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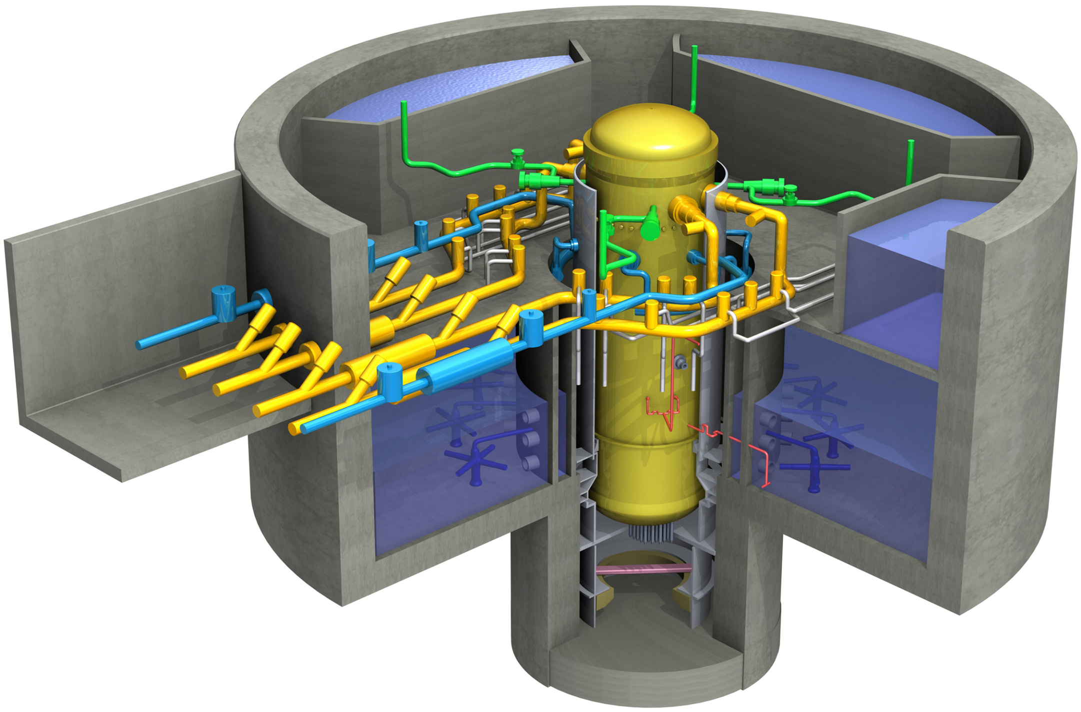

ESBWR Design Control Document
Tier 2

Chapter 12
Radiation Protection

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Abbreviations And Acronyms

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
ABWR	Advanced Boiling Water Reactor
ADS	Automatic Depressurization System
AFIP	Automated Fixed In-Core Probe
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
ARMS	Area Radiation Monitoring System
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
BAF	Bottom of Active Fuel
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CMS	Containment Monitoring System
COL	Combined Operating License
CONAVS	Reactor Building Contaminated Area HVAC Subsystem
CR	Control Rod
CRD	Control Rod Drive
DPV	Depressurization Valve
ENS	Emergency Notification System
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBFPVS	Fuel Building Fuel Pool Area HVAC Subsystem
FBGAVS	Fuel Building General Area HVAC Subsystem
FBVS	Fuel Building HVAC System
FHA	Fuel Handling Accident
FMCRD	Fine Motion Control Rod Drive
FTS	Fuel Transfer System
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GE-NE	GE Nuclear Energy
HCU	Hydraulic Control Unit
HEPA	High Efficiency Particulate Air/Absolute
HVAC	Heating, Ventilation and Air Conditioning
IC	Isolation Condenser
IFTS	Inclined Fuel Transfer System
ISI	In-Service Inspection
LOCA	Loss-of-Coolant-Accident
LPCI	Low Pressure Coolant Injection

Term	Definition
LPRM	Local Power Range Monitor
LWMS	Liquid Waste Management System
MCC	Motor Control Center
MSIV	Main Steam Isolation Valve
NBS	Nuclear Boiler System
NDE	Nondestructive Examination
NMS	Neutron Monitoring System
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
OGS	Offgas System
PAR	Passive Autocatalytic Recombiner
PCCS	Passive Containment Cooling System
PRMS	Process Radiation Monitoring System
RB	Reactor Building
RCCV	Reinforced Concrete Containment Vessel
RCCW	Reactor Component Cooling Water
RCCWS	Reactor Component Cooling Water System
REPAVS	Reactor Building Refueling and Pool Area HVAC Subsystem
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RPV	Reactor Pressure Vessel
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
SDC	Shutdown Cooling
SJAE	Steam Jet Air Ejector
SRP	Standard Review Plan
SRV	Safety Relief Valve
SWMS	Solid Waste Management System
TAF	Top of Active Fuel
USNRC	United States Nuclear Regulatory Commission
Vac / VAC	Volts Alternating Current

12. RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE ALARA

12.1.1 Policy Considerations

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel is kept as low as reasonably achievable (ALARA).

12.1.1.1 Design and Construction Policies

The ALARA (Reference 12.1-1) philosophy is applied during the initial design of the plant and implemented via internal design reviews. The design is reviewed in detail for ALARA considerations and is reviewed, updated and modified as necessary during the design phase as experience is gained from operating plants. Engineers review the plant design and integrate the layout, shielding, ventilation and monitoring instrument designs with traffic control, security, access control, and health physics aspects to ensure the overall design is conducive to maintaining exposures ALARA.

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

Operating plant results are continuously integrated during the design phase of the ESBWR Standard Plant.

12.1.1.2 Operational Policies

See Subsection 12.1.3.

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

Compliance of the ESBWR design with Title 10 of the Code of Federal Regulations, Part 20 (10 CFR 20), is ensured by the compliance of the design and operation of the facility within the guidelines of Regulatory Guides 8.8 (Reference 12.1-2), 8.10 (Reference 12.1-3), and 1.8 (Reference 12.1-4).

12.1.1.3.1 Compliance with Regulatory Guide 8.8

The policy considerations regarding plant operations contained in Regulatory Guide 8.8 are out of ESBWR Standard Plant Scope. The COL applicant will demonstrate compliance with Regulatory Guide 8.8 (COL 12.1-4-A).

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

12.1.1.3.2 Compliance with Regulatory Guide 8.10

Out of ESBWR Standard Plant scope. The COL applicant shall demonstrate compliance with Regulatory Guide 8.10 (COL 12.1-1-A).

12.1.1.3.3 Compliance with Regulatory Guide 1.8

Out of ESBWR Standard Plant scope. The COL applicant shall demonstrate compliance with Regulatory Guide 1.8 (COL 12.1-2-A).

12.1.2 Design Considerations

This subsection discusses the methods and features by which the policy considerations of Subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures ALARA are presented in detail in Subsections 12.3.1 and 12.3.2.

12.1.2.1 General Design Consideration for ALARA Exposures

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

- Minimizing the necessity for and amount of personnel time spent in radiation areas, and
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

Both equipment and facility designs are considered in maintaining exposures ALARA during plant operations. Events considered include normal operation maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, etc.

Descriptions and examples of general design features to maintain doses ALARA during normal power and shutdown operations are provided in Subsection 12.3.1.

The features of the plant design that ensure the plant can be operated and maintained with ALARA exposures also serve to assist in achieving ALARA exposures during the decommissioning process.

Examples of features that assist in maintaining low occupational exposures during decommissioning include the following:

- Provisions for draining, flushing, and decontaminating equipment and piping.
- Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of crud traps.
- Shielding which provides protection during maintenance or repairs and during decommissioning operations.
- Provision of means and adequate space for utilization of movable shielding.
- Separation of more highly radioactive equipment from less radioactive equipment and provision of separate shielded compartments for adjacent items of radioactive equipment.
- Provision for access hatches for the installation or removal of plant components.
- Provision of design features such as the Reactor Water Cleanup/Shutdown Cooling System and the condensate demineralizer to minimize crud buildup.

12.1.2.2 Equipment Design Considerations for ALARA Exposures

12.1.2.2.1 General Design Criteria

The engineering design procedures require that the component design engineer consider the applicable Regulatory Guides (including Regulatory Guide 8.8) as a part of the ALARA design criteria. In this way, the radiation problems of a component or system are considered. A summary survey of the components designs was made to determine the factors considered. The following paragraphs cite some examples of design considerations made to implement ALARA.

12.1.2.2.2 Equipment Design Considerations to Limit Time Spent in Radiation Areas

Equipment is designed to be operated and have its instrumentation and controls in accessible areas both during normal and abnormal operating conditions. Equipment such as the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) and the Fuel and Auxiliary Pool Cooling System (FAPCS) are remotely operated, including the backwashing and precoat operations.

Equipment is designed to facilitate maintenance. Equipment such as the Isolation Condenser heat exchanger is designed with an excess of tubes in order to permit plugging of some tubes. The heat exchanger has drains to allow draining of the shell-side water. Some of the valves have stem packing of the cartridge type that can be easily replaced. Refueling tools are designed for drainage and with smooth surfaces in order to reduce contamination. Vessel and piping insulation is of an easily removable type.

The materials selected for use in the system have been chosen to fulfill environmental requirements. Valves, for example, use grafoil stem packing to reduce leakage and maintenance.

Past experience has been factored into current designs. The steam relief valves have been redesigned as a result of in-service testing.

12.1.2.2.3 Equipment Design Considerations to Limit Component Radiation Levels

Equipment and piping are designed to reduce the accumulation of radioactive materials in the equipment. The piping, where possible, is constructed of seamless pipe as a means to reduce radiation accumulation on seams. The filter demineralizers in the RWCU/SDC and FAPCS are backwashed and flushed prior to maintenance.

Equipment designs include provisions for limiting leaks or controlling the fluid that does leak. This includes piping the released fluid to the sumps and using drip pans with drains piped to the floor drains.

The materials selected for use in the primary coolant system consist mainly of austenitic stainless steel, carbon steel and low alloy steel components.

The system design includes a RWCU/SDC on the reactor coolant. This system is designed to limit the radioactive isotopes in the coolant.

12.1.2.3 Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA

12.1.2.3.1 Minimizing Personnel Time Spent in Radiation Areas

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

- Locating equipment, instruments, and sampling stations, that require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas;
- Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment; and
- Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area. As an example, the ESBWR design includes a dedicated room for maintenance of the Hydraulic Control Units (HCUs). Room 1107 is designed for HCU maintenance, and its radiation zone classification in Figure 12.3-1 is lower than the radiation zone designation where the HCUs normally reside (Rooms 1110, 1120, 1130, and 1140).

12.1.2.3.2 Minimizing Radiation Levels in Plant Access Areas and Vicinity of Equipment

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

- Separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially high radioactive fluids not passing through occupied areas).
- Providing adequate shielding between radiation sources and access and service areas. Of special note, the reactor pressure vessel shield wall in the upper drywell extends to within half a meter of the upper drywell ceiling, thus permitting continued operation in the upper drywell during refueling and providing shielding in the case of a refueling accident.
- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.
- Providing central control panels to permit remote operation of all essential instrumentation and controls from the lowest radiation zone practicable. For example, the Remote Shutdown Control Panels (Rooms 1313 and 1323) reside in a Radiation Zone “A” environment, per Figure 12.3-3; the Control Rod Drive Maintenance Control Panel (Room 2202) resides in a Radiation Zone “B” environment, per Figure 12.3-2.
- Where practicable for package units, separating highly radioactive equipment from less radioactive equipment, instruments, and controls.
- Providing means and adequate space for utilizing moveable shielding for sources within the service area when required.
- Providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable.

- Providing means for decontamination of service areas.
- Providing space for pumps and valves outside of highly radioactive areas.
- Providing remotely-operated centrifugal discharge and/or back-flushable filter systems for highly radioactive radwaste and cleanup systems.
- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms.
- Providing adequate space in labyrinth entrances for easy access.
- Maintaining ventilation airflow patterns from areas of lower radioactivity to areas of higher radioactivity.

12.1.3 Operational Considerations

Out of ESBWR Standard Plant scope. COL applicants will provide the criteria and/or conditions under which various operating procedures and techniques will be implemented to ensure that occupational radiation exposures ALARA are implemented using the guidance of NUREG-1736 (Reference 12.1-5) (COL 12.1-3-A).

12.1.4 COL Information

12.1-1-A Regulatory Guide 8.10

The COL applicant will demonstrate compliance with Regulatory Guide 8.10 (Subsection 12.1.1.3.2).

12.1-2-A Regulatory Guide 1.8

The COL applicant will demonstrate compliance with Regulatory Guide 1.8 (Subsection 12.1.1.3.3).

12.1-3-A Operational Considerations

The COL applicant will provide the criteria and/or conditions under which various operating procedures and techniques will be implemented to ensure that occupational radiation exposures ALARA are implemented using the guidance of NUREG-1736 (Reference 12.1-5) (Subsection 12.1.3).

12.1-4-A Regulatory Guide 8.8

The COL applicant will demonstrate compliance with Regulatory Guide 8.8 (Subsection 12.1.1.3.1).

12.1.5 References

12.1-1 USNRC, Title 10 Code of Federal Regulations, Part 20.1101(b) ALARA.

12.1-2 USNRC, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.

12.1-3 USNRC, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, September 1975.

12.1-4 USNRC, "Qualification and Training of Personnel for Nuclear Power Plants," Regulatory Guide 1.8, Revision 3, May 2000.

12.1-5 USNRC, "Consolidated Guidance: 10CFR Part 20 – Standard for Protection Against Radiation," NUREG-1736, October 2001.

12.2 PLANT SOURCES

12.2.1 Contained Sources

12.2.1.1 Primary Containment Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR containment. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves due to deposition of corrosion or fission product species on the surfaces of the components. As such, the ESBWR unlike prior BWRs has only one significant source of radiation in the containment post operation, the reactor core. In addition, the Fine Motion Control Rod Drive (FMCRD) System provides the only other notable source of radiation in the containment. The ESBWR does not contain any recirculation pumps (external or internal), Traversing Incore Probe system, or heat exchangers that, as a function of normal usage, may become contaminated. Subsection 12.2.1.1 discusses the design of and sources found in the reactor core, while Subsection 12.2.1.2 discusses the Reactor Building source terms.

12.2.1.1.1 Reactor Vessel Core Sources

The information in this section defines a reactor vessel model and pertinent data necessary to calculate neutron and gamma fluxes inside and outside of the reactor core during normal operation. Calculation of excore particle fluxes from the reactor core during operation requires a detailed analysis of neutral particle transport, and, hence, requires the use of either a deterministic solution to the Boltzmann equation or the use of probabilistic modeling techniques. The primary source for both the neutron and gamma fluxes outside of the core is the fission process. Gammas are also created by the decay of fission products, and secondary gammas resulting from neutron absorption and scattering in structural materials both inside and outside of the core. Nuclide cross-section libraries contain gamma production data for all of these sources; therefore, it is necessary only to define the neutron fission source in the core, and then to perform a coupled neutron-gamma transport calculation. The data in this section is intended to supply adequate information to generate a neutron fission source and define geometric regions sufficient to perform a fixed source calculation using either of the methods.

Also contained in this section are post-operation gamma sources in the containment. After shutdown, the neutron fluxes are negligible and nitrogen-16 quickly decays to zero. Therefore, the most significant source is the gammas resulting from fission product decay in the reactor core.

Physical Data

Table 12.2-1 presents the physical data required to form the model in Figure 12.2-1. This model was selected to provide sufficient regions to adequately portray the reactor. The incore region was divided into 25 axial nodes, with one radial node per fuel bundle. A unique neutron fission source was determined for each of these nodes using the nodal cycle average power and exposure data. Water densities were determined at each of the 25 planes for peripheral bundles and in-core bundles. Table 12.2-1 provides nominal dimensions and material volume fractions for each boundary and region in the reactor model with core average data presented for the core. To describe the reactor core, Table 12.2-1 provides thermal power, power density, core dimensions, core average material volume fractions, and cycle average reactor power distributions and exposures. The reactor power distributions are given for both radial and axial distributions and represent the cycle averages for an equilibrium cycle.

Core Boundary and Vessel Neutron Fluxes

Table 12.2-2 presents multigroup neutron fluxes at the representative location of the core boundary and at the vessel. The multigroup neutron fluxes and the fast neutron flux (E>1 MeV) at the peak elevation of the core boundary, vessel inside surface, and ¼ thickness of the vessel are presented in Table 12.2-2, Part A. The uncertainty of the fast neutron flux at the vessel is estimated to be within ±19%. Normalized axial variations for the fast flux at the vessel inside surface are shown in Part B of Table 12.2-2.

Gamma Ray Source Energy Spectra

Table 12.2-3 presents the average gamma ray source energy spectra in both core and non-core regions. In Table 12.2-3, Part A, the energy spectrum in the core, bypass water, shroud, downcomer, and RPV is presented. This represents the average gamma ray energy released by energy group per unit volume of the region. The energy spectra in MeV per sec per cm^3 can be used with the power distributions to obtain the source in any part of the core.

The gamma ray energy spectrum includes the fission gamma rays, the fission product gamma rays, and the gamma rays resulting from inelastic neutron scattering and neutron capture. The total gamma ray energy released in the core is estimated to be accurate to within ±20%.

Post-Operation Gamma Sources

Table 12.2-3 Part B gives a gamma ray energy spectrum in MeV/sec per MW thermal in spent fuel as a function of time after operation. The data were prepared from the irradiation and decay calculation of a representative ESBWR fuel bundle to an average exposure of 35 GWd/MTU. To obtain shutdown sources in the core, the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectra and the thermal power of the element during operation.

Gamma Ray and Neutron Fluxes Outside the Vessel

Table 12.2-4 presents the maximum axial neutron and gamma ray fluxes outside the vessel. The maximum axial flux occurs typically near the core midplane elevation where the maximum power density is located for the peripheral bundle. This elevation can be located using the data from Table 12.2-1. The fluxes at this elevation represent the fluxes at the peak azimuth angle. The gamma ray calculations include gamma ray sources from all regions inside the vessel and the vessel itself.

12.2.1.1.2 Other Radioactive Sources

Radioactive Sources in the Control Rod Drive System

The control rod drive (CRD) source term data are provided in Table 12.2-5. The system is described in Subsection 3.9.4.

Reactor Startup Source

The Cf-252 reactor startup source is shipped to the site in a special cask designed with shielding. The source is transferred under water while in the cask and loaded into a stainless steel source holder. This is then loaded into the reactor while remaining under water. The source and source holder is removed from the reactor during the first refueling outage and moved to a designated location in the spent fuel pool (SFP). Operations and radiation protection personnel determine placement and duration of residence for the Cf-252 source and holder in the SFP.

12.2.1.2 Reactor Building and Fuel Building Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR reactor building. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves and pipes due to deposition of corrosion or fission products species on the surfaces of the components.

The reactor building (RB) is divided into three specific zones:

- Containment
- Contaminated areas
- Clean areas

Radioactive Sources in the Reactor Water Cleanup/Shutdown Cooling System

A description of the Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC) is given in Subsection 5.4.8. Radioactive sources contained in this system are the result of contamination of components by transit of reactor water through this system and accumulation of radioisotopes removed from the water. Components for this system include regenerative and non-regenerative heat exchangers, pumps, valves, and demineralizers. The accumulated sources in this system are given in Tables 12.2-6 and 12.2-7. The sources present in the demineralizers are present in all modes of operation. Therefore, backwashing capability is provided to remove residual activity with clean water plus chemical decontamination for effective radwaste handling.

12.2.1.2.1 Other Sources

Radioactive Sources in the Fuel and Auxiliary Pools Cooling System (FAPCS)

A description of the FAPCS is given in Subsection 9.1.3. The FAPCS is designed to service the fuel pools, suppression pool, GDCS pool, and isolation condenser/PCCS pools on a rotating basis. The accumulated activity in this system is the result of the accumulation of residual activity in each of the above pools. The filters are backwashed into a backwash receiving tank, which is then routed to the Radwaste Building systems. The sources for the FAPCS are given in Tables 12.2-8 and 12.2-9. Clean water connections are provided for this system to flush lines prior to switching between pools as necessary to prevent ancillary contamination between pools.

Radioactive Sources in the Spent Fuel Pool

The radiation sources in the spent fuel pool are given in Table 12.2-3 Part B in terms of MeV/sec-MWt. Water concentration is assumed as 1% of normal reactor water concentration (Section 11.1).

Radioactive Sources in the HVAC System

The HVAC System is described in Section 9.4 and employs a bypass HEPA filter train for use in the event of airborne contamination of the RB or controlled purge of the RB containment. The HEPA train is capable of removing all large particulate releases and up to 70% of small particulate releases.

Radioactive Sources in the Main Steam and Feedwater Lines

All radioactive material in the main steam system result from radioactive sources carried over from the reactor core during plant operations. In most components carrying live steam, N^{16} is the dominant source of radioactivity (Section 11.1). Otherwise, under conditions where sufficient decay time has removed the N^{16} source, noble radiogases become the dominant source term (Section 11.1). Flow in the feedwater lines is dominated by corrosion and fission products and is the result of the residual activity of reactor steam after treatment in the condenser filter-demineralizer system.

