2.29E-05
I-132	2.49E-04	2.59E-06	6.39E-05	2.53E-06
I-133	4.69E-04	2.38E-05	1.37E-04	2.75E-06
I-134	2.42E-04	8.80E-12	4.30E-05	0.00E+00
I-135	4.06E-04	5.30E-06	1.12E-04	2.67E-09
Rb-86	9.67E-07	7.30E-08	2.18E-07	8.79E-10
Rb-88	2.95E-04	1.19E-04	1.15E-04	0.00E+00
Rb-89	2.75E-05	0.00E+00	8.44E-07	0.00E+00
Cs-134	1.08E-04	8.39E-06	2.44E-05	1.13E-07
Cs-136	2.68E-05	2.00E-06	6.04E-06	2.29E-08
Cs-137	4.12E-05	3.21E-06	9.30E-06	4.34E-08

Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access
Sheet 2 of 3

Nuclide	Fuel Building at 1 hr post-LOCA (μCi/cc)	Safeguards Buildings at 20 hrs post-LOCA (μCi/cc)	Auxiliary Building at 1.8 hrs post-LOCA (μCi/cc)	Diesel Fuel Delivery Area at 4 days Post-LOCA (μCi/cc)
Cs-138	2.29E-04	0.00E+00	3.24E-05	0.00E+00
Sr-89	1.39E-07	9.47E-07	4.08E-06	1.81E-08
Sr-90	6.98E-09	9.98E-08	4.26E-07	1.99E-09
Sr-91	7.95E-08	2.84E-07	4.58E-06	2.21E-11
Sr-92	6.84E-08	7.59E-09	3.41E-06	0.00E+00
Y-90	1.48E-10	2.03E-08	1.27E-08	1.29E-09
Y-91m	2.72E-08	1.81E-07	2.18E-06	1.41E-11
Y-91	8.34E-10	1.77E-08	5.25E-08	3.79E-10
Y-92	1.45E-08	5.77E-08	1.31E-06	0.00E+00
Y-93	9.02E-10	3.50E-09	5.22E-08	0.00E+00
Zr-95	9.45E-10	1.34E-08	5.77E-08	2.58E-10
Zr-97	9.63E-10	6.32E-09	5.69E-08	5.58E-12
Nb-95	9.46E-10	1.35E-08	5.78E-08	2.69E-10
Mo-99	1.32E-08	1.55E-07	8.02E-07	1.39E-09
Tc-99m	1.17E-08	1.48E-07	7.15E-07	1.34E-09
Ru-103	1.25E-08	1.76E-07	7.62E-07	3.32E-09
Ru-105	8.66E-09	6.37E-09	4.67E-07	0.00E+00
Ru-106	7.39E-09	1.06E-07	4.51E-07	2.09E-09
Rh-103m	1.13E-08	1.59E-07	6.87E-07	2.99E-09
Rh-105	9.05E-09	1.01E-07	5.52E-07	4.56E-10
Rh-106	7.39E-09	1.06E-07	4.51E-07	2.09E-09
Sb-125	3.97E-09	5.67E-08	2.42E-07	1.13E-09
Sb-127	1.85E-08	2.29E-07	1.12E-06	2.58E-09
Sb-129	4.27E-08	2.90E-08	2.30E-06	0.00E+00
Te-127m	2.52E-09	3.60E-08	1.54E-07	7.14E-10
Te-127	1.85E-08	2.49E-07	1.13E-06	3.17E-09
Te-129m	7.31E-09	1.03E-07	4.46E-07	1.93E-09
Te-129	4.73E-08	1.02E-07	2.71E-06	1.26E-09
Te-131m	2.06E-08	1.90E-07	1.23E-06	6.54E-10

Table 12.3-13—Maximum Airborne Concentrations in Radiological Vital Areas at or Beyond the Time of First Access
Sheet 3 of 3