Post-Accident Radioactive Sources

The ESBWR design limits potential radiation exposure from accidents both to plant personnel and to the public by the use of passive safety features, containment and treatment of potential accident sources. The following describes those features of the ESBWR germane to post-accident radiation sources in the RB containment and the RB.

The RB containment is an inert steel-lined pressure boundary capable of containing all accident sources with minimal leakage to the environment or other plant areas. The containment is provided with redundant passive cooling systems (Subsections 5.4.6 and 6.2.2) to insure within a reasonable probability that this primary boundary does not exceed design criteria. Drywell spray provides additional capability to control pressure. Therefore, for all but the most improbable accident scenarios requiring massive failures of all major systems including passive systems, radioactive sources from the pressure vessel are adequately contained in the RB containment.

Surrounding the containment on all sides, the ESBWR employs a RB that provides a secondary holdup volume (Subsection 6.2.3) to trap containment penetration and valve leakage except direct bypass leakage via such lines as the main steam lines and feedwater lines. All major connections from the containment, except the isolation condenser steam lines and condensate lines and the main steam lines and feedwater lines requiring isolation valves, terminate with the second isolation valve in the RB. The RWCU/SDC is the only high energy line in the containment and RB that could produce potential releases in the containment or RB. High energy line rupture releases in the containment are isolated by the HVAC system for holdup and treatment, except potential high energy breaks, which are then routed to the turbine building for release via the plant stack. High energy line rupture releases in the RB are routed to the refueling floor where a rupture disk relieves the overpressure. See Section 15.4 for discussions of line break releases.

Estimates on sources and location for limiting design basis events are found in Section 15.4.

12.2.1.3 Turbine Building Source Terms

This section provides a summation of the significant radioactive source terms found in the ESBWR turbine building. These source terms consist of those elements which are found to contain significant quantities of radioactive materials but do not include sources due to incidental contamination such as sources in valves due to deposition of corrosion or fission products species on the surfaces of the components.

Normal Operating Sources

N^{16} in the steam flow from the pressure vessel. is the primary turbine building source of radioactivity. The N^{16} source results in significant gamma shine from the main steam lines and steam bearing components on the order of 0.2-0.5 Gy/hr (20-50 rad/hr) contact. Other major sources of radiation in the turbine building are the Offgas System (Section 11.3) and the Condenser and Feedwater System. The Offgas System consists of the steam jet air ejector, recombiner, offgas condenser, and offgas charcoal tanks. Table 12.2-10 provides the sources for the Offgas System. The sources for the turbine condenser and feedwater filter/demineralizer system are given in Tables 12.2-11 and 12.2-12.

N^{16} Skyshine Offsite Dose Contribution

The ESBWR design takes into account the hydrogen and/or noble metal injection chemistry, having conservatively used 11.1 MBq/g as the specific N^{16} activity in the vessel nozzle outlet steam. This is equivalent to using a value six times the normal value of 1.85 MBq/g.

The N^{16} skyshine contribution to offsite dose, as calculated using the SKYIII-PC code, is provided in Table 12.2-21.

Post-Accident Radioactive Sources

The turbine building contains no major sources of releasable radioactivity (discounting N^{16} because of the 7.7 second half-life) and potential releases are limited to liquid releases of low activity water from the feedwater and condenser systems. Two other sources exist which contain radioactive species but in a form not amenable for release. The potential for accident releases from these two sources, the offgas system, and the condenser demineralizers, is reduced due to heavy shielding and compartmentalizing of the components.

12.2.1.4 Radwaste Building Source Terms

The radwaste building is seismically designed in accordance with Regulatory Guide 1.143, Class RW-IIa. The tank cubicle concrete is provided with a sealant and a tank cubicle steel liner, as described in Subsection 11.2.2.3 to prevent any potential water releases from high activity areas to the environment.

Normal Operating Sources

Tables 12.2-13a through 12.2-13g and 12.2-14a through 12.2-14b provide source inventories for the major radwaste components for operation. These sources are based upon the stream concentrations given in Section 11.1 and represent sources for shielding calculations. These inventories should not be construed to represent sources for offsite release. A complete description of the ESBWR radwaste system is given in Sections 11.2 through 11.4.

Post-Accident Radioactive Sources

Potential releases in the radwaste building are contained by isolating the radwaste building atmosphere and sealing any water releases in the building. The radwaste building is seismically designed in accordance with Regulatory Guide 1.143 and the tank area concrete is provided with a sealant and a steel liner, as described in Subsection 15.3.16.1, to prevent any potential water releases from high activity areas.

12.2.2 Airborne and Liquid Sources for Environmental Consideration

This subsection deals with the models, parameters, and sources required to evaluate the airborne concentration of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected. This subsection also deals with the sources and parameters required to evaluate airborne and liquid releases during normal plant operation for compliance with 10 CFR 20 and 10 CFR 50, Appendix I criteria.

12.2.2.1 Airborne Releases Offsite

Airborne sources are calculated using the source terms given in Section 11.1. A ratio to an expected release rate is shown in Table 12.2-15 for average annual releases and subject to the criteria of Reference 12.2-1.

The bases for these calculations are shown in Table 12.2-15.

Since the ESBWR is designed for a generic site, the X/Q and D/Q values in Table 12.2-15 are the generic parameters used in the calculation of the gaseous effluent normal operation doses in Table 12.2-18b. Calculation of site-specific doses is discussed in Subsection 12.2.2.2.

Table 12.2-15 contains values used in calculating the annual airborne release source term provided in Table 12.2-16. Design basis noble gas, iodine, and other fission product concentrations are taken from the tables in Chapter 11. The methodology of NUREG-0016 was used in determining the annual airborne release values in Table 12.2-16.

Annual Releases

Based upon the above criteria, the normal operating source terms are given in Table 12.2-16 and a comparison to 10 CFR 20 criteria is given in Table 12.2-17.

12.2.2.2 Airborne Dose Evaluation Offsite

Airborne doses were calculated based upon the criteria specified in Subsection 12.2.2.1 for compliance with 10 CFR 50, Appendix I. Doses were calculated using methodologies and conversion factors consistent with Regulatory Guides 1.109 (Reference 12.2-7) and 1.111 (Reference 12.2-8) as implemented in References 12.2-1 and 12.2-2. The airborne offsite dose calculation bases are provided in Table 12.2-18a. Default parameters of Regulatory Guide 1.109 were used in determining the offsite dose, with the exception of the explicitly stated values in Table 12.2-18a. The results of the dose analysis are given in Table 12.2-18b. The COL applicant is responsible for ensuring that offsite dose (using site-specific parameters) due to radioactive airborne effluents complies with the regulatory dose limits in Sections II.B and II.C of 10 CFR 50, Appendix I. In addition, the COL applicant is responsible for compliance with Section II.D of 10 CFR 50, Appendix I; airborne effluent concentration limits of 10 CFR 20 Appendix B (Table 2, Column 1); and dose limits of 10 CFR Parts 20.1301 and 20.1302 to members of the public. (COL 12.2-2-A)

12.2.2.3 Liquid Releases Offsite

The ESBWR Radwaste System as described in Section 11.2 is designed to monitor and process all radioactive liquid streams in the ESBWR and to provide water management for those streams. Under normal conditions, the water management is not expected to result in any routine release of radioactive effluents in the liquid discharges. However, under some conditions such as high water inventory, some processed radioactive liquid effluents may be released. By administrative control, the discharge of these effluents through the discharge line is adjusted so that it can be shown that the discharge meets the requirements of 10 CFR 20 on isotopic concentration limits and Appendix I of 10 CFR 50 on annualized dose requirements.

The bounding annualized release is shown in Table 12.2-19b. Decontamination factors listed in Table 11.2-3 were used in determining the annual liquid release to the environment. The decontamination factors used were based on two in-series ion exchangers and weighted by liquid waste volume and activity for obtaining primary coolant activity values, which were used as input to the BWR-GALE computer code calculation (Reference 12.2-1). The BWR-GALE code input parameters for determining the Table 12.2-19b annual liquid release values are provided in Table 12.2-19a. These parameters are listed by cards, following the BWR-GALE format.

12.2.2.4 Liquid Doses Offsite

Liquid pathway doses were calculated based upon the criteria specified in Subsection 12.2.2.3 for compliance with 10 CFR 50, Appendix I. Dose conversion factors and methodologies consistent with Regulatory Guides 1.109 and 1.113 were used as described in References 12.2-7 and 12.2-4, respectively. The liquid effluent pathway offsite dose calculation bases are provided in Table 12.2-20a. It is assumed that an additional dilution factor of ten exists between the plant discharge point and the subsequent consumption or recreational activity involving liquid effluents. This assumption is expected to bound conditions found at actual sites. The LADTAPII code is used to perform the liquid effluent dose analysis (Reference 12.2-3). The results of the dose calculation are given in Table 12.2-20b. The COL applicant is responsible for ensuring that offsite dose (using site-specific parameters) due to radioactive liquid effluents complies with the regulatory dose limits in Section II.A of 10 CFR 50, Appendix I. In addition, the COL applicant is responsible for compliance with Section II.D of 10 CFR 50, Appendix I; liquid effluent concentration limits of 10 CFR 20 Appendix B (Table 2, Column 2); and dose limits of 10 CFR Parts 20.1301 and 20.1302 to members of the public. (COL 12.2-3-A)

12.2.3 Airborne Sources Onsite

The design focuses on keeping all radioactive material in containers. Leaks from process systems, refueling, and decontamination may lead to airborne radioactivity. Equipment cubicles, corridors, and areas routinely occupied by operating personnel do not contain significant airborne radioactivity sources. Radioactive equipment that could potentially leak is installed in separate shielded compartments not routinely occupied.

In general, airflow within the building ventilation systems is from areas of low potential for airborne contamination to areas of increasing potential. Thus, routinely occupied areas are maintained at low levels of airborne radioactivity. Data from operating BWRs corroborate the general lack of airborne activity in corridors and routinely occupied operating areas

(Reference 12.2-9). Air samples and surface contamination swipe samples are performed to verify the absence of airborne and surface contamination.

Process leakage results in potential release of noble gases and other volatile fission products via ventilation systems. Leakage of fluids from the process system results in the release of radionuclides into plant buildings. In general, the noble radiogases remain airborne and are released to the atmosphere with little delay via the building ventilation exhaust ducts. Other radionuclides partition between air and water and may plate-out on metal surfaces, concrete, and paint. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine (particulate, elemental, and hypoiodous acid forms).

12.2.3.1 Calculation of Airborne Radionuclides

See Appendix 12A.

12.2.3.2 Reactor Building

The Reactor Building HVAC system is discussed in Section 9.4.6. Subsection 12.3.3.2.3 discusses the radiation control aspects of the HVAC system.

12.2.3.2.1 Airborne Sources During Normal Operation

The main source of airborne activity in the Reactor Building is leakage of primary coolant. Therefore, airborne activities in the Reactor Building are expected to be low except for within the reactor water cleanup (RWCU) pump and valve cubicle. This cubicle is not normally occupied due to radiation levels.

The contaminated area system conditions and circulates air through the contaminated areas of the building. Flow into both areas is directed from the corridors (point of highest pressure) to the equipment alcove rooms, then to the rooms themselves, and finally to the external wall pipe chases and from the pipe chases back to the HVAC system.

Access into the containment drywell is not permitted during normal operation. The ventilation system inside merely circulates the air, without filtering it. The only airflow out of the drywell into accessible areas is minor leakage through the wall. During maintenance, the drywell air is purged before access is permitted.

As a consequence of normal steam and water leakage into the drywell, equilibrium drywell concentrations exist during normal operation. Purging of this activity from the drywell to the environment occurs via the drywell purge system, which can be routed and processed through a charcoal filtration system. These are minor contributions to total plant releases.

The assumptions and parameters used to determine the airborne activity levels in the Reactor Building are listed in Table 12.2-23a. The airborne concentrations are provided in Table 12.2-23b. Even though the values presented were obtained in a very conservative manner, they are below the limits established in 10CFR 20 Appendix B table 1 column 3.

12.2.3.2.2 Airborne Sources During Refueling

Experience at operating BWRs has shown that airborne radioactivity can result from the reactor vessel dryer and separator if their surfaces are allowed to dry. Other potential airborne sources could occur during vessel head venting and fuel movement. The airborne radioactive material sources resulting from reactor vessel head removal are minimized by venting prior to removal either to the drywell purge exhaust system or to the main condenser, with vacuum supplied by the mechanical vacuum pump. The contribution to the airborne radioactivity due to the reactor vessel internals is minimized by keeping them wet or submerged.

Airborne radioactivity during refueling is expected to be similar to that observed in operating sites. Experience has shown that airborne radioactivity can result from the water in the reactor cavity exceeding 38°C (100°F) and flaking of cobalt dioxide (CoO_2) from the steam dryer and separator if their surfaces are allowed to dry. Other potential airborne sources resulting from reactor vessel head and internals removal have been determined from experience. I-131, Co-60, Mn-54, Nb-95, Zr-95, Ru-103, and Ce-144 were the major radioisotopes found with Ce-141, Cs-137, Co-58, and Cr-51 at lower concentrations. The radioactive particulates ranged as high as 740 $\mu Bq/cm^3$ (2×10^{-8} Ci/cm^3) and I-131 as high as 1,500 $\mu Bq/cm^3$ (4×10^{-8} $\mu Ci/cm^3$).

To minimize airborne radioactivity the following actions are specified:

- Maintain steam dryer and separator surfaces wet or covered.
- Cool fuel pools through large heat capacity heat exchangers.
- Fuel pool ventilation system designed to sweep the pool surface and prevent pool releases from mixing with the area atmosphere.

12.2.3.3 Fuel Building

The Fuel Building HVAC system, including radiation control aspects of the system, is discussed in Subsections 9.4.2 and 12.3.3.2.5.

The source of airborne activity in the fuel building is in the spent fuel storage pool and equipment areas. The ventilation system is designed to sweep air from the spent fuel pool surface, thereby removing the major portion of potential airborne contamination. In addition, evaporation from the spent fuel pool is minimized by cooling of the pool.

The assumptions and parameters used to determine the airborne activity levels in the spent fuel storage pool and equipment areas are listed in Table 12.2-23a. The airborne concentrations are provided in Table 12.2-23c. Even though the values presented were obtained in a very conservative manner, they are below the limits established in 10 CFR 20 Appendix B Table 1 column 3.

12.2.3.4 Turbine Building

The Turbine Building HVAC system is discussed in Subsection 9.4.4.

The main potential source of airborne radioactivity within the Turbine Building is leakage from valves on large lines carrying high-pressure steam. The design provides for collection of this leakage and its transport back to the condenser. Therefore, noble gas airborne concentrations are expected to be negligible throughout the Turbine Building except for inside the steam jet air ejector (SJAE) cubicles. These areas are not normally occupied during operation, and the exhaust from these cubicles is exhausted to the environment after filtration to eliminate the possibility of contamination of adjoining areas.

Others sources of airborne activity in the Turbine Building atmosphere is equipment leakage.

The assumptions and parameters used to determine the airborne activity levels in the Turbine Building are listed in Table 12.2-23a. The airborne concentrations are provided in Table 12.2-23d. Even though the values presented were obtained in a very conservative manner, they are below the limits established in 10 CFR 20 Appendix B table 1 column 3.

12.2.3.5 Radwaste Building

The Radwaste Building HVAC system is discussed in Subsection 9.4.3. Subsection 12.3.3.2.4 discusses the radiation control aspects of the HVAC system.

Corridors and routine access operating areas within the Radwaste Building are not expected to have significant airborne radioactivity levels. Equipment cubicles are infrequently accessed and may contain low levels of airborne radioactivity, but design provisions are provided to minimize the release of radioactivity.

Radwaste Building tanks are filled from the top and as the water splashes into the tanks, dissolved and entrained radioactivity may become airborne. This activity is not released into the atmosphere in the rooms because the tank vents are connected directly to the building ventilation system. Pumps and valves for radioactive systems in the Radwaste Building are located in separate compartments that are not normally occupied. The Radwaste Building ventilation design provides airflow from areas of low potential for airborne contamination to areas of increasing potential. This insures that any leakage from radwaste pumps and valves is not directed into normally occupied areas of the building, but is exhausted from the building.

The assumptions and parameters used to determine the airborne activity levels in the Radwaste Building are listed in Table 12.2-23a. The airborne concentrations are provided in Table 12.2-23e. Even though the values presented were obtained in a very conservative manner, they are below the limits established in 10 CFR 20 Appendix B table 1 column 3.

12.2.4 COL Information

12.2-1-H Reactor Startup Source (Deleted)

12.2-2-A Airborne Effluents and Doses

The COL applicant is responsible for ensuring that offsite dose (using site-specific parameters) due to radioactive airborne effluents complies with the regulatory dose limits in Sections II.B and II.C of 10 CFR 50, Appendix I. In addition, the COL applicant is responsible for compliance with Section II.D of 10 CFR 50, Appendix I; airborne effluent concentration limits of 10 CFR 20 Appendix B (Table 2, Column 1); and dose limits of 10 CFR Parts 20.1301 and 20.1302 to members of the public (Subsection 12.2.2.2).

12.2-3-A Liquid Effluents and Doses

The COL applicant is responsible for ensuring that offsite dose (using site-specific parameters) due to radioactive liquid effluents complies with the regulatory dose limits in Section II.A of 10 CFR 50, Appendix I. In addition, the COL applicant is responsible for compliance with Section II.D of 10 CFR 50, Appendix I; liquid effluent concentration limits of 10 CFR 20 Appendix B (Table 2, Column 2); and dose limits of 10 CFR Parts 20.1301 and 20.1302 to members of the public (Subsection 12.2.2.4).

12.2.5 References

12.2-1 USNRC, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," NUREG-0016, Revision 1, January 1979.

12.2-2 USNRC, "GASPAR II Technical Reference and User Guide" NUREG/CR-4653, March 1987.

12.2-3 USNRC, "LADTAP II Technical Reference and User Guide" NUREG/CR-4013, April 1986.

12.2-4 USNRC, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," Regulatory Guide 1.113, Revision 1, April 1977.

12.2-5 Deleted

12.2-6 Deleted.

12.2-7 USNRC, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109, Revision 1, October 1977.

12.2-8 USNRC, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Regulatory Guide 1.111, Revision 1, July 1977.

12.2-9 Sources of Radioiodine at Boiler Water Reactors, EPRI NP-495, Research Project 274-1, Final Report, February 1978.

Table 12.2-1
Basic Reactor Data

A.	Reactor Thermal Power		4500 MW
B.	Average Power Density		54.33 kW/L
C.	Physical Dimensions		Fig. 12.2-1
	1.	Core Equivalent Radius, mm	2941.1
	2.	Inside Shroud Radius, mm	3171
	3.	Outside Shroud Radius, mm	3221
	4.	Inside Vessel Radius – Average, mm	3556
	5.	Outside Vessel Radius – Average, mm	3738
	6.	Outside Top Guide Radius, mm	3286
	7.	Vessel Top Head Inside Radius, mm	4866
	8.	Vessel Bottom Head Inside Radius, mm	4866
	9.	Bottom Head to Shell Knuckle Radius, mm	1092
	Elevation		
	10.	Outside of Vessel Bottom Head, (mm)	-263
	11.	Inside of Vessel Bottom Head, (mm)	0
	12.	Intersect of Bottom Head Radius & Vessel Wall, (mm)	2007
	13.	Bottom of Core Support Plate, (mm)	4127.6
	14.	Top of Core Support Plate, (mm)	4178.4
	15.	Bottom of Active Fuel, (mm)	4405
	16.	Top of Active Fuel, (mm)	7453
	17.	Bottom of Top Guide, (mm)	7718.2
	18.	Top of Fuel Channel, (mm)	7896.1
	19.	Normal Vessel Water Level, (mm)	20720
	20.	Top of Steam Dryer, (mm)	24775.5
	21.	Vessel Top Head Knuckle, (mm)	25648
	22.	Inside of Vessel Top Head, (mm)	27560
	23.	Outside of Vessel Top Head, (mm)	27720

Table 12.2-1
Basic Reactor Data

D. Material Densities* (g/cc)				
Region	**Coolant**	**UO_2**	**Zircaloy**	**316L Stainless**
(A) Lower Plenum	0.768	0.000	0.000	0.178
(B) Core Plate & Beam	0.338	0.000	0.000	4.35
(C) Below Active Fuel	0.597	0.000	0.166	1.70
(D1) Peripheral Assemblies				
Plane 25	0.308	1.49	1.03	0.000
Plane 24	0.310	1.78	0.77	0.000
Plane 23	0.312	1.78	0.77	0.000
Plane 22	0.316	1.78	0.77	0.000
Plane 21	0.321	1.78	0.77	0.000
Plane 20	0.326	1.78	0.77	0.000
Plane 19	0.332	1.78	0.77	0.000
Plane 18	0.340	1.78	0.77	0.000
Plane 17	0.326	1.78	1.09	0.000
Plane 16	0.333	1.78	1.09	0.000
Plane 15	0.342	2.11	0.85	0.000
Plane 14	0.352	2.11	0.85	0.000
Plane 13	0.364	2.11	0.85	0.000
Plane 12	0.377	2.11	0.85	0.000
Plane 11	0.391	2.11	0.85	0.000
Plane 10	0.407	2.11	0.85	0.000
Plane 9	0.424	2.10	0.85	0.000
Plane 8	0.440	2.10	0.85	0.000
Plane 7	0.455	2.10	0.85	0.000

Table 12.2-1
Basic Reactor Data

D. Material Densities* (g/cc)				
Region	**Coolant**	**UO_2**	**Zircaloy**	**316L Stainless**
Plane 6	0.467	2.10	0.85	0.000
Plane 5	0.474	2.10	0.85	0.000
Plane 4	0.478	2.10	0.85	0.000
Plane 3	0.480	2.10	0.85	0.000
Plane 2	0.482	2.10	0.85	0.000
Plane 1	0.483	2.17	0.85	0.000
Axial Avg	0.386	1.97	0.85	0.000
(D2) Interior Assemblies				
Plane 25	0.253	1.51	1.03	0.000
Plane 24	0.254	1.78	0.77	0.000
Plane 23	0.256	1.78	0.77	0.000
Plane 22	0.258	1.78	0.77	0.000
Plane 21	0.260	1.78	0.77	0.000
Plane 20	0.264	1.78	0.77	0.000
Plane 19	0.267	1.78	0.77	0.000
Plane 18	0.274	1.78	0.77	0.000
Plane 17	0.264	1.78	1.09	0.000
Plane 16	0.269	1.78	1.09	0.000
Plane 15	0.274	2.11	0.85	0.000
Plane 14	0.280	2.11	0.85	0.000
Plane 13	0.286	2.11	0.85	0.000
Plane 12	0.295	2.11	0.85	0.000
Plane 11	0.304	2.11	0.85	0.000
Plane 10	0.315	2.11	0.85	0.000
Plane 9	0.329	2.10	0.85	0.000
Plane 8	0.345	2.10	0.85	0.000
Plane 7	0.364	2.10	0.85	0.000
Plane 6	0.386	2.10	0.85	0.000
Plane 5	0.412	2.10	0.85	0.000

Table 12.2-1
Basic Reactor Data

D. Material Densities* (g/cc)				
Region	**Coolant**	**UO_2**	**Zircaloy**	**316L Stainless**
Plane 4	0.441	2.10	0.85	0.000
Plane 3	0.466	2.10	0.85	0.000
Plane 2	0.479	2.10	0.85	0.000
Plane 1	0.483	2.17	0.85	0.000
Axial Avg	0.323	1.97	0.85	0.000
(E) Bypass Water	0.735	0.000	0.000	0.000
(F) Above Active Fuel	0.234	0.000	1.10	0.255
(G) Top Guide	0.240	0.000	1.00	1.21
(H) Chimney	0.390	0.000	0.000	0.000
(J,K) Downcomer	0.768	0.000	0.000	0.000

* See Figure 12.2-1 for location schematic.

Table 12.2-1
Basic Reactor Data

E. Equilibrium Cycle Relative Power Distribution
Two-Dimensional Distribution at Core Midplane

Node	1	2	3	4	5	6	7	8	9	10
1										
2										
3										
4									0.51	0.65
5								0.58	0.82	0.97
6							0.58	0.85	1.04	1.14
7						0.58	0.73	1.06	1.13	1.26
8					0.58	0.85	1.06	1.19	1.31	1.22
9				0.51	0.82	1.04	1.13	1.31	1.32	1.43
10				0.65	0.97	1.14	1.26	1.22	1.43	1.30
11			0.54	0.87	1.09	1.23	1.06	1.26	1.43	1.45
12			0.65	1.01	1.21	1.22	1.23	1.15	1.47	1.33
13		0.47	0.86	1.13	1.24	1.41	1.44	1.49	1.42	1.52
14		0.61	0.99	1.21	1.36	1.30	1.51	1.49	1.56	1.38
15	0.44	0.70	1.04	1.16	1.40	1.47	1.32	1.48	1.42	1.53
16	0.51	0.90	1.06	1.25	1.42	1.32	1.41	1.47	1.57	1.53
17	0.56	0.94	1.12	1.33	1.20	1.47	1.36	1.54	1.40	1.54
18	0.58	0.94	1.17	1.31	1.42	1.31	1.48	1.37	1.52	1.36
19	0.55	0.92	0.93	1.21	1.27	1.44	1.24	1.26	1.35	1.45

Table 12.2-1
Basic Reactor Data

E. Equilibrium Cycle Relative Power Distribution (Cont.)
Two-Dimensional Distribution at Core Midplane

Node	11	12	13	14	15	16	17	18	19	
1					0.44	0.51	0.56	0.58	0.55	
2			0.47	0.61	0.70	0.90	0.94	0.94	0.92	
3	0.54	0.65	0.86	0.99	1.04	1.06	1.12	1.17	0.93	
4	0.87	1.01	1.13	1.21	1.16	1.25	1.33	1.31	1.21	
5	1.09	1.21	1.24	1.36	1.40	1.42	1.20	1.42	1.27	
6	1.23	1.22	1.41	1.30	1.47	1.32	1.47	1.31	1.44	
7	1.06	1.23	1.44	1.51	1.32	1.41	1.36	1.48	1.24	
8	1.26	1.15	1.49	1.49	1.48	1.47	1.54	1.37	1.26	
9	1.43	1.47	1.42	1.56	1.42	1.57	1.40	1.52	1.35	
10	1.45	1.33	1.52	1.38	1.53	1.53	1.54	1.36	1.45	
11	1.17	1.25	1.36	1.51	1.22	1.28	1.35	1.46	1.09	
12	1.25	1.23	1.50	1.50	1.28	1.26	1.49	1.44	1.10	
13	1.36	1.50	1.36	1.52	1.35	1.49	1.35	1.48	1.31	
14	1.51	1.50	1.52	1.35	1.47	1.33	1.50	1.35	1.46	
15	1.22	1.28	1.35	1.47	1.15	1.20	1.46	1.47	1.19	
16	1.28	1.26	1.49	1.33	1.20	1.20	1.46	1.36	1.23	
17	1.35	1.49	1.35	1.50	1.46	1.46	1.32	1.47	1.30	
18	1.46	1.44	1.48	1.35	1.47	1.36	1.47	1.30	1.40	
19	1.09	1.10	1.31	1.46	1.19	1.23	1.30	1.40	1.06	

Table 12.2-1
Basic Reactor Data

F. End of Equilibrium Cycle Average Exposure Distribution
Bundle Average Exposure (MWd/StU)

Node	1	2	3	4	5	6	7	8	9	10
1										
2										
3										
4									42135	42200
5								44916	24660	13536
6							42187	26288	14627	16072
7						42187	44815	14800	32578	17936
8					44916	26288	14800	16864	18611	37338
9				42135	24660	14627	32578	18611	35229	19936
10				42200	13536	16072	17936	37338	19936	39140
11			42236	27737	15354	17369	44183	33665	19870	19946
12			41974	13968	16968	36545	35724	44901	20370	38251
13		45216	29357	15688	34026	19729	19933	20644	38763	20957
14		41742	13389	16683	18918	38647	20493	33781	21393	39583
15	39035	42286	33302	36166	19141	19854	41822	33313	39236	20969
16	42146	29095	34717	35216	19393	39063	37227	33683	21197	20859
17	41231	28516	33232	18158	45645	20062	38553	20869	39248	21126
18	39965	27820	15960	17998	19523	38907	20278	38586	20929	38840
19	40864	12094	41543	28901	37990	19733	38479	38338	38535	20013

Table 12.2-1
Basic Reactor Data

F. End of Equilibrium Cycle Average Exposure Distribution (Cont.)
Bundle Average Exposure (MWd/StU)

Node	11	12	13	14	15	16	17	18	19	
1					39035	42146	41231	39965	40864	
2			45216	41742	42286	29095	28516	27820	12094	
3	42236	41974	29357	13389	33302	34717	33232	15960	41543	
4	27737	13968	15688	16683	36166	35216	18158	17998	28901	
5	15354	16968	34026	18918	19141	19393	45645	19523	37990	
6	17369	36545	19729	38647	19854	39063	20062	38907	19733	
7	44183	35724	19933	20493	41822	37227	38553	20278	38479	
8	33665	44901	20644	33781	33313	33683	20869	38586	38338	
9	19870	20370	38763	21393	39236	21197	39248	20929	38535	
10	19946	38251	20957	39583	20969	20859	21126	38840	20013	
11	43322	36545	38199	20683	44544	37649	38963	20135	36588	
12	36545	37338	20460	20500	37348	37800	20495	19905	35631	
13	38199	20460	39755	20863	38755	20465	39197	20501	38429	
14	20683	20500	20863	39511	20109	38489	20602	38692	20003	
15	44544	37348	38755	20109	43177	36882	19949	20152	42689	
16	37649	37800	20465	38489	36882	37064	20028	38502	37701	
17	38963	20495	39197	20602	19949	20028	39261	20324	38629	
18	20135	19905	20501	38692	20152	38502	20324	39137	19492	
19	36588	35631	38429	20003	42689	37701	38629	19492	36118	

Table 12.2-1
Basic Reactor Data

G. Axial Power Distribution		
Node	**Node Mid-Point Elevation (mm above BAF)**	**Relative Power**
25	2987.0	0.19
24	2865.1	0.38
23	2743.2	0.53
22	2621.3	0.66
21	2499.4	0.77
20	2377.4	0.86
19	2255.5	0.93
18	2133.6	0.97
17	2011.7	1.00
16	1889.8	1.01
15	1767.8	1.16
14	1645.9	1.19
13	1524.0	1.22
12	1402.1	1.24
11	1280.2	1.26
10	1158.2	1.27
9	1036.3	1.28
8	914.4	1.28
7	792.5	1.29
6	670.6	1.30
5	548.6	1.29
4	426.7	1.26
3	304.8	1.17
2	182.9	0.96
1	61.0	0.52

Table 12.2-2

Neutron Fluxes at Core Boundary and RPV

Part A. Neutron Spectrum at Peak Elevation			
Upper Energy (eV)	Core Boundary (neutrons/cm^2-sec)	RPV Inside Surface (neutrons/cm^2-sec)	RPV 1/4T Thickness (neutrons/cm^2-sec)
2.000E+7	2.7E+10	1.1E+08	5.9E+07
1.000E+7	4.8E+11	9.9E+08	5.1E+08
6.065E+6	2.0E+12	2.0E+09	1.0E+09
3.679E+6	4.0E+12	2.5E+09	1.4E+09
2.231E+6	4.3E+12	2.8E+09	2.1E+09
1.353E+6	3.8E+12	2.8E+09	2.6E+09
8.209E+5	3.8E+12	3.1E+09	3.4E+09
4.979E+5	2.6E+12	2.3E+09	2.7E+09
3.020E+5	2.2E+12	1.6E+09	1.7E+09
1.832E+5	3.2E+12	2.3E+09	2.4E+09
6.738E+4	2.4E+12	1.4E+09	1.2E+09
2.479E+4	2.1E+12	1.2E+09	9.6E+08
9.119E+3	2.0E+12	1.1E+09	7.4E+08
3.355E+3	2.0E+12	1.1E+09	7.4E+08
1.234E+3	1.9E+12	1.1E+09	6.9E+08
4.540E+2	2.0E+12	1.1E+09	6.9E+08
1.670E+2	2.0E+12	1.1E+09	7.8E+08
6.144E+1	1.8E+12	9.2E+08	4.3E+08
2.260E+1	8.4E+11	4.8E+08	2.3E+08
1.371E+1	8.4E+11	4.8E+08	2.3E+08
8.315E+0	7.4E+11	4.7E+08	2.2E+08
5.044E+0	7.8E+11	4.6E+08	2.1E+08
3.059E+0	1.5E+12	8.7E+08	3.3E+08
1.125E+0	1.4E+12	7.7E+08	2.3E+08
4.140E-1	1.1E+12	6.5E+08	1.3E+08
1.523E-1	1.5E+13	5.1E+10	3.3E+09
1.389E-4			
Total Flux	6.5E+13	8.5E+10	2.9E+10
Fast Flux (E>1 MeV)	1.3E+13	1.0E+10	6.6E+09

Table 12.2-2
Neutron Fluxes at Core Boundary and RPV

Part B. Relative Fast Flux (E>1 MeV)	
Distance from Bottom of Active Fuel (mm)	Relative Flux at RPV Inside Surface
3048.0 (TAF)	0.241
2987.0	0.301
2865.1	0.443
2743.2	0.590
2621.3	0.720
2499.4	0.822
2377.4	0.892
2255.5	0.937
2133.6	0.959
2011.7	0.974
1889.8	0.985
1767.8	0.995
1645.9	1.000
1524.0	0.996
1402.1	0.980
1280.2	0.952
1158.2	0.917
1036.3	0.873
914.4	0.823
792.5	0.767
670.6	0.706
548.6	0.640
426.7	0.567
304.8	0.481
182.9	0.380
61.0	0.269
0.0 (BAF)	0.218

Table 12.2-3

Gamma Ray Source Energy Spectra

A. Gamma Ray Sources During Operation (MeV/sec-cm^3)					
Upper Energy (MeV)	Core	Bypass Water	Shroud	Downcomer	RPV
30	0.0E+00	0.0E+00	3.7E+04	0.0E+00	1.8E+02
17	2.8E+02	7.7E+01	6.3E+05	5.5E+00	1.7E+03
12	8.8E+04	1.9E+03	8.1E+07	1.2E+02	5.1E+05
10	3.0E+05	1.3E+04	1.4E+11	6.9E+02	3.2E+08
9	3.4E+08	8.9E+06	4.6E+11	3.8E+05	3.4E+08
8	2.5E+10	2.0E+05	1.0E+12	7.1E+03	3.6E+09
7	1.1E+11	6.8E+05	2.2E+11	2.2E+04	5.8E+08
6	4.3E+11	2.4E+06	2.4E+11	6.0E+04	6.9E+08
5	1.3E+12	1.2E+07	1.7E+11	4.6E+05	5.7E+08
4	2.6E+12	6.1E+08	1.8E+11	2.1E+07	6.2E+08
3	2.5E+12	4.3E+07	8.8E+10	1.5E+06	3.5E+08
2.5	3.9E+12	7.0E+11	1.4E+11	1.6E+10	5.0E+08
2	3.0E+12	0.0E+00	1.6E+11	0.0E+00	6.1E+08
1.66	3.4E+12	0.0E+00	1.4E+11	0.0E+00	1.9E+08
1.33	3.4E+12	8.3E+07	2.7E+10	1.9E+06	1.1E+08
1	3.5E+12	7.3E+07	2.4E+11	1.7E+06	1.0E+09
0.75	3.0E+12	3.4E+05	1.7E+10	1.2E+04	4.5E+07
0.525	3.2E+11	0.0E+00	9.9E+08	0.0E+00	1.7E+06
0.5	1.7E+12	1.2E+05	1.2E+10	5.6E+03	4.2E+07
0.3	6.9E+11	9.3E+05	1.2E+09	4.3E+04	2.6E+06
0.2	5.2E+11	2.1E+06	1.6E+09	7.6E+04	6.4E+06
0.1	1.3E+11	0.0E+00	6.9E+07	0.0E+00	1.2E+05
0.06	7.4E+10	0.0E+00	1.9E+09	0.0E+00	7.9E+06
0.03	2.5E+10	0.0E+00	7.8E+08	0.0E+00	3.5E+06
0.01					
Total	3.1E+13	7.0E+11	3.3E+12	1.7E+10	9.6E+09

Table 12.2-3
Gamma Ray Source Energy Spectra

B. Post-Operation Gamma Sources in Core (MeV/sec-MWt)				
Upper Energy (MeV)	Time after Shutdown			
	0 Sec	1 Day	4 Days	1 Month
11	9.1E+09	1.0E+05	1.0E+05	9.7E+04
8	2.7E+13	6.7E+05	6.6E+05	6.2E+05
6	2.2E+15	2.5E+10	4.2E+06	3.9E+06
4	3.2E+15	2.5E+12	2.1E+12	5.7E+11
3	4.7E+15	2.2E+14	2.0E+14	4.9E+13
2.5	9.1E+15	2.3E+14	1.7E+14	5.9E+13
2	1.4E+16	3.9E+15	3.3E+15	8.3E+14
1.5	2.7E+16	1.5E+15	7.0E+14	1.9E+14
1	3.2E+16	7.8E+15	5.7E+15	3.5E+15
0.7	2.0E+16	5.5E+15	3.2E+15	1.3E+15
0.45	8.3E+15	1.2E+15	7.6E+14	1.6E+14
0.3	8.8E+15	2.4E+15	1.1E+15	1.1E+14
0.15	3.3E+15	1.7E+15	8.6E+14	1.5E+14
0.1	2.5E+15	4.8E+14	2.7E+14	4.7E+13
0.07	1.1E+15	1.8E+14	1.1E+14	4.3E+13
0.045	7.3E+14	1.8E+14	1.1E+14	3.8E+13
0.03	5.7E+14	9.9E+13	6.3E+13	2.3E+13
0.02	1.2E+15	2.9E+14	1.5E+14	3.7E+13
0				
Total	1.4E+17	2.6E+16	1.7E+16	6.5E+15

Table 12.2-4

Neutron and Gamma Ray Fluxes Outside the Vessel Wall

Neutron		Gamma Ray	
Upper Energy (eV)	Neutron Flux (neutrons/cm^2-sec)	Upper Energy (MeV)	Gamma Ray Energy Flux (MeV/cm^2-sec)
2.000E+7	8.0E+06	30	3.3E+03
1.000E+7	5.8E+07	17	2.9E+04
6.065E+6	9.8E+07	12	2.2E+06
3.679E+6	1.7E+08	10	5.7E+08
2.231E+6	3.3E+08	9	1.5E+09
1.353E+6	6.3E+08	8	5.3E+09
8.209E+5	1.0E+09	7	2.1E+09
4.979E+5	8.8E+08	6	2.3E+09
3.020E+5	5.2E+08	5	2.5E+09
1.832E+5	7.2E+08	4	2.8E+09
6.738E+4	2.4E+08	3	1.5E+09
2.479E+4	2.0E+08	2.5	2.2E+09
9.119E+3	9.7E+07	2	1.5E+09
3.355E+3	8.2E+07	1.66	1.3E+09
1.234E+3	6.9E+07	1.33	1.3E+09
4.540E+2	5.9E+07	1	1.5E+09
1.670E+2	6.9E+07	0.75	7.4E+08
6.144E+1	3.0E+07	0.525	1.0E+08
2.260E+1	1.4E+07	0.5	8.5E+08
1.371E+1	1.3E+07	0.3	4.8E+08
8.315E+0	1.2E+07	0.2	2.9E+08
5.044E+0	1.0E+07	0.1	3.3E+07
3.059E+0	1.4E+07	0.06	8.4E+05
1.125E+0	7.0E+06	0.03	6.0E+03
4.140E-1	2.6E+06	0.01	
1.523E-1	1.2E+06	Total	2.9E+10
1.389E-4			
Total	5.3E+09		

Table 12.2-5

Radioactive Sources in the Control Rod Drive System

Control Rod Drive Radiation Survey Data		
	Gamma Dose Measured at Contact, mSv/hr	
Upper Component	**Before Cleaning**	**After Cleaning**
Rotating Ball Spindle	0.0E+00	3.0E-01
Hollow Piston	7.5E-01	3.8E-01
Labyrinth Seal	6.0E-01	6.0E-01
Guide Tube	4.5E-01	3.0E-01
Outer Tube/Flange	3.3E+00	3.0E-01

Control Blade Principal Isotopes	
Isotope	**MBq/Blade**
Cr-51	5.2E+09
Mn-54	3.4E+08
Fe-55	5.9E+09
Co-58m	3.3E+08
Co-60	4.1E+09
Ni-63	1.9E+08
Total	1.6E+10