Nuclide	Fuel Building at 1 hr post-LOCA (μCi/cc)	Safeguards Buildings at 20 hrs post-LOCA (μCi/cc)	Auxiliary Building at 1.8 hrs post-LOCA (μCi/cc)	Diesel Fuel Delivery Area at 4 days Post-LOCA (μCi/cc)
Te-131	2.80E-08	4.27E-08	6.55E-07	1.47E-10
Te-132	2.03E-07	2.45E-06	1.23E-05	2.49E-08
Te-134	9.55E-08	0.00E+00	2.63E-06	0.00E+00
Ba-137m	3.90E-05	3.03E-06	8.80E-06	4.11E-08
Ba-139	6.55E-08	6.63E-11	2.67E-06	0.00E+00
Ba-140	1.04E-07	1.42E-06	6.33E-06	2.39E-08
La-140	2.81E-09	4.35E-07	2.56E-07	2.10E-08
La-141	8.35E-10	4.19E-10	4.43E-08	0.00E+00
La-142	6.20E-10	1.76E-12	2.64E-08	0.00E+00
Ce-141	2.31E-09	3.26E-08	1.41E-07	6.07E-10
Ce-143	2.31E-09	2.21E-08	1.38E-07	8.94E-11
Ce-144	1.76E-09	2.51E-08	1.07E-07	4.96E-10
Pr-143	9.37E-10	1.39E-08	5.73E-08	2.69E-10
Pr-144	1.66E-09	2.51E-08	1.06E-07	4.96E-10
Nd-147	3.89E-10	5.30E-09	2.37E-08	8.67E-11
Np-239	3.90E-08	4.42E-07	2.36E-06	3.47E-09
Pu-238	1.51E-11	2.16E-10	9.21E-10	4.30E-12
Pu-239	6.34E-13	9.10E-12	3.87E-11	0.00E+00
Pu-240	1.45E-12	2.07E-11	8.83E-11	0.00E+00
Pu-241	2.62E-10	3.75E-09	1.60E-08	7.48E-11
Am-241	1.19E-13	1.72E-12	7.27E-12	0.00E+00
Cm-242	5.41E-11	7.71E-10	3.30E-09	1.52E-11
Cm-244	2.87E-11	4.11E-10	1.75E-09	8.19E-12

Table 12.3-14—Deleted

Table 12.3-15—Component Shielding Input Parameters for the U.S. EPR Fuel Building (UFA)

Component[a,b]	Dimensions (cm) Inside Diameter	Height	Thickness	Source Volume (m^3)	Source Table	Source Density (g/cm^3)
Volume Control Tank	257	164	1.5 (top & wall)	8.51		
Water phase					Table 12.2-7	1
Gas Phase					Table 12.2-8	0.00122
Boric Acid Storage Tank	396.4	717	1.8	88.5	Table 12.2-70	1
Spent Fuel Pool Heat Exchanger[c]		590 (straight length)	2.8	6.2	Table 12.2-48	1
Inner Diameter	21.22					
Outer Diameter	117.6[d]					
Reactor Coolant Pump Decontamination Room[e]	92	–	–	–	Table 12.2-6	–
Spent Fuel Pool Air	2700	1270	–	7271	Table 12.2-48	0.00122
	Width	**Depth**	**Length**			
Spent Fuel Pool Liquid	2140	1435	1255	3854	Table 12.2-48	1

a Shell material is stainless steel density of 7.86 g/cm^3, unless otherwise noted.
b Tanks are modeled as a cylinder and any shaping for the torospherical head is neglected.
c The spent fuel pool heat exchanger modeled as an annular cylinder source. Since the heat exchanger consists of two horizontally oriented tan, dose results for sides and ends were doubled.
d Outside diameter is 117.6 cm (Inner diameter (21.22 cm) + outer cylindrical source thickness of 48.16 cm thick).
e Reactor Coolant Pump decontamination room modeled as a disc shaped surface source.

Table 12.3-16—Component Shielding Input Parameters for the U.S. EPR Radwaste Building (UKS)

Component[a]	Dimensions (cm)			Source Volume (m^3)	Source Table	Source Density (g/cm^3)
	Inside Diameter	Height	Thickness			
Resin Waste Tanks	246.4	315	1.8	15.0	Table 12.2-38	1
Resin Proportioning Tank	79.7	80	0.8	0.4		1
Concentrate Buffer Tank	237.4	226	1.3 (wall) 1.5 (top)	10.0	Table 12.2-58	1
Condensate Collection Tank	78.8	103	0.6	0.5	Table 12.2-52	1
Concentrate Tank	267.4	607	1.3 (wall) 1.5 (top)	34.1	Table 12.2-52	1
Evaporator Shell	173.4	333.33	0.8	7.87	Table 12.2-16	0.33[d]
Evaporator Tube	173.4	166.67	0.8	3.94	Table 12.2-16	
Evaporator Upper Column	158	502	1 (wall) 0.8 (top)	9.84	Table 12.2-16	6.4E-4 (steam)
Evaporator Lower Column	158	251	1 (wall) 0.8 (top)	4.92	Table 12.2-16	1
Distillate Tank	68.8	175	0.6	0.65		
Sludge Tank	83.8	91	0.6	0.5	Table 12.2-15	1
Liquid Waste Storage Tanks	327.4	831	1.3 (wall) 1.5 (top)	70.0	Table 12.2-15	1
Monitoring Tanks	327.4	831	1.3 (wall) 1.5 (top)	70.0	Table 12.2-53	1
Waste Drums	57.25	83.828	0.1214	0.215	Table 12.2-58	1

a Shell material is stainless steel (density of 7.86 g/cm^3, unless otherwise noted.

b Steam phase is 1/10000 of the liquid phase source spectra (Table 12.2-16) distributed between the evaporator shell side and the upper evaporator column uniformly.

c Liquid phase source spectra for the tube side of the evaporator and lower portion of the evaporator is distributed uniformly between the evaporator and the evaporator column.

d The liquid and steam phase sources are homogenized (tube side is water at 1 g/cm^3 and shell is steam at 6.4E-04 g/cm^3) volumetrically.