Table 12.2-6a
RWCU/SDC
Regenerative Heat Exchanger Tube Side Activity

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	1.75E+02		**Class 6**	Sr-89	7.40E+00
	I-132	1.64E+03			Sr-91	2.85E+02
	I-133	1.18E+03			Sr-92	6.77E+02
	I-134	3.03E+03			Y-91	2.95E+00
	I-135	1.70E+03			Y-92	4.13E+02
					Y-93	2.86E+02
Class 3	Rb-89	3.08E+02			Zr-95	5.93E-01
	Cs-134	1.99E+00			Nb-95	5.93E-01
	Cs-136	1.33E+00			Mo-99	1.47E+02
	Cs-137	5.30E+00			Tc-99m	1.47E+02
	Cs-138	6.19E+02			Ru-103	1.47E+00
	Ba-137m	5.30E+00			Rh-103m	1.47E+00
					Rh-106	2.21E-01
Class 4	N-16	1.09E+04			Ag-110m	7.40E-02
					Te-129m	2.95E+00
Class 6	Na-24	1.44E+02			Te-131m	7.31E+00
	Cr-51	2.21E+02			Te-132	7.35E-01
	Mn-54	2.58E+00			Ba-140	2.95E+01
	Mn-56	1.69E+03			La-140	2.95E+01
	Fe-59	2.21E+00			Ce-141	2.21E+00
	Co-58	7.40E+00			Ce-144	2.21E-01
	Co-60	1.47E+01			Pr-144	2.21E-01
	Cu-64	2.16E+02			W-187	2.18E+01
	Zn-65	7.40E+01			Np-239	5.88E+02
					Total	**2.46E+04**

Table 12.2-6b
RWCU/SDC
Non-Regenerative Heat Exchanger Tube Side Activity

Class	**Isotope**	**MBq**		**Class**	**Isotope**	**MBq**
Class 2	I-131	2.01E+02		**Class 6**	Sr-89	8.51E+00
	I-132	1.89E+03			Sr-91	3.28E+02
	I-133	1.36E+03			Sr-92	7.79E+02
	I-134	3.48E+03			Y-91	3.40E+00
	I-135	1.96E+03			Y-92	4.75E+02
					Y-93	3.29E+02
Class 3	Rb-89	3.55E+02			Zr-95	6.82E-01
	Cs-134	2.29E+00			Nb-95	6.82E-01
	Cs-136	1.53E+00			Mo-99	1.69E+02
	Cs-137	6.10E+00			Tc-99m	1.69E+02
	Cs-138	7.12E+02			Ru-103	1.70E+00
	Ba-137m	6.10E+00			Rh-103m	1.70E+00
					Rh-106	2.55E-01
Class 4	N-16	3.27E+03			Ag-110m	8.51E-02
					Te-129m	3.40E+00
Class 6	Na-24	1.66E+02			Te-131m	8.41E+00
	Cr-51	2.55E+02			Te-132	8.46E-01
	Mn-54	2.97E+00			Ba-140	3.39E+01
	Mn-56	1.95E+03			La-140	3.39E+01
	Fe-59	2.55E+00			Ce-141	2.55E+00
	Co-58	8.51E+00			Ce-144	2.55E-01
	Co-60	1.70E+01			Pr-144	2.55E-01
	Cu-64	2.48E+02			W-187	2.51E+01
	Zn-65	8.51E+01			Np-239	6.77E+02
					Total	**1.90E+04**

yep

Table 12.2-6c

RWCU/SDC

Regenerative Heat Exchanger Shell Side

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	1.02E+02		**Class 6**	Sr-89	4.29E+00
	I-132	9.51E+02			Sr-91	1.65E+02
	I-133	6.85E+02			Sr-92	3.93E+02
	I-134	1.75E+03			Y-91	1.71E+00
	I-135	9.87E+02			Y-92	2.39E+02
					Y-93	1.66E+02
Class 3	Rb-89	1.79E+02			Zr-95	3.44E-01
	Cs-134	5.78E+00			Nb-95	3.44E-01
	Cs-136	3.85E+00			Mo-99	8.50E+01
	Cs-137	1.54E+01			Tc-99m	8.50E+01
	Cs-138	1.80E+03			Ru-103	8.55E-01
	Ba-137m	3.08E+00			Rh-103m	8.55E-01
					Rh-106	1.28E-01
Class 4	N-16	1.17E+01			Ag-110m	4.29E-02
					Te-129m	1.71E+00
Class 6	Na-24	8.37E+01			Te-131m	4.24E+00
	Cr-51	1.28E+02			Te-132	4.26E-01
	Mn-54	1.50E+00			Ba-140	1.71E+01
	Mn-56	9.82E+02			La-140	1.71E+01
	Fe-59	1.28E+00			Ce-141	1.28E+00
	Co-58	4.29E+00			Ce-144	1.28E-01
	Co-60	8.55E+00			Pr-144	1.28E-01
	Cu-64	4.29E+01			W-187	1.26E+01
	Zn-65	8.37E+01			Np-239	3.41E+02
					Total	**9.41E+03**

Table 12.2-7

RWCU Demineralizer Activity

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	9.85E+06		**Class 6**	Sr-89	2.47E+06
	I-132	1.09E+06			Sr-91	8.10E+05
	I-133	6.97E+06			Sr-92	5.39E+05
	I-134	7.77E+05			Y-91	1.08E+06
	I-135	3.31E+06			Y-92	4.28E+05
					Y-93	8.64E+05
Class 3	Rb-89	2.30E+04			Zr-95	2.33E+05
	Cs-134	8.09E+05			Nb-95	1.51E+04
	Cs-136	6.41E+04			Mo-99	2.85E+06
	Cs-137	2.33E+06			Tc-99m	2.59E+05
	Cs-138	4.90E+04			Ru-103	3.94E+05
	Ba-137m	6.56E+01			Rh-103m	4.12E+02
					Rh-106	5.49E-01
Class 4	N-16	6.03E+02			Ag-110m	5.14E+04
					Te-129m	6.83E+05
Class 6	Na-24	6.36E+05			Te-131m	6.44E+04
	Cr-51	4.29E+07			Te-132	1.68E+04
	Mn-54	1.88E+06			Ba-140	2.66E+06
	Mn-56	1.28E+06			La-140	3.49E+05
	Fe-59	6.54E+05			Ce-141	4.95E+05
	Co-58	3.08E+06			Ce-144	1.58E+05
	Co-60	1.26E+07			Pr-144	1.88E+01
	Cu-64	8.11E+05			W-187	1.54E+05
	Zn-65	5.10E+07			Np-239	9.74E+06
					Total	**1.64E+08**

Table 12.2-8

FAPCS Filter Activity

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	0.00E+00		**Class 6**	Sr-89	5.34E+04
	I-132	0.00E+00			Sr-91	4.84E+04
	I-133	0.00E+00			Sr-92	3.22E+04
	I-134	0.00E+00			Y-91	2.18E+04
	I-135	0.00E+00			Y-92	2.56E+04
					Y-93	5.16E+04
Class 3	Rb-89	0.00E+00			Zr-95	4.45E+03
	Cs-134	0.00E+00			Nb-95	3.91E+03
	Cs-136	0.00E+00			Mo-99	1.70E+05
	Cs-137	0.00E+00			Tc-99m	1.55E+04
	Cs-138	0.00E+00			Ru-103	1.00E+04
	Ba-137m	0.00E+00			Rh-103m	2.46E+01
					Rh-106	3.28E-02
Class 4	N-16	0.00E+00			Ag-110m	6.21E+02
					Te-129m	1.93E+04
Class 6	Na-24	3.80E+04			Te-131m	3.85E+03
	Cr-51	1.37E+06			Te-132	1.00E+03
	Mn-54	2.19E+04			Ba-140	1.28E+05
	Mn-56	7.65E+04			La-140	2.08E+04
	Fe-59	1.55E+04			Ce-141	1.43E+04
	Co-58	5.62E+04			Ce-144	1.87E+03
	Co-60	1.28E+05			Pr-144	1.12E+00
	Cu-64	4.84E+04			W-187	9.17E+03
	Zn-65	6.21E+05			Np-239	5.82E+05
					Total	**3.59E+06**

Table 12.2-8a

FAPCS Demineralizer Activity

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	5.88E+05		**Class 6**	Sr-89	2.26E+04
	I-132	6.49E+04			Sr-91	9.68E+03
	I-133	4.16E+05			Sr-92	6.44E+03
	I-134	4.64E+04			Y-91	9.54E+03
	I-135	1.97E+05			Y-92	5.11E+03
					Y-93	1.03E+04
Class 3	Rb-89	1.37E+03			Zr-95	2.01E+03
	Cs-134	2.51E+04			Nb-95	1.45E+03
	Cs-136	3.79E+03			Mo-99	3.41E+04
	Cs-137	6.94E+04			Tc-99m	3.10E+03
	Cs-138	2.92E+03			Ru-103	3.90E+03
	Ba-137m	3.92E+00			Rh-103m	4.92E+00
					Rh-106	6.56E-03
Class 4	N-16	0.00E+00			Ag-110m	3.44E+02
					Te-129m	7.05E+03
Class 6	Na-24	7.60E+03			Te-131m	7.69E+02
	Cr-51	4.63E+05			Te-132	2.01E+02
	Mn-54	1.23E+04			Ba-140	3.16E+04
	Mn-56	1.53E+04			La-140	4.17E+03
	Fe-59	6.26E+03			Ce-141	5.16E+03
	Co-58	2.59E+04			Ce-144	1.04E+03
	Co-60	7.62E+04			Pr-144	2.24E-01
	Cu-64	9.69E+03			W-187	1.83E+03
	Zn-65	3.43E+05			Np-239	1.16E+05
					Total	**2.65E+06**

Table 12.2-8b

FAPCS Heat Exchanger Tube Side Activity

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	4.45E+00		**Class 6**	Sr-89	1.88E-01
	I-132	4.17E+01			Sr-91	7.24E+00
	I-133	3.00E+01			Sr-92	1.72E+01
	I-134	7.69E+01			Y-91	7.50E-02
	I-135	4.32E+01			Y-92	1.05E+01
					Y-93	7.26E+00
Class 3	Rb-89	7.83E+00			Zr-95	1.51E-02
	Cs-134	5.06E-02			Nb-95	1.51E-02
	Cs-136	3.37E-02			Mo-99	3.72E+00
	Cs-137	1.35E-01			Tc-99m	3.72E+00
	Cs-138	1.57E+01			Ru-103	3.75E-02
	Ba-137m	1.35E-01			Rh-103m	3.75E-02
					Rh-106	5.63E-03
Class 4	N-16	0.00E+00			Ag-110m	1.88E-03
					Te-129m	7.50E-02
Class 6	Na-24	3.67E+00			Te-131m	1.86E-01
	Cr-51	5.63E+00			Te-132	1.87E-02
	Mn-54	6.56E-02			Ba-140	7.49E-01
	Mn-56	4.30E+01			La-140	7.49E-01
	Fe-59	5.63E-02			Ce-141	5.63E-02
	Co-58	1.88E-01			Ce-144	5.63E-03
	Co-60	3.75E-01			Pr-144	5.63E-03
	Cu-64	5.48E+00			W-187	5.53E-01
	Zn-65	1.88E+00			Np-239	1.49E+01
					Total	**3.48E+02**

Table 12.2-9

FAPCS Backwash Receiving Tank Activity

Class	Isotope	MBq		Class	Isotope	MBq
Class 2	I-131	0.00E+00		**Class 6**	Sr-89	5.34E+04
	I-132	0.00E+00			Sr-91	4.84E+04
	I-133	0.00E+00			Sr-92	3.22E+04
	I-134	0.00E+00			Y-91	2.18E+04
	I-135	0.00E+00			Y-92	2.56E+04
					Y-93	5.16E+04
Class 3	Rb-89	0.00E+00			Zr-95	4.45E+03
	Cs-134	0.00E+00			Nb-95	3.91E+03
	Cs-136	0.00E+00			Mo-99	1.70E+05
	Cs-137	0.00E+00			Tc-99m	1.55E+04
	Cs-138	0.00E+00			Ru-103	1.00E+04
	Ba-137m	0.00E+00			Rh-103m	2.46E+01
					Rh-106	3.28E-02
Class 4	N-16	0.00E+00			Ag-110m	6.21E+02
					Te-129m	1.93E+04
Class 6	Na-24	3.80E+04			Te-131m	3.85E+03
	Cr-51	1.37E+06			Te-132	1.00E+03
	Mn-54	2.19E+04			Ba-140	1.28E+05
	Mn-56	7.65E+04			La-140	2.08E+04
	Fe-59	1.55E+04			Ce-141	1.43E+04
	Co-58	5.62E+04			Ce-144	1.87E+03
	Co-60	1.28E+05			Pr-144	1.12E+00
	Cu-64	4.84E+04			W-187	9.17E+03
	Zn-65	6.21E+05			Np-239	5.82E+05
					Total	**3.59E+06**

Table 12.2-10a
Offgas System
Steam Jet Air Ejector Inventory

Isotope	1st Stage Ejector (MBq)	Intercooler Condenser (MBq)	2nd Stage Ejector (MBq)
Class 1			
Kr-83m	2.00E-01	5.99E+00	9.57E-02
Kr-85m	3.39E-01	1.02E+01	1.62E-01
Kr-85	1.36E-03	4.07E-02	6.50E-04
Kr-87	1.12E+00	3.34E+01	5.34E-01
Kr-88	1.12E+00	3.35E+01	5.35E-01
Kr-89	7.12E+00	1.91E+02	3.13E+00
Xe-131m	1.12E-03	3.36E-02	5.36E-04
Xe-133m	1.66E-02	4.99E-01	7.97E-03
Xe-133	4.74E-01	1.42E+01	2.27E-01
Xe-135m	1.50E+00	4.39E+01	7.03E-01
Xe-135	1.29E+00	3.87E+01	6.18E-01
Xe-137	8.84E+00	2.42E+02	3.94E+00
Xe-138	5.11E+00	1.50E+02	2.40E+00
Class 2			
I-131	1.22E-01	3.67E+00	4.99E-02
I-132	1.15E+00	3.43E+01	4.66E-01
I-133	8.28E-01	2.48E+01	3.37E-01
I-134	2.12E+00	6.32E+01	8.59E-01
I-135	1.19E+00	3.58E+01	4.86E-01
Class 3			
Rb-89	1.08E-02	3.17E-01	4.31E-03
Cs-134	6.97E-05	2.09E-03	2.84E-05
Cs-136	4.65E-05	1.39E-03	1.89E-05
Cs-137	1.86E-04	5.57E-03	7.56E-05
Cs-138	2.17E-02	6.43E-01	8.75E-03
Ba-137m	1.86E-04	4.86E-03	6.66E-05
Class 4			
N-16	6.70E+00	5.18E+04	8.99E-01
Class-5			
H-3	5.77E+00	1.73E+02	2.35E+00
Class 6			
Na-24	5.05E-03	1.51E-01	2.06E-03

Table 12.2-10a
Offgas System
Steam Jet Air Ejector Inventory

Isotope	1st Stage Ejector (MBq)	Intercooler Condenser (MBq)	2nd Stage Ejector (MBq)
P-32	1.03E-04	3.10E-03	4.21E-05
Cr-51	7.75E-03	2.32E-01	3.16E-03
Mn-54	9.04E-05	2.71E-03	3.69E-05
Mn-56	5.92E-02	1.77E+00	2.41E-02
Fe-55	2.59E-03	7.76E-02	1.06E-03
Fe-59	7.75E-05	2.32E-03	3.16E-05
Co-58	2.59E-04	7.76E-03	1.06E-04
Co-60	5.16E-04	1.55E-02	2.10E-04
Ni-63	2.59E-06	7.76E-05	1.06E-06
Cu-64	7.55E-03	2.26E-01	3.08E-03
Zn-65	2.59E-03	7.75E-02	1.05E-03
Sr-89	2.59E-04	7.76E-03	1.06E-04
Sr-90	1.81E-05	5.43E-04	7.37E-06
Y-90	1.81E-05	5.40E-04	7.34E-06
Sr-91	9.98E-03	2.99E-01	4.07E-03
Sr-92	2.37E-02	7.09E-01	9.64E-03
Y-91	1.03E-04	3.10E-03	4.21E-05
Y-92	1.44E-02	4.32E-01	5.88E-03
Y-93	9.99E-03	3.00E-01	4.07E-03
Zr-95	2.07E-05	6.22E-04	8.45E-06
Nb-95	2.07E-05	6.22E-04	8.45E-06
Mo-99	5.13E-03	1.54E-01	2.09E-03
Tc-99m	5.13E-03	1.54E-01	2.09E-03
Ru-103	5.16E-05	1.55E-03	2.10E-05
Rh-103m	5.16E-05	1.54E-03	2.09E-05
Ru-106	7.75E-06	2.32E-04	3.16E-06
Rh-106	7.75E-06	1.16E-04	1.66E-06
Ag-110m	2.59E-06	7.76E-05	1.06E-06
Te-129m	1.03E-04	3.10E-03	4.21E-05
Te-131m	2.56E-04	7.67E-03	1.04E-04
Te-132	2.57E-05	7.72E-04	1.05E-05
Ba-140	1.03E-03	3.10E-02	4.21E-04
La-140	1.03E-03	3.10E-02	4.21E-04
Ce-141	7.75E-05	2.32E-03	3.16E-05

Table 12.2-10a
Offgas System
Steam Jet Air Ejector Inventory

Isotope	1st Stage Ejector (MBq)	Intercooler Condenser (MBq)	2nd Stage Ejector (MBq)
Ce-144	7.75E-06	2.32E-04	3.16E-06
Pr-144	7.75E-06	2.28E-04	3.10E-06
W-187	7.62E-04	2.29E-02	3.11E-04
Np-239	2.06E-02	6.17E-01	8.39E-03
Total	**4.52E+01**	**5.29E+04**	**1.79E+01**

Table 12.2-10b

Offgas System

Isotopic Inventory for Preheater through Charcoal Tanks

Isotope	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Cond. (MBq)	Dryer (MBq)	Guard Bed (MBq)	Tank 1&5 (MBq)	Tank 2&6 (MBq)	Tank 3&7 (MBq)	Tank 4&8 (MBq)
Class 1										
Kr-83m	5.87E+01	2.11E+02	2.44E+04	8.94E+02	4.39E+04	7.29E+05	1.32E+05	9.96E+01	0.00E+00	0.00E+00
Kr-85m	1.07E+02	3.84E+02	4.48E+04	1.65E+03	8.20E+04	1.91E+06	1.24E+06	6.26E+04	3.17E+03	1.60E+02
Kr-85	4.75E-01	1.71E+00	2.00E+02	7.41E+00	3.72E+02	1.14E+04	3.09E+04	3.09E+04	3.09E+04	3.09E+04
Kr-87	2.99E+02	1.08E+03	1.24E+05	4.50E+03	2.19E+05	2.86E+06	2.38E+05	6.20E+00	0.00E+00	0.00E+00
Kr-88	3.47E+02	1.25E+03	1.45E+05	5.33E+03	2.64E+05	5.29E+06	1.86E+06	1.57E+04	1.33E+02	0.00E+00
Kr-89	3.41E+00	1.22E+01	9.73E+02	2.29E+01	5.79E+02	1.60E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	3.89E-01	1.40E+00	1.64E+02	6.08E+00	3.05E+02	1.54E+05	2.64E+05	1.15E+05	5.03E+04	2.19E+04
Xe-133m	5.87E+00	2.11E+01	2.47E+03	9.15E+01	4.59E+03	1.71E+06	6.63E+05	7.69E+03	9.02E+01	0.00E+00
Xe-133	1.65E+02	5.96E+02	6.97E+04	2.58E+03	1.30E+05	5.98E+07	6.12E+07	9.35E+06	1.43E+06	2.18E+05
Xe-135m	1.33E+02	4.80E+02	5.16E+04	1.75E+03	7.50E+04	2.02E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	4.32E+02	1.56E+03	1.82E+05	6.71E+03	3.35E+05	3.77E+07	1.58E+05	0.00E+00	0.00E+00	0.00E+00
Xe-137	1.39E+01	4.97E+01	4.22E+03	1.08E+02	3.04E+03	1.21E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	4.11E+02	1.48E+03	1.58E+05	5.31E+03	2.25E+05	5.56E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Class 2										
I-131	2.89E-02	1.04E-01	1.22E+01	8.28E-04	2.66E-03	3.19E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	2.70E-01	9.73E-01	1.13E+02	7.58E-03	2.39E-02	3.34E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	1.96E-01	7.04E-01	8.23E+01	5.59E-03	1.79E-02	2.31E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	4.98E-01	1.79E+00	2.05E+02	1.35E-02	4.15E-02	2.14E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	2.82E-01	1.01E+00	1.18E+02	8.01E-03	2.56E-02	1.06E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Class 3										
Rb-89	2.49E-03	8.98E-03	9.61E-01	5.94E-05	1.62E-04	2.03E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.65E-05	5.93E-05	6.94E-03	4.71E-07	1.51E-06	1.69E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 12.2-10b

Offgas System

Isotopic Inventory for Preheater through Charcoal Tanks

Isotope	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Cond. (MBq)	Dryer (MBq)	Guard Bed (MBq)	Tank 1&5 (MBq)	Tank 2&6 (MBq)	Tank 3&7 (MBq)	Tank 4&8 (MBq)
Cs-136	1.10E-05	3.95E-05	4.63E-03	3.14E-07	1.01E-06	1.96E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	4.38E-05	1.58E-04	1.85E-02	1.25E-06	4.03E-06	6.62E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-138	5.07E-03	1.82E-02	2.05E+00	1.34E-04	3.98E-04	1.23E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	3.85E-05	1.38E-04	1.01E-02	3.88E-07	5.49E-07	4.89E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Class 4										
N-16	2.60E+00	8.35E+00	4.20E+01	2.58E-11	1.39E-12	5.24E-30	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Class 5										
H-3	1.36E+00	4.91E+00	5.75E+02	3.90E-02	1.25E-01	5.94E-01	3.39E+04	0.00E+00	0.00E+00	0.00E+00
Class 6										
Na-24	1.19E-03	4.30E-03	5.02E-01	3.41E-05	1.09E-04	1.02E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
P-32	2.44E-05	8.78E-05	1.03E-02	6.98E-07	2.24E-06	4.87E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.83E-03	6.59E-03	7.72E-01	5.24E-05	1.68E-04	6.99E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	2.14E-05	7.69E-05	9.01E-03	6.12E-07	1.96E-06	9.18E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-56	1.40E-02	5.03E-02	5.84E+00	3.93E-04	1.24E-03	1.96E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	6.11E-04	2.20E-03	2.58E-01	1.75E-05	5.62E-05	8.26E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	1.83E-05	6.59E-05	7.72E-03	5.24E-07	1.68E-06	1.13E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	6.11E-05	2.20E-04	2.58E-02	1.75E-06	5.62E-06	6.00E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	1.22E-04	4.39E-04	5.14E-02	3.49E-06	1.12E-05	3.21E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	6.11E-07	2.20E-06	2.58E-04	1.75E-08	5.62E-08	3.07E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cu-64	1.78E-03	6.42E-03	7.50E-01	5.08E-05	1.63E-04	1.29E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zn-65	6.10E-04	2.20E-03	2.57E-01	1.75E-05	5.61E-05	2.04E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	6.11E-05	2.20E-04	2.58E-02	1.75E-06	5.62E-06	4.26E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	4.27E-06	1.54E-05	1.80E-03	1.22E-07	3.93E-07	6.18E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 12.2-10b

Offgas System

Isotopic Inventory for Preheater through Charcoal Tanks

Isotope	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Cond. (MBq)	Dryer (MBq)	Guard Bed (MBq)	Tank 1&5 (MBq)	Tank 2&6 (MBq)	Tank 3&7 (MBq)	Tank 4&8 (MBq)
Y-90	4.25E-06	1.53E-05	1.79E-03	1.22E-07	3.90E-07	1.56E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-91	2.36E-03	8.49E-03	9.91E-01	6.71E-05	2.15E-04	1.29E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	5.59E-03	2.01E-02	2.34E+00	1.57E-04	4.98E-04	8.19E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.44E-05	8.80E-05	1.03E-02	6.99E-07	2.25E-06	1.96E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-92	3.41E-03	1.23E-02	1.43E+00	9.63E-05	3.06E-04	6.65E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	2.36E-03	8.50E-03	9.93E-01	6.73E-05	2.15E-04	1.35E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	4.90E-06	1.76E-05	2.07E-03	1.40E-07	4.51E-07	4.38E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	4.90E-06	1.76E-05	2.07E-03	1.40E-07	4.51E-07	2.35E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99	1.21E-03	4.36E-03	5.11E-01	3.47E-05	1.11E-04	4.60E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc-99m	1.21E-03	4.36E-03	5.09E-01	3.44E-05	1.10E-04	4.06E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.22E-05	4.39E-05	5.14E-03	3.49E-07	1.12E-06	6.64E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh-103m	1.21E-05	4.27E-05	1.38E-03	9.82E-09	4.59E-09	3.67E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.83E-06	6.59E-06	7.72E-04	5.24E-08	1.68E-07	9.24E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh-106	9.58E-07	3.35E-06	7.35E-05	1.32E-10	3.97E-11	1.27E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ag-110m	6.11E-07	2.20E-06	2.58E-04	1.75E-08	5.62E-08	2.12E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m	2.44E-05	8.80E-05	1.03E-02	6.99E-07	2.25E-06	1.14E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-131m	6.04E-05	2.18E-04	2.54E-02	1.73E-06	5.54E-06	1.03E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	6.08E-06	2.19E-05	2.56E-03	1.74E-07	5.58E-07	2.70E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	2.44E-04	8.78E-04	1.03E-01	6.98E-06	2.24E-05	4.28E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-140	2.44E-04	8.78E-04	1.03E-01	6.97E-06	2.24E-05	5.59E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.83E-05	6.59E-05	7.72E-03	5.24E-07	1.68E-06	8.15E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	1.83E-06	6.59E-06	7.72E-04	5.24E-08	1.68E-07	7.14E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.80E-06	6.46E-06	7.02E-04	4.41E-08	1.23E-07	1.91E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 12.2-10b

Offgas System

Isotopic Inventory for Preheater through Charcoal Tanks

Isotope	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Cond. (MBq)	Dryer (MBq)	Guard Bed (MBq)	Tank 1&5 (MBq)	Tank 2&6 (MBq)	Tank 3&7 (MBq)	Tank 4&8 (MBq)
W-187	1.80E-04	6.49E-04	7.58E-02	5.15E-06	1.65E-05	2.45E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Np-239	4.86E-03	1.75E-02	2.05E+00	1.39E-04	4.46E-04	1.56E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	1.98E+03	7.13E+03	8.08E+05	2.90E+04	1.38E+06	1.11E+08	6.58E+07	9.58E+06	1.51E+06	2.71E+05

Table 12.2-11

Turbine Condenser Inventory

Isotope	Activity (MBq)	Isotope	Activity (MBq)
Class 1		**Class 6**	
Kr-83m	8.82E+03	Na-24	5.32E+01
Kr-85m	1.49E+04	P-32	1.09E+00
Kr-85	5.98E+01	Cr-51	8.16E+01
Kr-87	4.93E+04	Mn-54	9.52E-01
Kr-88	4.93E+04	Mn-56	6.24E+02
Kr-89	3.14E+05	Fe-55	2.73E+01
Xe-131m	4.93E+01	Fe-59	8.16E-01
Xe-133m	7.32E+02	Co-58	2.73E+00
Xe-133	2.09E+04	Co-60	5.44E+00
Xe-135m	6.58E+04	Ni-63	2.73E-02
Xe-135	5.68E+04	Cu-64	7.95E+01
Xe-137	3.89E+05	Zn-65	2.73E+01
Xe-138	2.25E+05	Sr-89	2.73E+00
Total	**1.21E+06**	Sr-90	1.90E-01
Class 2		Y-90	1.90E-01
I-131	1.29E+03	Sr-91	1.05E+02
I-132	1.21E+04	Sr-92	2.50E+02
I-133	8.72E+03	Y-91	1.09E+00
I-134	2.23E+04	Y-92	1.52E+02
I-135	1.26E+04	Y-93	1.05E+02
Total	**5.70E+04**	Zr-95	2.18E-01
Class 3		Nb-95	2.18E-01
Rb-89	1.14E+02	Mo-99	5.40E+01
Cs-134	7.34E-01	Tc-99m	5.40E+01
Cs-136	4.89E-01	Ru-103	5.44E-01
Cs-137	1.95E+00	Rh-103m	5.44E-01
Cs-138	2.28E+02	Ru-106	8.16E-02
Ba-137m	1.95E+00	Rh-106	8.16E-02
Total	**3.47E+02**	Ag-110m	2.73E-02
Class 4		Te-129m	1.09E+00
N-16	1.26E+08	Te-131m	2.69E+00
Class 5		Te-132	2.71E-01
H-3	6.08E+04	Ba-140	1.09E+01
		La-140	1.09E+01
		Ce-141	8.16E-01
		Ce-144	8.16E-02
		Pr-144	8.16E-02
		W-187	8.03E+00
		Np-239	2.17E+02
		Total	**1.27E+08**

Table 12.2-12

Isotopic Inventory in the Ion Exchanger Filters

Isotope	Activity (MBq)
Class 2	
I-131	1.94E+06
I-132	2.15E+05
I-133	1.41E+06
I-134	1.52E+05
I-135	6.57E+05
Class 3	
Rb-89	2.28E+02
Cs-134	9.10E+03
Cs-136	1.18E+03
Cs-137	2.52E+04
Cs-138	9.55E+02
Ba-137m	6.48E-01
Class 6	
Sr-89	3.86E+03
Sr-90	4.93E+02
Y-90	3.17E-01
Sr-91	1.58E+03
Sr-92	1.05E+03
Y-91	1.66E+03
Y-92	8.39E+02
Y-93	1.66E+03
Zr-95	3.50E+02
Nb-95	2.47E+02
Mo-99	5.61E+03
Tc-99m	5.06E+02
Ru-103	6.68E+02
Rh-103m	7.93E-01
Ru-106	1.93E+02
Rh-106	1.06E-03
Te-129m	1.21E+03
Te-131m	1.26E+02
Te-132	3.30E+01
Ba-140	5.18E+03
La-140	6.82E+02
Ce-141	8.76E+02
Ce-144	1.88E+02
Pr-144	3.67E-02
Np-239	1.91E+04
Total	**4.47E+06**

Table 12.2-13a

Liquid Waste Management System Equipment Drain Collection Tank Activity

Source Volume = 140 m^3

Class	**Isotope**	**Activity Conc.** (MBq/m^3)		**Class**	**Isotope**	**Activity Conc.** (MBq/m^3)
Class 2	I-131	6.89E+02		**Class 6**	Sr-89	1.43E+02
	I-132	6.58E+01			Sr-90	2.23E+01
	I-133	5.51E+02			Y-90	6.95E-01
	I-134	4.38E+01			Sr-91	5.68E+01
	I-135	2.19E+02			Sr-92	3.25E+01
					Y-91	6.28E+01
Class 3	Rb-89	1.25E+00			Y-92	2.67E+01
	Cs-134	7.36E+01			Y-93	5.98E+01
	Cs-136	7.25E+00			Zr-95	1.34E+01
	Cs-137	2.09E+02			Nb-95	8.76E+00
	Cs-138	5.62E+00			Mo-99	2.07E+02
	Ba-137m	3.71E-03			Tc-99m	1.72E+01
					Ru-103	2.39E+01
Class 5	H-3	9.73E+01			Rh-103m	2.33E-02
					Ru-106	8.17E+00
Class 6	Na-24	4.74E+01			Rh-106	2.95E-05
	P-32	1.98E+01			Ag-110m	2.67E+00
	Cr-51	2.61E+03			Te-129m	4.29E+01
	Mn-54	9.83E+01			Te-131m	4.85E+00
	Mn-56	7.59E+01			Te-132	1.21E+00
	Fe-55	3.08E+03			Ba-140	1.75E+02
	Fe-59	3.82E+01			La-140	2.62E+01
	Co-58	1.76E+02			Ce-141	2.97E+01
	Co-60	6.25E+02			Ce-144	7.86E+00
	Ni-63	3.24E+00			Pr-144	1.03E-03
	Cu-64	5.92E+01			W-187	1.15E+01
	Zn-65	2.65E+03			Np-239	7.17E+02
					Total	**1.31E+04**

Table 12.2-13b

Liquid Waste Management System Equipment Drain Sample Tank Activity

Source Volume = 140 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	5.72E-01		**Class 6**	Sr-89	1.39E-01
	I-132	1.01E-08			Sr-90	2.23E-02
	I-133	9.84E-02			Y-90	3.98E-04
	I-134	6.03E-20			Sr-91	1.40E-03
	I-135	1.05E-03			Sr-92	5.31E-08
					Y-91	0.00E+00
Class 3	Rb-89	0.00E+00			Y-92	0.00E+00
	Cs-134	7.34E-01			Y-93	0.00E+00
	Cs-136	6.47E-02			Zr-95	0.00E+00
	Cs-137	2.09E+00			Nb-95	0.00E+00
	Cs-138	0.00E+00			Mo-99	1.21E-01
	Ba-137m	0.00E+00			Tc-99m	4.37E-05
					Ru-103	0.00E+00
Class 5	H-3	9.73E+01			Rh-103m	0.00E+00
					Ru-106	0.00E+00
Class 6	Na-24	0.00E+00			Rh-106	0.00E+00
	P-32	0.00E+00			Ag-110m	0.00E+00
	Cr-51	0.00E+00			Te-129m	4.11E-02
	Mn-54	0.00E+00			Te-131m	1.47E-03
	Mn-56	0.00E+00			Te-132	7.62E-04
	Fe-55	0.00E+00			Ba-140	1.56E-01
	Fe-59	0.00E+00			La-140	0.00E+00
	Co-58	0.00E+00			Ce-141	0.00E+00
	Co-60	0.00E+00			Ce-144	0.00E+00
	Ni-63	0.00E+00			Pr-144	0.00E+00
	Cu-64	0.00E+00			W-187	0.00E+00
	Zn-65	0.00E+00			Np-239	3.80E-01
					Total	**1.02E+02**

Table 12.2-13c

Liquid Waste Management System Floor Drain Collection Tank Activity

Source Volume = 130 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	3.24E-01		**Class 6**	Sr-89	1.58E-02
	I-132	9.72E-02			Sr-90	1.13E-03
	I-133	6.32E-01			Y-90	6.26E-04
	I-134	6.74E-02			Sr-91	7.07E-02
	I-135	2.88E-01			Sr-92	4.74E-02
					Y-91	6.36E-03
Class 3	Rb-89	1.97E-03			Y-92	3.80E-02
	Cs-134	4.36E-03			Y-93	7.40E-02
	Cs-136	2.55E-03			Zr-95	1.27E-03
	Cs-137	1.16E-02			Nb-95	1.25E-03
	Cs-138	8.76E-03			Mo-99	1.83E-01
	Ba-137m	5.86E-06			Tc-99m	2.29E-02
					Ru-103	3.10E-03
Class 5	H-3	3.64E-01			Rh-103m	3.58E-05
					Ru-106	4.71E-04
Class 6	Na-24	5.60E-02			Rh-106	4.66E-08
	P-32	5.82E-03			Ag-110m	1.63E-04
	Cr-51	4.44E-01			Te-129m	6.22E-03
	Mn-54	5.79E-03			Te-131m	5.34E-03
	Mn-56	1.11E-01			Te-132	9.99E-04
	Fe-55	1.63E-01			Ba-140	5.72E-02
	Fe-59	4.55E-03			La-140	2.74E-02
	Co-58	1.60E-02			Ce-141	4.48E-03
	Co-60	3.23E-02			Ce-144	4.70E-04
	Ni-63	1.64E-04			Pr-144	1.61E-06
	Cu-64	7.11E-02			W-187	1.30E-02
	Zn-65	1.63E-01			Np-239	6.79E-01
					Total	**4.14E+00**

Table 12.2-13d

Liquid Waste Management System Floor Drain Sample Tank Activity

Source Volume = 130 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	2.09E-05		**Class 6**	Sr-89	1.48E-06
	I-132	0.00E+00			Sr-90	1.13E-07
	I-133	1.08E-06			Y-90	1.68E-08
	I-134	0.00E+00			Sr-91	1.12E-09
	I-135	9.56E-11			Sr-92	1.01E-19
					Y-91	0.00E+00
Class 3	Rb-89	0.00E+00			Y-92	0.00E+00
	Cs-134	2.17E-05			Y-93	0.00E+00
	Cs-136	9.72E-06			Zr-95	0.00E+00
	Cs-137	5.82E-05			Nb-95	0.00E+00
	Cs-138	0.00E+00			Mo-99	5.15E-06
	Ba-137m	0.00E+00			Tc-99m	1.71E-12
					Ru-103	0.00E+00
Class 5	H-3	3.63E-02			Rh-103m	0.00E+00
					Ru-106	0.00E+00
Class 6	Na-24	0.00E+00			Rh-106	0.00E+00
	P-32	0.00E+00			Ag-110m	0.00E+00
	Cr-51	0.00E+00			Te-129m	5.61E-07
	Mn-54	0.00E+00			Te-131m	3.19E-08
	Mn-56	0.00E+00			Te-132	3.37E-08
	Fe-55	0.00E+00			Ba-140	4.34E-06
	Fe-59	0.00E+00			La-140	0.00E+00
	Co-58	0.00E+00			Ce-141	0.00E+00
	Co-60	0.00E+00			Ce-144	0.00E+00
	Ni-63	0.00E+00			Pr-144	0.00E+00
	Cu-64	0.00E+00			W-187	0.00E+00
	Zn-65	0.00E+00			Np-239	1.52E-05
					Total	**3.66E-02**

Table 12.2-13e

Liquid Waste Management System Chemical Collection Tank Activity

Source Volume = 4 m^3

Class	**Isotope**	**Activity Conc.** (MBq/m^3)		**Class**	**Isotope**	**Activity Conc.** (MBq/m^3)
Class 2	I-131	7.56E+00		**Class 6**	Sr-89	3.25E-01
	I-132	7.42E+00			Sr-90	2.26E-02
	I-133	3.22E+01			Y-90	1.91E-02
	I-134	5.14E+00			Sr-91	4.85E+00
	I-135	2.12E+01			Sr-92	3.62E+00
					Y-91	1.30E-01
Class 3	Rb-89	1.50E-01			Y-92	2.90E+00
	Cs-134	8.73E-02			Y-93	5.02E+00
	Cs-136	5.62E-02			Zr-95	2.60E-02
	Cs-137	2.33E-01			Nb-95	2.59E-02
	Cs-138	6.68E-01			Mo-99	5.51E+00
	Ba-137m	4.47E-04			Tc-99m	1.71E+00
					Ru-103	6.40E-02
Class 5	H-3	7.28E+00			Rh-103m	2.73E-03
					Ru-106	9.45E-03
Class 6	Na-24	3.31E+00			Rh-106	3.55E-06
	P-32	1.27E-01			Ag-110m	3.27E-03
	Cr-51	9.31E+00			Te-129m	1.29E-01
	Mn-54	1.16E-01			Te-131m	2.26E-01
	Mn-56	8.48E+00			Te-132	2.85E-02
	Fe-55	3.27E+00			Ba-140	1.26E+00
	Fe-59	9.36E-02			La-140	1.01E+00
	Co-58	3.25E-01			Ce-141	9.33E-02
	Co-60	6.48E-01			Ce-144	9.45E-03
	Ni-63	3.27E-03			Pr-144	1.23E-04
	Cu-64	4.48E+00			W-187	6.17E-01
	Zn-65	3.27E+00			Np-239	2.17E+01
					Total	**1.65E+02**

Table 12.2-13f

Liquid Waste Management System Detergent Collection Tank Activity

Source Volume = 15 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	1.71E+01		**Class 6**	Sr-89	7.98E-01
	I-132	6.59E+00			Sr-90	5.64E-02
	I-133	4.15E+01			Y-90	3.61E-02
	I-134	4.57E+00			Sr-91	4.79E+00
	I-135	1.96E+01			Sr-92	3.22E+00
					Y-91	3.20E-01
Class 3	Rb-89	1.33E-01			Y-92	2.58E+00
	Cs-134	2.18E-01			Y-93	5.01E+00
	Cs-136	1.32E-01			Zr-95	6.42E-02
	Cs-137	5.82E-01			Nb-95	6.31E-02
	Cs-138	5.94E-01			Mo-99	1.05E+01
	Ba-137m	3.98E-04			Tc-99m	1.56E+00
					Ru-103	1.57E-01
Class 5	H-3	1.82E+01			Rh-103m	2.43E-03
					Ru-106	2.36E-02
Class 6	Na-24	3.76E+00			Rh-106	3.16E-06
	P-32	3.00E-01			Ag-110m	8.14E-03
	Cr-51	2.26E+01			Te-129m	3.15E-01
	Mn-54	2.90E-01			Te-131m	3.37E-01
	Mn-56	7.54E+00			Te-132	5.63E-02
	Fe-55	8.17E+00			Ba-140	2.96E+00
	Fe-59	2.30E-01			La-140	1.67E+00
	Co-58	8.04E-01			Ce-141	2.27E-01
	Co-60	1.62E+00			Ce-144	2.35E-02
	Ni-63	8.19E-03			Pr-144	1.09E-04
	Cu-64	4.80E+00			W-187	8.40E-01
	Zn-65	8.14E+00			Np-239	3.96E+01
					Total	**2.43E+02**

Table 12.2-13g

Liquid Waste Management System Detergent Sample Tank Activity

Source Volume = 15 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	1.24E+01		**Class 6**	Sr-89	7.58E-01
	I-132	9.11E-12			Sr-90	5.64E-02
	I-133	2.07E+00			Y-90	1.36E-02
	I-134	0.00E+00			Sr-91	7.60E-03
	I-135	1.79E-03			Sr-92	2.72E-10
					Y-91	0.00E+00
Class 3	Rb-89	0.00E+00			Y-92	0.00E+00
	Cs-134	2.17E-01			Y-93	0.00E+00
	Cs-136	1.08E-01			Zr-95	0.00E+00
	Cs-137	5.82E-01			Nb-95	0.00E+00
	Cs-138	0.00E+00			Mo-99	4.12E+00
	Ba-137m	0.00E+00			Tc-99m	4.75E-05
					Ru-103	0.00E+00
Class 5	H-3	1.82E+01			Rh-103m	0.00E+00
					Ru-106	0.00E+00
Class 6	Na-24	0.00E+00			Rh-106	0.00E+00
	P-32	0.00E+00			Ag-110m	0.00E+00
	Cr-51	0.00E+00			Te-129m	2.92E-01
	Mn-54	0.00E+00			Te-131m	4.23E-02
	Mn-56	0.00E+00			Te-132	2.53E-02
	Fe-55	0.00E+00			Ba-140	2.42E+00
	Fe-59	0.00E+00			La-140	0.00E+00
	Co-58	0.00E+00			Ce-141	0.00E+00
	Co-60	0.00E+00			Ce-144	0.00E+00
	Ni-63	0.00E+00			Pr-144	0.00E+00
	Cu-64	0.00E+00			W-187	0.00E+00
	Zn-65	0.00E+00			Np-239	1.31E+01
					Total	**5.44E+01**