Table 12.3-17—Component Shielding Input Parameters for the U.S. EPR Auxiliary Building (UKA)
Sheet 1 of 2

Component[a]	Dimensions (cm) Inside Diameter	Height	Thickness	Source Volume (m^3)	Source Table	Source Density (g/cm^3)
Recuperative Boric Acid Cooler[g]	13.17	160	0.4	0.0218	Table 12.2-68	1
Electrical Heater	34.8	270	0.4	0.257	Table 12.2-68	1
Vapor Compressor	81.6	261	0.5	1.36	Table 12.2-66	6.04E-4
Condenser	79	529	0.5	2.59	Table 12.2-13	1.04E-3
Gas Cooler	31.4	558.5	0.5	0.432	Table 12.2-13	1.04E-3
Degasifier Column[f] Liquid	176	70	2	1.70	Table 12.2-12	1
Gas	176	224	2	5.45	Table 12.2-13	1.04E-03
Total	176	294	2	7.15		
Gas Drier	21	280	0.5	0.097	Table 12.2-17	9.05E-04
Gel Drier[d]	31.3	117	0.5	0.09		9.05E-04
Primary Coolant Purification (CPS)	–	–	–		Table 12.2-36	–
CPS Cartridge Filters	35	70	0.5	0.067		1
Equivalent steel plate above cartridges	–	–	20.5			7.85
CPS Cartridge to Filter Case[b]	–	–	7			7.85
Below Cartridge[b]	–	–	24			1
CPS Mixed Bed Ion Exchanger[c]	134.6	141	1.2	2.01	Table 12.2-10	1
Water Below Resin	–	85	–			1.22E-03
Water Above Resin	–	30	–			1.22E-03
Coolant Treatment Demineralizer	91.4	91.45	0.8	0.60	Table 12.2-64	1
Top	–	–	1.4			1.22E-03
Bottom	–	–	0.77			1.22E-03
FPCPS Mixed Bed Ion Exchanger[c]	164.8	141	1.2	3.0	Table 12.2-14	1
Water Below Resin[b]	–	–	85			1.22E-03
Water Above Resin[b]	–	–	30			1.22E-03

Table 12.3-17—Component Shielding Input Parameters for the U.S. EPR Auxiliary Building (UKA)
Sheet 2 of 2

Component[a]	Dimensions (cm) Inside Diameter	Height	Thickness	Source Volume (m^3)	Source Table	Source Density (g/cm^3)
Coolant Storage Tank[f]	394.78	939.4	2 (wall) 1 (top)	115.0	Table 12.2-60 (water phase) Table 12.2-62 (gas phase)	9.05E-4 (gas phase) 1 (water)
Charcoal Delay Bed Fill Volume	117	442	1.5	4.75	Table 12.2-17	0.52 (charcoal)
SGBS Cation Demineralizer	186.4	372.5	1.8 (Shell) 2.0 (Head)	9.04	Table 12.2-44	0.988
SGBS Mixed Bed Demineralizer	160	216.4	1.4 (Shell) 1.6 (Head)	4.2	Table 12.2-42	0.988
Overflow Tank (KBE17 BB001)	39	41.86	0.5	0.05	Table 11.1-2	1
Effluent Tank	127	316	1.5	4.0	Table 11.1-2[e]	0.699

a Shell material is stainless steel (density of 7.85 g/cm^3), unless otherwise noted.
b Boundary material is water, but is conservatively analyzed assuming air.
c Elevation of tank bottom/demineralizer above floor is 130 cm.
d Conservatively modeled as all gas. No credit for gel.
e Photon spectra is density corrected to account for differences in the RCS density of 0.699 g/cm^3 versus the density of fluid in the effluent tank of 1.0 g/cm^3 (i.e., increased by a factor of 1.43).
f Component conservatively modeled the entire volume as gas and liquid (i.e., dose results summed).
g Entire volume conservatively modeled as being boric acid column fluid and no cooling water.

Table 12.3-18—Component Shielding Input Parameters for the U.S. EPR Safeguards Building (UJH)

Component[a]	Dimensions (cm)			Source Volume (m^3)	Source Table	Source Density (g/cm^3)
	Inside Diameter	Height	Thickness			
Low Head Safety Injection (RHR) Heat Exchanger	115.3	765	1.8	7.99	Table 12.2-22 (3 hr decay)	1

a Shell material is stainless steel (density of 7.86 g/cm^3), unless otherwise noted.

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 12.3-5—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