Table 12.2-14a

Solid Waste Management System High Activity Resin Holdup Tank Activity

Source Volume = 70 m^3

Class	**Isotope**	**Activity Conc.** (MBq/m^3)		**Class**	**Isotope**	**Activity Conc.** (MBq/m^3)
Class 2	I-131	5.01E+05		**Class 6**	Sr-90	
	I-132	5.53E+04			Y-90	
	I-133	3.54E+05			Sr-91	4.12E+04
	I-134	3.95E+04			Sr-92	2.74E+04
	I-135	1.68E+05			Y-91	7.07E+04
					Y-92	2.17E+04
Class 3	Rb-89	1.17E+03			Y-93	4.39E+04
	Cs-134	1.10E+05			Zr-95	1.60E+04
	Cs-136	3.26E+03			Nb-95	1.04E+03
	Cs-137	3.49E+05			Mo-99	1.45E+05
	Cs-138	2.49E+03			Tc-99m	1.32E+04
	Ba-137m	3.33E+00			Ru-103	2.28E+04
					Rh-103m	2.09E+01
Class 6	Na-24	3.23E+04			Ru-106	
	P-32				Rh-106	2.79E-02
	Cr-51	2.29E+06			Ag-110m	5.80E+03
	Mn-54	2.23E+05			Te-129m	3.79E+04
	Mn-56	6.51E+04			Te-131m	3.27E+03
	Fe-55				Te-132	8.56E+02
	Fe-59	3.92E+04			Ba-140	1.35E+05
	Co-58	2.20E+05			La-140	1.77E+04
	Co-60	1.82E+06			Ce-141	2.72E+04
	Ni-63				Ce-144	1.84E+04
	Cu-64	4.12E+04			Pr-144	9.53E-01
	Zn-65	5.70E+06			W-187	7.81E+03
	Sr-89	1.54E+05			Np-239	4.95E+05
					Total	**1.33E+07**

Table 12.2-14b

Solid Waste Management System Low Activity Resin Holdup Tank Activity

Source Volume = 70 m^3

Class	**Isotope**	**Activity Conc.** (MBq/m^3)		**Class**	**Isotope**	**Activity Conc.** (MBq/m^3)
Class 2	I-131	1.46E+05		**Class 6**	Sr-90	
	I-132	4.37E-04			Y-90	
	I-133	5.17E+03			Sr-91	6.19E+01
	I-134	2.60E-15			Sr-92	2.29E-03
	I-135	4.56E+01			Y-91	1.61E+04
					Y-92	9.25E-03
Class 3	Rb-89	1.61E-63			Y-93	1.55E+01
	Cs-134	3.87E+05			Zr-95	3.66E+03
	Cs-136	3.47E+03			Nb-95	1.58E+03
	Cs-137	1.34E+06			Mo-99	1.25E+04
	Cs-138	3.18E-27			Tc-99m	1.89E+00
	Ba-137m	0.00E+00			Ru-103	4.72E+03
					Rh-103m	4.51E-18
Class 6	Na-24	4.17E+01			Ru-106	
	P-32				Rh-106	0.00E+00
	Cr-51	3.89E+05			Ag-110m	1.49E+03
	Mn-54	5.99E+04			Te-129m	3.78E+04
	Mn-56	6.24E-04			Te-131m	9.12E+01
	Fe-55				Te-132	8.93E+01
	Fe-59	8.28E+03			Ba-140	6.10E+04
	Co-58	5.09E+04			La-140	1.58E+02
	Co-60	5.52E+05			Ce-141	5.04E+03
	Ni-63				Ce-144	4.61E+03
	Cu-64	3.27E+01			Pr-144	9.14E-57
	Zn-65	1.46E+06			W-187	2.85E+01
	Sr-89	1.68E+05			Np-239	3.49E+04
					Total	**9.50E+06**

Table 12.2-14c

Solid Waste Management System Low Activity Sludge Phase Separator Tank Activity

Source Volume = 55 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	0.00E+00		**Class 6**	Sr-90	
	I-132	0.00E+00			Y-90	
	I-133	0.00E+00			Sr-91	2.83E+04
	I-134	0.00E+00			Sr-92	1.86E+04
	I-135	0.00E+00			Y-91	1.24E+05
					Y-92	1.50E+04
Class 3	Rb-89	0.00E+00			Y-93	3.07E+04
	Cs-134	0.00E+00			Zr-95	2.68E+04
	Cs-136	0.00E+00			Nb-95	1.68E+04
	Cs-137	0.00E+00			Mo-99	9.25E+04
	Cs-138	0.00E+00			Tc-99m	9.03E+03
	Ba-137m	0.00E+00			Ru-103	4.61E+04
					Rh-103m	1.39E+01
Class 6	Na-24	2.51E+04			Ru-106	
	P-32				Rh-106	1.17E-03
	Cr-51	4.95E+06			Ag-110m	7.34E+03
	Mn-54	2.86E+05			Te-129m	4.60E+03
	Mn-56	4.50E+04			Te-131m	2.24E+03
	Fe-55				Te-132	5.20E+02
	Fe-59	7.45E+04			Ba-140	3.89E+04
	Co-58	3.56E+05			La-140	2.20E+04
	Co-60	2.38E+06			Ce-141	5.69E+04
	Ni-63				Ce-144	2.24E+04
	Cu-64	3.04E+04			Pr-144	6.05E-01
	Zn-65	7.21E+06			W-187	7.23E+03
	Sr-89	1.19E+04			Np-239	3.25E+05
					Total	**1.63E+07**

Table 12.2-14d

Solid Waste Management System Condensate Resin Holdup Tank Activity

Source Volume = 70 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	2.30E+05		**Class 6**	Sr-90	
	I-132	2.54E+04			Y-90	
	I-133	1.67E+05			Sr-91	1.87E+02
	I-134	1.80E+04			Sr-92	1.24E+02
	I-135	7.78E+04			Y-91	1.96E+02
					Y-92	9.93E+01
Class 3	Rb-89	2.70E+01			Y-93	1.96E+02
	Cs-134	1.08E+03			Zr-95	4.14E+01
	Cs-136	1.40E+02			Nb-95	2.92E+01
	Cs-137	2.98E+03			Mo-99	6.64E+02
	Cs-138	1.13E+02			Tc-99m	5.99E+01
	Ba-137m	7.67E-02			Ru-103	7.91E+01
					Rh-103m	9.38E-02
Class 6	Na-24	0.00E+00			Ru-106	
	P-32				Rh-106	1.25E-04
	Cr-51	0.00E+00			Ag-110m	0.00E+00
	Mn-54	0.00E+00			Te-129m	1.43E+02
	Mn-56	0.00E+00			Te-131m	1.49E+01
	Fe-55				Te-132	3.91E+00
	Fe-59	0.00E+00			Ba-140	6.13E+02
	Co-58	0.00E+00			La-140	8.07E+01
	Co-60	0.00E+00			Ce-141	1.04E+02
	Ni-63				Ce-144	2.22E+01
	Cu-64	0.00E+00			Pr-144	4.34E-03
	Zn-65	0.00E+00			W-187	0.00E+00
	Sr-89	4.57E+02			Np-239	2.26E+03
					Total	**5.27E+05**

Table 12.2-14e

Solid Waste Management System Concentrate Waste Tank Activity

Source Volume = 60 m^3

Class	Isotope	Activity Conc. (MBq/m^3)		Class	Isotope	Activity Conc. (MBq/m^3)
Class 2	I-131	1.38E+00		**Class 6**	Sr-90	
	I-132	1.83E-17			Y-90	
	I-133	2.57E-02			Sr-91	2.62E-05
	I-134	1.01E-43			Sr-92	2.37E-15
	I-135	2.24E-06			Y-91	4.73E-02
					Y-92	7.31E-13
Class 3	Rb-89	8.31E-156			Y-93	8.17E-06
	Cs-134	6.19E-01			Zr-95	1.05E-02
	Cs-136	1.91E-02			Nb-95	5.50E-03
	Cs-137	1.93E+00			Mo-99	1.67E-01
	Cs-138	5.87E-71			Tc-99m	3.99E-08
	Ba-137m	0.00E+00			Ru-103	1.56E-02
					Rh-103m	7.80E-45
Class 6	Na-24	9.47E-05			Ru-106	
	P-32				Rh-106	0.00E+00
	Cr-51	1.54E+00			Ag-110m	3.43E-03
	Mn-54	1.33E-01			Te-129m	1.34E-01
	Mn-56	3.10E-16			Te-131m	7.94E-04
	Fe-55				Te-132	1.19E-03
	Fe-59	2.61E-02			Ba-140	4.22E-01
	Co-58	1.44E-01			La-140	1.78E-03
	Co-60	1.02E+00			Ce-141	1.83E-02
	Ni-63				Ce-144	1.04E-02
	Cu-64	4.36E-05			Pr-144	2.17E-134
	Zn-65	3.37E+00			W-187	1.82E-04
	Sr-89	5.11E-01			Np-239	4.57E-01
					Total	**1.20E+01**

Table 12.2-15

Airborne Sources Calculation

Calculation Bases	
Noble Gas Source at t=30 min	740 MBq/sec (20,000 μCi/sec)
I^{131} Release Rate	3.7 MBq/sec (100 μCi/sec)
Meteorology χ/Q	2.0E-06 s/m^3
Meteorology D/Q	4.0E-09 m^{-2}
Meteorology Boundary	800 m
Plant Availability Factor	0.92
Offgas System:	
Offgas stream temperature	100°F
Flow rate at 100°F	54 m^3/hr
K_d (Kr)	19 cm^3/g
K_d (Xe)	330 cm^3/g
K_d (Ar)	6 cm^3/g
Guard tank charcoal mass	7,500 kg (single tank)
Adsorber tank charcoal mass	27,750 kg (each)
Adsorber tank arrangement	2 parallel trains of 4 tanks each
Turbine Gland Sealing System Exhaust:	
I-131 release	0.81 Ci/yr per μCi/g of I-131 in coolant
I-133 release	0.22 Ci/yr per μCi/g of I-131 in coolant

Table 12.2-16

Annual Airborne Releases for Offsite Dose Evaluations (MBq)

Nuclide	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Drywell
Kr-83m						1.4E-04	3.7E+01
Kr-85m	6.9E+04	5.7E+05				6.8E+03	1.5E+02
Kr-85						4.3E+06	3.3E+01
Kr-87	4.6E+04	1.4E+06				8.6E-10	1.4E+02
Kr-88	9.2E+04	2.1E+06				1.5E+01	3.0E+02
Kr-89	4.6E+04	1.3E+07	6.7E+05				3.7E+01
Kr-90							1.3E+01
Xe-131m						1.1E+05	1.8E+01
Xe-133m						8.1E-01	8.5E+01
Xe-133	2.5E+06	3.4E+06	5.1E+06	1.9E+07		8.3E+05	5.0E+03
Xe-135m	1.4E+06	9.2E+06	1.2E+07				3.7E+01
Xe-135	2.9E+06	7.6E+06	6.4E+06	7.4E+06		4.3E-37	1.2E+03
Xe-137	4.1E+06	2.3E+07	1.9E+06				5.5E+01
Xe-138	1.8E+05	2.3E+07	4.6E+04				1.2E+02
Xe-139							1.6E+01
I-131	9.4E+02	5.2E+03	3.4E+02	1.8E+03	4.7E+01		6.8E+03
I-132	8.5E+03	4.6E+04	3.0E+03				9.9E+02
I-133	6.2E+03	3.4E+04	2.2E+03		8.4E+01		6.5E+03
I-134	1.5E+04	8.4E+04	5.5E+03				6.9E+02
I-135	8.6E+03	4.7E+04	3.1E+03				2.9E+03
H-3	1.3E+06	1.3E+06					2.6E+05
C-14						3.5E+05	
Na-24							5.4E-01
P-32							1.3E-01
Ar-41						2.9E+02	
Cr-51	2.7E+01	2.2E+01	1.7E+01				1.1E+01
Mn-54	3.4E+01	1.5E+01	9.8E+01				1.7E-01
Mn-56							1.1E+00
Fe-55							4.7E+00

Table 12.2-16

Annual Airborne Releases for Offsite Dose Evaluations (MBq)

Nuclide	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Drywell
Fe-59	9.5E+00	2.4E+00	7.3E+00				1.2E-01
Co-58	7.3E+00	2.4E+01	4.9E+00				4.4E-01
Co-60	1.2E+02	2.4E+01	1.7E+02				9.4E-01
Ni-63							4.7E-03
Cu-64							6.9E-01
Zn-65	1.2E+02	1.5E+02	7.3E+00				4.6E+00
Rb-89							2.0E-02
Sr-89	1.2E+00	1.5E+02					4.3E-01
Sr-90	2.4E-01	4.9E-01					3.3E-02
Y-90							3.3E-02
Sr-91							6.7E-01
Sr-92							4.6E-01
Y-91							1.7E-01
Y-92							3.7E-01
Y-93							7.2E-01
Zr-95	2.4E+01	9.8E-01	2.0E+01				3.5E-02
Nb-95	2.4E+02	1.5E-01	9.8E-02				3.3E-02
Mo-99	1.6E+03	4.9E+01	7.3E-02				2.4E+00
Tc-99m							2.2E-01
Ru-103	1.0E+02	1.2E+00	2.4E-02				8.2E-02
Rh-103m							8.2E-02
Ru-106							1.4E-02
Rh-106							1.4E-02
Ag-110m	5.9E-02						1.3E-07
Sb-124	1.2E+00	2.4E+00	1.7E+00				
Te-129m							1.6E-01
Te-131m							5.5E-02
Te-132							1.4E-02
Cs-134	1.1E+02	4.9E+00	5.9E+01				1.3E-01
Cs-136	1.2E+01	2.4E+00					5.8E-02
Cs-137	1.5E+02	2.4E+01	9.8E+01				3.4E-01
Cs-138							8.5E-02
Ba-140	5.4E+02	2.4E+02	9.8E-02				1.3E+00
La-140							1.3E+00

Table 12.2-16

Annual Airborne Releases for Offsite Dose Evaluations (MBq)

Nuclide	Reactor Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Drywell
Ce-141	2.2E+01	2.4E+02	1.7E-01				1.2E-01
Ce-144							1.3E-02
Pr-144							1.3E-02
W-187							1.3E-01
Np-239							8.3E+00

Table 12.2-17
Comparison of Airborne Concentrations with 10 CFR 20 Concentrations

	Airborne Release	Concentration	10 CFR 20
Nuclide	**MBq/yr**	**Bq/m^3**	**Bq/m^3**
Kr-83m	3.73E+01	2.36E-06	2.E+06
Kr-85m	6.50E+05	4.12E-02	4.E+03
Kr-85	4.29E+06	2.72E-01	3.E+04
Kr-87	1.45E+06	9.17E-02	7.E+02
Kr-88	2.18E+06	1.38E-01	3.E+02
Kr-89	1.40E+07	8.90E-01	4.E+01
Kr-90	1.25E+01	7.94E-07	4.E+01
Xe-131m	1.10E+05	6.97E-03	7.E+04
Xe-133m	8.59E+01	5.44E-06	2.E+04
Xe-133	3.11E+07	1.97E+00	2.E+04
Xe-135m	2.27E+07	1.44E+00	1.E+03
Xe-135	2.43E+07	1.54E+00	3.E+03
Xe-137	2.90E+07	1.84E+00	4.E+01
Xe-138	2.32E+07	1.47E+00	7.E+02
Xe-139	1.57E+01	9.93E-07	4.E+01
I-131	1.51E+04	9.57E-04	7.E+00
I-132	5.89E+04	3.74E-03	7.E+02
I-133	4.88E+04	3.09E-03	4.E+01
I-134	1.06E+05	6.72E-03	2.E+03
I-135	6.14E+04	3.89E-03	2.E+02
H-3	2.80E+06	1.78E-01	4.E+03
C-14	3.54E+05	2.24E-02	1.E+02
Na-24	5.42E-01	3.44E-08	3.E+02
P-32	1.34E-01	8.50E-09	2.E+01
Ar-41	2.85E+02	1.81E-05	4.E+02
Cr-51	7.73E+01	4.90E-06	1.E+03
Mn-54	1.47E+02	9.29E-06	4.E+01
Mn-56	1.07E+00	6.80E-08	7.E+02

Table 12.2-17

Comparison of Airborne Concentrations with 10 CFR 20 Concentrations

	Airborne Release	Concentration	10 CFR 20
Nuclide	MBq/yr	Bq/m^3	Bq/m^3
Fe-55	4.72E+00	2.99E-07	1.E+02
Fe-59	1.94E+01	1.23E-06	2.E+01
Co-58	3.70E+01	2.35E-06	4.E+01
Co-60	3.18E+02	2.02E-05	2.E+00
Ni-63	4.74E-03	3.01E-10	4.E+01
Cu-64	6.93E-01	4.39E-08	1.E+03
Zn-65	2.80E+02	1.78E-05	1.E+01
Rb-89	2.01E-02	1.27E-09	7.E+03
Sr-89	1.48E+02	9.38E-06	7.E+00
Sr-90	7.65E-01	4.85E-08	2.E-01
Y-90	3.27E-02	2.07E-09	3.E+01
Sr-91	6.72E-01	4.26E-08	2.E+02
Sr-92	4.63E-01	2.93E-08	3.E+02
Y-91	1.74E-01	1.10E-08	7.E+00
Y-92	3.68E-01	2.33E-08	4.E+02
Y-93	7.23E-01	4.58E-08	1.E+02
Zr-95	4.49E+01	2.85E-06	1.E+01
Nb-95	2.44E+02	1.55E-05	7.E+01
Mo-99	1.66E+03	1.05E-04	7.E+01
Tc-99m	2.23E-01	1.41E-08	7.E+03
Ru-103	1.04E+02	6.58E-06	3.E+01
Rh-103m	8.24E-02	5.22E-09	7.E+04
Ru-106	1.35E-02	8.56E-10	7.E-01
Rh-106	1.35E-02	8.56E-10	4.E+01
Ag-110m	5.86E-02	3.71E-09	4.E+00
Sb-124	5.37E+00	3.40E-07	1.E+01
Te-129m	1.63E-01	1.03E-08	1.E+01
Te-131m	5.50E-02	3.49E-09	4.E+01
Te-132	1.41E-02	8.91E-10	3.E+01
Cs-134	1.78E+02	1.13E-05	7.E+00
Cs-136	1.47E+01	9.31E-07	3.E+01
Cs-137	2.69E+02	1.70E-05	7.E+00
Cs-138	8.50E-02	5.39E-09	3.E+03

Table 12.2-17

Comparison of Airborne Concentrations with 10 CFR 20 Concentrations

	Airborne Release	Concentration	10 CFR 20
Nuclide	**MBq/yr**	**Bq/m^3**	**Bq/m^3**
Ba-140	7.82E+02	4.96E-05	7.E+01
La-140	1.29E+00	8.19E-08	7.E+01
Ce-141	2.66E+02	1.69E-05	3.E+01
Ce-144	1.35E-02	8.53E-10	7.E-01
Pr-144	1.35E-02	8.53E-10	7.E+00
W-187	1.29E-01	8.21E-09	4.E+02
Np-239	8.28E+00	5.25E-07	1.E+02

Table 12.2-18a

Airborne Offsite Dose Calculation Bases

Meteorology χ/Q	Table 12.2-15
Meteorology D/Q	Table 12.2-15
Meteorology Boundary	Table 12.2-15
Airborne Release Source Term	Table 12.2-16
Calculation Methodology	Regulatory Guide 1.109
Computer Code Utilized	GASPAR II (NUREG/CR-4653)
Individual Consumption Rates	Table E-5 of Reg. Guide 1.109
Misc. Calculation Inputs (other than Reg. Guide 1.109 default values):	
Midpoint of plant operating life	30 years
Fraction of year that leafy vegetables are grown	0.75
Fraction of year that animals graze on pasture	0.5
Fraction of daily feed that is pasture grass when the animal grazes on pasture	0.75
Animal milk considered for milk pathway	Cow
Annual Average Doses from Airborne Releases	Table 12.2-18b

Table 12.2-18b

ESBWR Annual Average Doses from Airborne Releases

	Annual Dose (mSv/year)							
PATHWAY	**T. BODY**	**GI-TRACT**	**BONE**	**LIVER**	**KIDNEY**	**THYROID**	**LUNG**	**SKIN**
PLUME	7.73E-03	7.73E-03	7.73E-03	7.73E-03	7.73E-03	7.73E-03	7.86E-03	2.08E-02
GROUND	5.00E-04	5.00E-04	5.00E-04	5.00E-04	5.00E-04	5.00E-04	5.00E-04	5.87E-04
VEGET								
ADULT	1.29E-03	1.25E-03	5.56E-03	1.37E-03	1.32E-03	2.75E-02	1.14E-03	1.12E-03
TEEN	2.00E-03	1.96E-03	9.12E-03	2.19E-03	2.11E-03	3.60E-02	1.84E-03	1.81E-03
CHILD	4.55E-03	4.43E-03	2.21E-02	4.96E-03	4.80E-03	6.89E-02	4.38E-03	4.33E-03
MEAT								
ADULT	4.29E-04	4.71E-04	2.05E-03	4.35E-04	4.27E-04	8.96E-04	4.17E-04	4.16E-04
TEEN	3.56E-04	3.80E-04	1.73E-03	3.65E-04	3.58E-04	6.97E-04	3.51E-04	3.49E-04
CHILD	6.61E-04	6.70E-04	3.25E-03	6.72E-04	6.63E-04	1.18E-03	6.54E-04	6.53E-04
MILK								
ADULT	5.67E-04	4.99E-04	2.32E-03	6.24E-04	5.85E-04	1.40E-02	4.76E-04	4.66E-04
TEEN	9.70E-04	8.89E-04	4.27E-03	1.12E-03	1.06E-03	2.23E-02	8.67E-04	8.47E-04
CHILD	2.21E-03	2.09E-03	1.05E-02	2.52E-03	2.41E-03	4.46E-02	2.09E-03	2.06E-03
INFANT	4.51E-03	4.38E-03	2.04E-02	5.22E-03	4.85E-03	1.08E-01	4.34E-03	4.28E-03
INHALE								
ADULT	5.32E-05	6.40E-05	2.43E-05	7.62E-05	9.64E-05	5.56E-03	1.06E-04	3.45E-05
TEEN	5.74E-05	6.88E-05	3.39E-05	9.17E-05	1.19E-04	7.14E-03	1.44E-04	3.48E-05
CHILD	5.43E-05	5.24E-05	4.56E-05	8.52E-05	1.09E-04	8.57E-03	1.21E-04	3.08E-05
INFANT	3.37E-05	2.87E-05	3.45E-05	6.58E-05	6.81E-05	7.82E-03	8.68E-05	1.77E-05
TOTAL	**Annual Dose (mSv/year)**							
ADULT	1.06E-02	1.05E-02	1.82E-02	1.07E-02	1.07E-02	5.62E-02	1.05E-02	2.34E-02
TEEN	1.16E-02	1.15E-02	2.34E-02	1.20E-02	1.19E-02	7.44E-02	1.16E-02	2.44E-02
CHILD	1.57E-02	1.55E-02	4.41E-02	1.65E-02	1.62E-02	1.31E-01	1.56E-02	2.85E-02
INFANT	1.28E-02	1.26E-02	2.87E-02	1.35E-02	1.31E-02	1.24E-01	1.28E-02	2.57E-02

Annual beta air dose = 1.32E-02 mGy
Annual gamma air dose = 1.16E-02 mGy

Table 12.2-19a

Average Annual Liquid Release Calculation Parameters

BWR-GALE Card Number	Parameter	Data
1	Name of reactor	GE-ESBWR
	Type	BWR
2	Thermal power level	4500 MWth
3	Total steam flow	1.93E+07 lb/hr
4	Mass of water in reactor vessel	6.74E+05 lb
5	Clean-up demineralizer flow	1.93E+05 lb/hr
6	Condensate demineralizer regenerative time	0 days
7	Cooper tubing for condenser	0 (no)
8	Fraction feed water through condensate demineralizer	1.0
9	High purity waste input	17,173 gal/day
	High purity: Fraction of reactor coolant activity	0.268
10	High purity: Decontamination factor for iodine	1000
	High purity: Decontamination factor for Cs and Rb	100
	High purity: Decontamination factor for others	1000
11	High purity: Collection time	2.958 days
	High purity: Process and discharge time	0.233 days
	High purity: Fraction discharged	0.01
12	Low purity waste input	6,750 gal/day
	Low purity: Fraction of reactor coolant activity	0.001
13	Low purity: Decontamination factor for iodine	10,000
	Low purity: Decontamination factor for Cs and Rb	200
	Low purity: Decontamination factor for others	10,000
14	Low purity: Collection time	9.455 days
	Low purity: Process and discharge time	0.289 days
	Low purity: Fraction discharged	0.1
15	Chemical waste input	793 gal/day
	Chemical: Fraction of reactor coolant activity	0.02
16	Chemical: Decontamination factor for iodine	10,000
	Chemical: Decontamination factor for Cs and Rb	200
	Chemical: Decontamination factor for others	10,000
17	Chemical: Collection time	1.255 days
	Chemical: Process and discharge time	0.289 days
	Chemical: Fraction discharged	0.1
18	Detergent waste input	1,057 gal/day

Table 12.2-19a

Average Annual Liquid Release Calculation Parameters

BWR-GALE Card Number	Parameter	Data
19	Detergent: Decontamination factor for iodine	1
	Detergent: Decontamination factor for Cs and Rb	1
	Detergent: Decontamination factor for others	1
20	Detergent: Collection time	3.0 days
	Detergent: Process and discharge time	0.25 days
	Detergent: Fraction discharged	0.100
21 to 33	Data only of gaseous releases	0
34	Detergent waste decontamination factor - Laundry	0

Table 12.2-19b
Average Annual Liquid Releases

	Annual Release	Concentration	10CFR20 MPC
Nuclide	MBq/yr	Bq/ml	Bq/ml
I-131	1.55E+02	1.48E-05	3.70E-02
I-132	3.03E+01	2.88E-06	3.70E+00
I-133	7.77E+02	7.39E-05	2.59E-01
I-134	1.48E+00	1.41E-07	1.48E+01
I-135	2.00E+02	1.90E-05	1.11E+00
H-3	5.18E+05	4.92E-02	3.70E+01
Na-24	1.89E+02	1.79E-05	1.85E+00
P-32	1.55E+01	1.48E-06	3.33E-01
Cr-51	4.81E+02	4.57E-05	1.85E+01
Mn-54	5.92E+00	5.63E-07	1.11E+00
Mn-56	4.81E+01	4.57E-06	2.59E+00
Fe-55	8.51E+01	8.09E-06	3.70E+00
Fe-59	2.59E+00	2.46E-07	3.70E-01
Co-58	1.63E+01	1.55E-06	7.40E-01
Co-60	3.33E+01	3.17E-06	1.11E-01
Cu-64	4.81E+02	4.57E-05	7.40E+00
Zn-65	1.67E+01	1.58E-06	1.85E-01
Zn-69m	3.40E+01	3.24E-06	2.22E+00
Br-83	3.33E+00	3.17E-07	3.33E+01
Sr-89	8.14E+00	7.74E-07	2.96E-01
Sr-90	7.40E-01	7.03E-08	1.85E-02
Sr-91	4.44E+01	4.22E-06	7.40E-01
Y-91	5.18E+00	4.92E-07	2.96E-01
Sr-92	1.07E+01	1.02E-06	1.48E+00
Y-92	4.07E+01	3.87E-06	1.48E+00
Y-93	4.44E+01	4.22E-06	7.40E-01
Zr-95	7.40E-01	7.03E-08	7.40E-01
Nb-95	7.40E-01	7.03E-08	1.11E+00
Mo-99	1.11E+02	1.06E-05	7.40E-01
Tc-99m	2.04E+02	1.93E-05	3.70E+01
Ru-103	1.48E+00	1.41E-07	1.11E+00
Ru-105	6.29E+00	5.98E-07	2.59E+00
Te-129m	3.33E+00	3.17E-07	2.59E-01
Te-131m	3.70E+00	3.52E-07	2.96E-01

Table 12.2-19b

Average Annual Liquid Releases

	Annual Release	Concentration	10CFR20 MPC
Nuclide	MBq/yr	Bq/ml	Bq/ml
Te-132	7.40E-01	7.03E-08	3.33E-01
Cs-134	2.52E+01	2.39E-06	3.33E-02
Cs-136	1.52E+01	1.44E-06	2.22E-01
Cs-137	6.66E+01	6.33E-06	3.70E-02
Ba-139	1.48E+00	1.41E-07	7.40E+00
Ba-140	3.03E+01	2.88E-06	2.96E-01
Ce-141	2.59E+00	2.46E-07	1.11E+00
La-142	1.11E+00	1.06E-07	3.70E+00
Ce-143	1.11E+00	1.06E-07	7.40E-01
Pr-143	3.33E+00	3.17E-07	7.40E-01
W-187	8.88E+00	8.44E-07	1.11E+00
Np-239	4.07E+02	3.87E-05	7.40E-01

Table 12.2-20a

Liquid Pathway Offsite Dose Calculation Bases

Calculation Methodology	Regulatory Guide 1.109
Computer Code Utilized	LADTAP II (NUREG/CR-4013)
Individual Consumption/Exposure Rates	Table E-5 of Reg. Guide 1.109
Site Water Type	Freshwater
Discharge Canal Flow Rate	2.0E+04 liters/min
Shore-Width Factor	0.2
Dilution Factor	10
Transit times from discharge to the receiving water body to exposure location	- All pathways except drinking water: instantaneous - Drinking water: 12 hours - Irrigated foods: instantaneous
Irrigation rate	0.001 m^3/m^2-day
Fraction of year that leafy vegetables are grown	0.75
Fraction of year that animals graze on pasture	0.5
Fraction of daily feed that is pasture grass when the animal grazes on pasture	0.75
Animal milk considered for milk pathway	Cow
Liquid Pathway Offsite Annual Doses	Table 12.2-20b

Table 12.2-20b
Liquid Pathway Dose Results in mSv/year

	Annual Doses (mSv/yr)							
PATHWAY	**SKIN**	**BONE**	**LIVER**	**T. BODY**	**THYROID**	**KIDNEY**	**LUNG**	**GI-LLI**
Drinking								
Adult	0.00E+00	3.93E-05	8.50E-05	7.64E-05	8.20E-04	7.36E-05	6.09E-05	9.06E-05
Teenager	0.00E+00	3.66E-05	6.65E-05	5.18E-05	7.08E-04	5.57E-05	4.39E-05	6.58E-05
Child	0.00E+00	1.03E-04	1.31E-04	9.35E-05	1.78E-03	1.08E-04	8.43E-05	1.05E-04
Infant	0.00E+00	1.09E-04	1.46E-04	9.02E-05	2.75E-03	1.09E-04	8.36E-05	9.58E-05
Fish								
Adult	0.00E+00	1.62E-02	2.22E-03	1.48E-03	3.31E-04	4.46E-04	1.37E-04	1.87E-03
Teenager	0.00E+00	1.76E-02	2.34E-03	1.17E-03	3.16E-04	4.56E-04	1.63E-04	1.52E-03
Child	0.00E+00	2.27E-02	2.15E-03	1.05E-03	3.46E-04	3.87E-04	1.29E-04	6.46E-04
Shoreline								
Adult	1.75E-06	1.49E-06	1.49E-06	1.49E-06	1.49E-06	1.49E-06	1.49E-06	1.49E-06
Teenager	9.77E-06	8.35E-06	8.35E-06	8.35E-06	8.35E-06	8.35E-06	8.35E-06	8.35E-06
Child	2.04E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06
Irrigated Foods:								
Vegetables								
Adult		5.77E-05	8.14E-05	7.01E-05	1.53E-04	5.56E-05	4.58E-05	5.35E-05
Teenager		9.12E-05	1.16E-04	7.78E-05	2.16E-04	7.38E-05	5.91E-05	6.48E-05
Child		2.08E-04	1.91E-04	1.05E-04	4.08E-04	1.18E-04	9.31E-05	9.04E-05
Leafy Vegetables								
Adult		7.69E-06	1.02E-05	8.77E-06	5.03E-05	7.14E-06	5.66E-06	7.44E-06
Teenager		6.63E-06	7.95E-06	5.30E-06	3.99E-05	5.19E-06	3.95E-06	4.93E-06
Child		1.14E-05	9.83E-06	5.41E-06	5.89E-05	6.23E-06	4.67E-06	4.91E-06
Milk								
Adult		1.58E-05	2.77E-05	2.24E-05	1.17E-04	1.76E-05	1.26E-05	1.35E-05
Teenager		2.84E-05	4.34E-05	2.61E-05	1.83E-04	2.57E-05	1.76E-05	1.75E-05
Child		6.81E-05	7.17E-05	3.39E-05	3.63E-04	4.12E-05	2.78E-05	2.52E-05
Meat								
Adult		3.10E-06	6.18E-06	5.35E-06	7.11E-06	4.87E-06	4.16E-06	5.97E-06
Teenager		2.54E-06	4.16E-06	3.08E-06	4.66E-06	3.11E-06	2.58E-06	3.51E-06
Child		4.70E-06	5.20E-06	3.53E-06	6.36E-06	3.79E-06	3.13E-06	3.45E-06
Total								
Adult	**1.75E-06**	**1.63E-02**	**2.43E-03**	**1.66E-03**	**1.48E-03**	**6.06E-04**	**2.68E-04**	**2.04E-03**
Teenager	**9.77E-06**	**1.78E-02**	**2.59E-03**	**1.34E-03**	**1.48E-03**	**6.28E-04**	**2.98E-04**	**1.68E-03**
Child	**2.04E-06**	**2.31E-02**	**2.56E-03**	**1.29E-03**	**2.96E-03**	**6.66E-04**	**3.44E-04**	**8.77E-04**
Infant		**1.09E-04**	**1.46E-04**	**9.02E-05**	**2.75E-03**	**1.09E-04**	**8.36E-05**	**9.58E-05**

Table 12.2-21
N-16 Skyshine Annual Dose

Distance from Site (m)	Annual Dose (mrem/yr)
800	5.93E-04
1000	1.66E-04

Table 12.2-22

Radiation Sources Parameters

Component	Room	Assumed Shielding Source						
		Source Approx Geometry Rt. Cylinder (r, l)		Source Characteristics				Quantity
		Length (m)	Radius (m)	Type	Material	Density (g/cm^3)	Equipment Self-Shielding	
RWCU/SDC (Reactor Building)								
Non regenerative Heat Exchanger Tube side	1151/1250 1161/1260	7.00	0.16	Homogeneous	Water	0.967	Steel 2cm thick	Three
Regenerative Heat Exchanger Tube side Shell side	1151/1250 1161/1260	7.00 7.00	0.16 0.25	Homogeneous Homogeneous	Water Water	0.836 0.990	Steel 2cm thick	Two
Demineralizer	1251/52/61/62	4.12	0.48	Homogeneous	Resins	0.69	Steel 1cm thick	Four
FAPCS (Fuel Building)								
Heat Exchanger	2150/2160	0.96	0.30	Homogeneous	Water	1.00	Steel 2cm thick	Two
Filter / Demineralizer	2251/2261	2.06	1.12	Homogeneous	Resins	0.69	Steel 1cm thick	Two
Backwash Receiving Tank	2102	1.00	0.56	Homogeneous	Water	1.00	Steel 1cm thick	One
OFF-GAS System (Turbine Building)								
Steam Jet Air Ejectors	4206/4207			Homogeneous	Offgas	5.95E-05	Steel 1cm thick	Two
Preheater/Recombiner/Condenser	4381/4382	10.45m^3		Homogeneous	Offgas	6.5E-04	Steel 1cm thick	Two
Cooler Condenser	4381/4382	0.12 m^3		Homogeneous	Offgas	1.04E-03	Steel 1cm thick	Two
Dryer		5.81 m^3		Homogeneous	Offgas	1.02E-03	Steel 1cm thick	Two
Guard Bed	4108	1.4	2.1	Homogeneous	Offgas	1.02E-03	Steel 1cm thick	Two
Delay Bed	4108	7.5	1.5	Homogeneous	Offgas	1.02E-03	Steel 1cm thick	Eight
CPS (Turbine Building)								
Condensate Demineralizer	42F1A to F1H	0.92	1.75	Homogeneous	Resins	0.69	Steel 2cm thick	Eight
Turbine Condenser (Turb Building)								
Main Condenser Shell Well	4186	1284 m^3 2136 m^3		Homogeneous Homogeneous	Water Water	7.21E-04 1	Steel 1cm thick Steel 1cm thick	Three (Bodies)
LWMS (Radwaste Building)								
Drain Collection Tank	6103/4/5	140m^3		Homogeneous	Water	1	Steel 1cm thick	Three
Floor Drain Collection Tank	6150/6160	130m^3		Homogeneous	Water	1	Steel 1cm thick	Two
Chemical Drain Collection Tank	6201	4m^3		Homogeneous	Water	1	Steel 1cm thick	One
Detergent Drain Collection Tank	6184	15m^3		Homogeneous	Water	1	Steel 1cm thick	Two

Table 12.2-22

Radiation Sources Parameters

Component	Room	Assumed Shielding Source						
		Source Approx Geometry Rt. Cylinder (r, l)		Source Characteristics				Quantity
		Length (m)	Radius (m)	Type	Material	Density (g/cm^3)	Equipment Self-Shielding	
Drain Sample Tank	6172	140m^3		Homogeneous	Water	1	Steel 1cm thick	Two
Floor Drain Sample Tank	6171	130m^3		Homogeneous	Water	1	Steel 1cm thick	Two
Detergent Drain Sample Tank	6282	15m^3		Homogeneous	Water	1	Steel 1cm thick	Two
SWMS (Radwaste Building)								
High Activity Resin Holdup Tank	6108	3.26	2.00	Homogeneous	Resins	0.69	Steel 1cm thick	One
Low Activity Resin Holdup Tank	6107	0.48	2.00	Homogeneous	Water	0.69	Steel 1cm thick	One
Low Activity Sludge Phase Separator	6151/6161	0.48	2.00	Homogeneous	Water	1.00	Steel 1cm thick	Two
Condensate Resin Holdup Tank	6106	2.70	2.00	Homogeneous	Resins	0.69	Steel 1cm thick	One
Concentrate Waste Tank	6109	3.98	2.00	Homogeneous	Water	1.03	Steel 1cm thick	One

Table 12.2-23a

Parameters and Assumptions used for Calculating Inside the Building Airborne Radioactivity Concentrations

Parameter/Assumption	Value
Reactor Building outside Containment	
Source Term	Radioisotopes in reactor water and steam. See Section 11.1. Liquid Phase
Leakage Flowrate	3.9E-04 kg/s
Contaminated Volume	1781.5 m^3
Flashing Fraction	0.4
Normal HVAC flowrate	12.6 m^3/s
Fuel Building	
Source Term	Radioisotopes in spent fuel pool: 1% of radioisotopes in reactor water (see section 11.1), except H-3 (100% of section 11.1 value
Leakage Flowrate	9.4E-02 kg/s
Contaminated Volume	12897 m^3
Flashing Fraction	0.4
Normal HVAC flowrate	14.2 m^3/s
Turbine Building	
Source Term	Radioisotopes in reactor water and steam. See Section 11.1. Steam phase and liquid phase
Leakage Flowrate	0.12 kg/s
Contaminated Volume	93565 m^3
Flashing Fraction	0.4
Carry-over ratio	1 for noble gases 0.02 for iodines 0.001 for other isotopes
Normal HVAC flowrate	47.5 m^3/s
Radwaste Building	
Source Term	Radioisotopes in reactor water and steam See Section 11.1. Liquid Phase
Leakage Flowrate	1.5E-04 kg/s
Contaminated Volume	10447 m^3
Flashing Fraction	0.4
Normal HVAC flowrate	5E-04 /s

Table 12.2-23b
Reactor Building Outside Containment Airborne Radioactivity Concentrations During Normal Operation

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
I-131	4.8E+00	7.4E+02
I-132	4.5E+01	1.1E+05
I-133	3.3E+01	3.7E+03
I-134	8.2E+01	7.4E+05
I-135	4.7E+01	2.6E+04
Rb-89	7.7E+00	2.2E+06
Cs-134	5.6E-02	1.5E+03
Cs-136	3.7E-02	1.1E+04
Cs-137	1.5E-01	2.2E+03
Cs-138	1.6E+01	7.4E+05
Ba-137m	9.1E-02	3.7E+03
H-3	4.6E+00	7.4E+05
Na-24	4.0E+00	7.4E+04
P-32	8.2E-02	7.4E+03
Cr-51	6.2E+00	3.0E+05
Mn-54	7.2E-02	1.1E+04
Mn-56	4.7E+01	2.2E+05
Fe-55	2.1E+00	3.0E+04
Fe-59	6.2E-02	3.7E+03
Co-58	2.1E-01	1.1E+04
Co-60	4.1E-01	3.7E+02
Ni-63	2.1E-03	1.1E+04
Cu-64	5.9E+00	3.3E+05
Zn-65	2.1E+00	3.7E+03
Sr-89	2.1E-01	2.2E+03
Sr-90	1.5E-02	7.4E+01
Y-90	1.5E-02	1.1E+04
Sr-91	7.9E+00	3.7E+04
Sr-92	1.8E+01	1.1E+05

Table 12.2-23b
Reactor Building Outside Containment Airborne Radioactivity Concentrations During Normal Operation

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
Y-91	8.2E-02	1.9E+03
Y-92	1.1E+01	1.1E+05
Y-93	7.9E+00	3.7E+04
Zr-95	1.6E-02	1.9E+03
Nb-95	1.6E-02	1.9E+04
Mo-99	4.1E+00	2.2E+04
Tc-99m	4.1E+00	2.2E+06
Ru-103	4.1E-02	1.1E+04
Rh-103m	4.0E-02	1.9E+07
Ru-106	6.2E-03	1.9E+02
Rh-106	1.4E-03	3.7E+03
Ag-110m	2.1E-03	1.5E+03
Te-129m	8.2E-02	3.7E+03
Te-131m	2.0E-01	7.4E+03
Te-132	2.0E-02	3.3E+03
Ba-140	8.2E-01	2.2E+04
La-140	8.2E-01	1.9E+04
Ce-141	6.2E-02	7.4E+03
Ce-144	6.2E-03	2.2E+02
Pr-144	5.7E-03	1.9E+06
W-187	6.1E-01	1.5E+05
Np-239	1.6E+01	3.3E+04

Table 12.2-23c
Spent Fuel Pool and Equipment Areas Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
I-131	1.0E+01	7.4E+02
I-132	9.0E+01	1.1E+05
I-133	7.0E+01	3.7E+03
I-134	1.5E+02	7.4E+05
I-135	9.7E+01	2.6E+04
Rb-89	1.1E+01	2.2E+06
Cs-134	1.2E-01	1.5E+03
Cs-136	7.9E-02	1.1E+04
Cs-137	3.2E-01	2.2E+03
Cs-138	2.8E+01	7.4E+05
Ba-137m	6.2E-02	3.7E+03
H-3	9.7E+02	7.4E+05
Na-24	8.3E+00	7.4E+04
P-32	1.7E-01	7.4E+03
Cr-51	1.3E+01	3.0E+05
Mn-54	1.5E-01	1.1E+04
Mn-56	9.4E+01	2.2E+05
Fe-55	4.5E+00	3.0E+04
Fe-59	1.3E-01	3.7E+03
Co-58	4.5E-01	1.1E+04
Co-60	8.7E-01	3.7E+02
Ni-63	4.5E-03	1.1E+04
Cu-64	1.2E+01	3.3E+05
Zn-65	4.5E+00	3.7E+03
Sr-89	4.5E-01	2.2E+03
Sr-90	3.2E-02	7.4E+01
Y-90	3.1E-02	1.1E+04
Sr-91	1.7E+01	3.7E+04
Sr-92	3.7E+01	1.1E+05

Table 12.2-23c

Spent Fuel Pool and Equipment Areas Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
Y-91	1.7E-01	1.9E+03
Y-92	2.3E+01	1.1E+05
Y-93	1.7E+01	3.7E+04
Zr-95	3.4E-02	1.9E+03
Nb-95	3.4E-02	1.9E+04
Mo-99	8.7E+00	2.2E+04
Tc-99m	8.4E+00	2.2E+06
Ru-103	8.7E-02	1.1E+04
Rh-103m	7.3E-02	1.9E+07
Ru-106	1.3E-02	1.9E+02
Rh-106	6.0E-04	3.7E+03
Ag-110m	4.5E-03	1.5E+03
Te-129m	1.7E-01	3.7E+03
Te-131m	4.2E-01	7.4E+03
Te-132	4.2E-02	3.3E+03
Ba-140	1.7E+00	2.2E+04
La-140	1.7E+00	1.9E+04
Ce-141	1.3E-01	7.4E+03
Ce-144	1.3E-02	2.2E+02
Pr-144	8.2E-03	1.9E+06
W-187	1.3E+00	1.5E+05
Np-239	3.4E+01	3.3E+04

Table 12.2-23d

Turbine Building Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
Ar-41	1.7E+03	1.1E+05
Kr-83m	1.1E+02	3.7E+08
Kr-85m	2.1E+02	7.4E+05
Kr-85	9.1E-01	3.7E+06
Kr-87	5.9E+02	1.9E+05
Kr-88	6.7E+02	7.4E+04
Kr-89	5.9E+02	3.7E+03
Xe-131m	7.6E-01	1.5E+07
Xe-133m	1.1E+01	3.7E+06
Xe-133	3.3E+02	3.7E+06
Xe-135m	4.1E+02	3.3E+05
Xe-135	8.5E+02	3.7E+05
Xe-137	8.8E+02	3.7E+03
Xe-138	1.4E+03	1.5E+05
I-131	7.9E+00	7.4E+02
I-132	6.4E+01	1.1E+05
I-133	5.4E+01	3.7E+03
I-134	9.6E+01	7.4E+05
I-135	7.3E+01	2.6E+04
Rb-89	2.8E-01	2.2E+06
Cs-134	4.6E-03	1.5E+03
Cs-136	3.0E-03	1.1E+04
Cs-137	1.2E-02	2.2E+03
Cs-138	8.3E-01	7.4E+05
Ba-137m	1.2E-03	3.7E+03
H-3	3.8E+02	7.4E+05
Na-24	3.2E-01	7.4E+04
P-32	6.7E-03	7.4E+03
Cr-51	5.1E-01	3.0E+05

Table 12.2-23d

Turbine Building Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
Mn-54	5.9E-03	1.1E+04
Mn-56	3.4E+00	2.2E+05
Fe-55	1.7E-01	3.0E+04
Fe-59	5.1E-03	3.7E+03
Co-58	1.7E-02	1.1E+04
Co-60	3.3E-02	3.7E+02
Ni-63	1.7E-04	1.1E+04
Cu-64	4.7E-01	3.3E+05
Zn-65	1.7E-01	3.7E+03
Sr-89	1.7E-02	2.2E+03
Sr-90	1.2E-03	7.4E+01
Y-90	1.2E-03	1.1E+04
Sr-91	6.2E-01	3.7E+04
Sr-92	1.3E+00	1.1E+05
Y-91	6.7E-03	1.9E+03
Y-92	8.5E-01	1.1E+05
Y-93	6.3E-01	3.7E+04
Zr-95	1.3E-03	1.9E+03
Nb-95	1.3E-03	1.9E+04
Mo-99	3.3E-01	2.2E+04
Tc-99m	3.1E-01	2.2E+06
Ru-103	3.3E-03	1.1E+04
Rh-103m	2.4E-03	1.9E+07
Ru-106	5.1E-04	1.9E+02
Rh-106	1.1E-05	3.7E+03
Ag-110m	1.7E-04	1.5E+03
Te-129m	6.7E-03	3.7E+03
Te-131m	1.6E-02	7.4E+03
Te-132	1.6E-03	3.3E+03
Ba-140	6.7E-02	2.2E+04
La-140	6.6E-02	1.9E+04
Ce-141	5.1E-03	7.4E+03
Ce-144	5.1E-04	2.2E+02
Pr-144	2.2E-04	1.9E+06

Table 12.2-23d

Turbine Building Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
W-187	4.9E-02	1.5E+05
Np-239	1.3E+00	3.3E+04

Table 12.2-23e

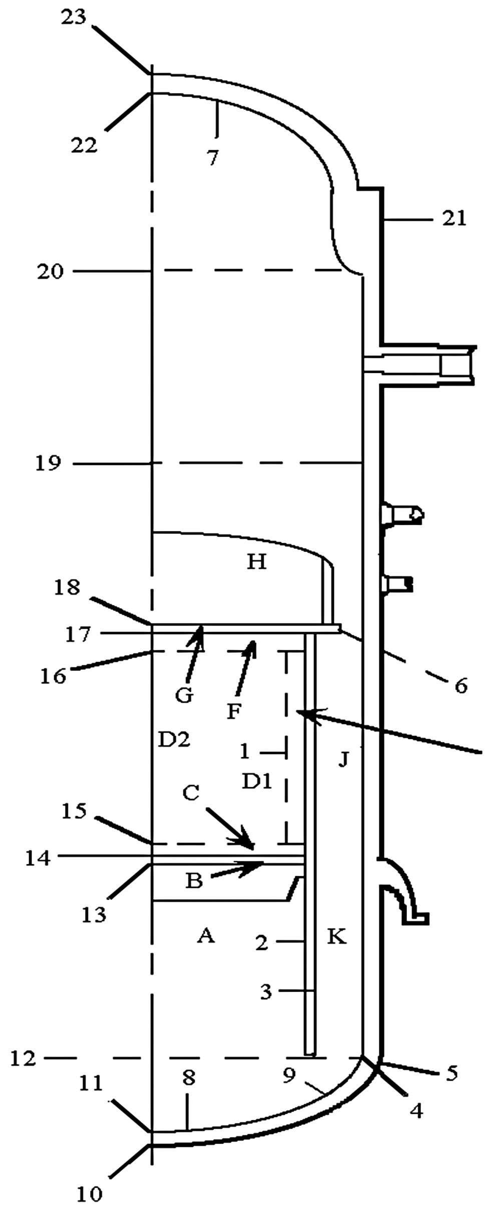
Radwaste Building Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
I-131	4.3E+00	7.4E+02
I-132	3.5E+01	1.1E+05
I-133	3.0E+01	3.7E+03
I-134	5.3E+01	7.4E+05
I-135	4.0E+01	2.6E+04
Rb-89	3.1E+00	2.2E+06
Cs-134	5.0E-02	1.5E+03
Cs-136	3.3E-02	1.1E+04
Cs-137	1.3E-01	2.2E+03
Cs-138	9.1E+00	7.4E+05
Ba-137m	1.3E-02	3.7E+03
H-3	4.1E+00	7.4E+05
Na-24	3.5E+00	7.4E+04
P-32	7.4E-02	7.4E+03
Cr-51	5.6E+00	3.0E+05
Mn-54	6.5E-02	1.1E+04
Mn-56	3.7E+01	2.2E+05
Fe-55	1.9E+00	3.0E+04
Fe-59	5.6E-02	3.7E+03
Co-58	1.9E-01	1.1E+04
Co-60	3.7E-01	3.7E+02
Ni-63	1.9E-03	1.1E+04
Cu-64	5.2E+00	3.3E+05
Zn-65	1.9E+00	3.7E+03
Sr-89	1.9E-01	2.2E+03
Sr-90	1.3E-02	7.4E+01
Y-90	1.3E-02	1.1E+04
Sr-91	6.9E+00	3.7E+04
Sr-92	1.5E+01	1.1E+05
Y-91	7.4E-02	1.9E+03

Table 12.2-23e

Radwaste Building Airborne Radioactivity Concentrations

Nuclide	Concentration Bq/m^3	10 CFR 20 Bq/m^3
Y-92	9.4E+00	1.1E+05
Y-93	6.9E+00	3.7E+04
Zr-95	1.5E-02	1.9E+03
Nb-95	1.5E-02	1.9E+04
Mo-99	3.7E+00	2.2E+04
Tc-99m	3.5E+00	2.2E+06
Ru-103	3.7E-02	1.1E+04
Rh-103m	2.6E-02	1.9E+07
Ru-106	5.6E-03	1.9E+02
Rh-106	1.2E-04	3.7E+03
Ag-110m	1.9E-03	1.5E+03
Te-129m	7.4E-02	3.7E+03
Te-131m	1.8E-01	7.4E+03
Te-132	1.8E-02	3.3E+03
Ba-140	7.4E-01	2.2E+04
La-140	7.3E-01	1.9E+04
Ce-141	5.6E-02	7.4E+03
Ce-144	5.6E-03	2.2E+02
Pr-144	2.4E-03	1.9E+06
W-187	5.4E-01	1.5E+05
Np-239	1.4E+01	3.3E+04



Note: See Table 12.2-1 for component designations.

Figure 12.2-1. Radiation Source Model

12.3 RADIATION PROTECTION

12.3.1 Facility Design Features

The ESBWR Standard Plant is designed in accordance with Regulatory Guide 8.8 (Reference 12.3-11), i.e., to keep radiation exposures to plant personnel as low as reasonably achievable (ALARA). This section describes the component and system designs in addition to the equipment layout employed to maintain radiation exposures ALARA. Consideration of individual systems is provided to illustrate the application of these principles. The details in this section serve as input to the final design configuration and serve to determine the adequacy of the design with respect to radiation protection. Compliance to NUREG-0800 radiation protection acceptance criteria (Reference 12.3-15) for the specific systems, structures and components (SSCs) are further described in the applicable sections of the DCD.

Material selection for primary coolant piping, tubing, vessel internal surfaces, and other components in contact with the primary coolant is discussed in the following pages.

Carbon steel is used in a large portion of the system piping and equipment outside of the nuclear steam supply system. Carbon steel is typically low in nickel content and contains a very small amount of cobalt impurity.

Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is in the 9 to 10.5% range and is controlled in accordance with applicable ASME material specifications. Cobalt content is controlled to less than 0.05% in the XM-19 alloy used in the control rod drives.

A previous review of materials certifications indicated average cobalt content of only 0.15% in austenitic stainless steels.

Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750, which have high nickel content, are used in some reactor vessel internal components. These materials are used in applications for which there are special requirements to be satisfied (such as possessing specific thermal expansion characteristics along with adequate corrosion resistance) and for which no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.

Stellite is used for hard facing of components that must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. An alternative material (Colmonoy) has been used for some hard facings in the core area.

Main condenser tubes and tube sheets are made of stainless steel or titanium alloys to minimize the introduction of foreign material into the reactor system as a result of condenser tube leakage.

12.3.1.1 Equipment Design for Maintaining Exposure ALARA

This subsection describes specific components as well as system design features that aid in maintaining the exposure of plant personnel during system operation and maintenance ALARA. Equipment layouts to provide ALARA exposures of plant personnel are discussed in Subsection 12.3.1.2.

12.3.1.1.1 Pumps

Pumps located in radiation areas are designed to minimize the time required for maintenance. Quick-change cartridge-type seals on pumps, and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping are employed to minimize exposure time during pump maintenance. The configuration of piping about pumps is designed to provide sufficient space for efficient pump maintenance. Toward this end, systems that contain pumps generally have the pumps in a separate alcove with piping routed to the back of the alcove into shielded pipe chases. Provisions are made for flushing and in certain cases chemically cleaning pumps prior to maintenance. Pump casing drains provide a means for draining pumps to the sumps prior to disassembly, thus reducing the exposure of personnel and decreasing the potential for contamination. Where two or more pumps conveying highly radioactive fluids are required for operational reasons to be located adjacent to each other, shielding is provided between the pumps to maintain exposure levels ALARA. Pumps adjacent to other highly radioactive equipment are also shielded to reduce the maintenance exposure, for example, in the radwaste system.

Whenever possible, operation of the pumps and associated valves for radioactive systems is accomplished remotely through reach rods or electric controls. Pump control instrumentation is located outside high radiation areas in separate alcoves, and motor- or pneumatic-operated valves and valve extension stems are employed to allow operation from these areas.

12.3.1.1.2 Instrumentation

Instruments are located in low radiation areas such as shielded valve galleries, corridors, or control rooms, whenever possible. Shielded valve galleries provided for this purpose include those for the RWCU/SDC, FAPCS, and radwaste (cleanup phase separator and spent resin tank) systems. Instruments that are required to be located in high radiation areas due to operations requirements are designed such that removal of these instruments to low radiation areas for maintenance is possible. Sensing lines are routed from taps on the primary system in order to avoid placing the transmitters or readout devices in high radiation areas. For example, reactor water level sensing instruments are located outside the drywell.

Liquid service equipment for systems containing radioactive fluids is provided with vent and backflush provisions. Instrument lines, except those for the reactor vessel, are designed with provisions for backflushing and maintaining a clean fill in the sensing lines. The reactor vessel sensing lines may be flushed with condensate following reactor blowdown.

12.3.1.1.3 Heat Exchangers

Heat exchangers are constructed of stainless steel or Cu/Ni tubes to minimize the possibility of failure and reduce maintenance requirements. The heat exchanger design allows for the complete drainage of fluids from the exchanger, avoiding pooling effects that could lead to radioactive crud deposition. Connections are available for condensate or demineralized water flushing of the heat exchangers. For the RWCU/SDC, separate connections are also provided for introducing chemical cleaning solutions for decontaminating the heat exchangers. The fuel pool heat exchanger is downstream of the filter/demineralizer and is therefore not subjected to flows containing significant amounts of fission or activation products.

Instrumentation and valves are remotely operable to the maximum extent possible in the shielded heat exchanger cubicles, to reduce the need for entering these high radiation areas.

12.3.1.1.4 Valves

Valve packing and gasket material are selected on a conservative basis, accounting for environmental conditions such as temperature, pressure, and radiation tolerance requirements to provide a long operating life. Valves have back seats to minimize the leakage through the packing. Straight-through valve configurations were selected where practical, over those that exhibit flow discontinuities or internal crevices to minimize crud trapping. Teflon gaskets are not used.

Wherever possible, valves in systems containing radioactive fluids are separated from those for "clean" services to reduce the radiation exposure from adjacent valves and piping during maintenance.

Pneumatic or mechanically operated valves are employed in high radiation areas, whenever practical, to minimize the need for entering these areas. For certain situations, manually operated valves are required, and in such cases extension valve stems are provided which are operated from a shielded area. Flushing and drain provisions are employed in radioactive systems to reduce exposure to personnel during maintenance.

For areas in which especially high radiation levels are encountered, valves are reduced to the maximum extent possible with the bulk of the valve and piping located in an adjacent valve gallery where the radiation levels are lower.

12.3.1.1.5 Piping

Piping was selected to provide a service life equivalent to the design life of the plant, with consideration given to corrosion allowances and environmental conditions. Piping in radioactive systems such as the RWCU/SDC has butt-welded connections, rather than socket welds, to reduce crud traps. Distinction is made between piping conveying radioactive and non radioactive fluids, and separate routing through shielded pipe chases is provided whenever possible. Piping conveying highly contaminated fluids is usually routed through shielded pipe chases and shielded cubicles. However, when these options are not feasible, the radioactive piping is embedded in concrete walls and floors.

12.3.1.1.6 Lighting

Lighting is designed to provide sufficient illumination in radiation areas to allow quick and efficient surveillance and maintenance operations. To reduce the need for immediate replacement of defective bulbs, multiple lighting fixtures are provided in shielded cubicles. Incandescent lamps are the only type of lamp used within the primary containment, the main steam tunnel, and the refueling level of the reactor building. They require less time for servicing and, hence, the personnel exposure is reduced. Consideration is also given to locating lighting fixtures in easily accessible locations, thus reducing the exposure time for bulb replacement.

12.3.1.1.7 Floor Drains

Floor drains with appropriately sloped floors are provided in shielded cubicles where the potential for spills exist. Those drain lines having a potential for containing highly radioactive fluids are routed through pipe chases, shielded cubicles, or are embedded in concrete walls and floors. Smooth epoxy-type coatings are employed to facilitate decontamination when a spill does occur.

12.3.1.1.8 Ventilation

The Reactor Building Contaminated Area HVAC Subsystem (CONAVS) supplies air to the containment during reactor shutdown for personnel access to the containment area. During normal operation, the preheated outside air travels through the Air Handling Units (AHU) where particulates are removed from the air by low and high efficiency filters; heat is transferred between the air and the hot/chilled water coils; and the conditioned air is distributed to the controlled areas by the supply fan.

The exhaust subsystem consists of redundant exhaust fans connected to common collection and discharge duct systems. During normal operation, the operating fan exhausts air from the controlled areas directly to the atmosphere through the plant vent stack. During purge operation, the operating purge fan exhausts air from the containment area through the purge exhaust filter unit prior to discharge to the plant vent stack.

The Reactor Building Refueling and Pool Area HVAC Subsystem (REPAVS) is a once-through ventilation system that distributes conditioned air to the refueling area of the reactor and spent fuel pool area of the Fuel Building. During normal operation, outside air travels through the AHU's stages where particulates are removed from the air by low and high efficiency filters; heat is transferred between the air and the hot/chilled water coils; and the conditioned air is distributed to the refueling area and spent fuel pool surfaces. Air is ducted to the exhaust fan and exhausted to the outside atmosphere through the plant vent stack. The exhaust system has the manual capability to divert the exhaust for filtration by the purge exhaust filter unit, prior to discharge to the plant vent stack.

12.3.1.2 Plant Design for Maintaining Exposure ALARA

This subsection describes features of equipment layout and design that are employed to maintain personnel exposures ALARA.

12.3.1.2.1 Penetrations

Penetrations through shield walls are avoided whenever possible to reduce the number of streaming paths provided by these penetrations. Whenever penetrations are required through shield walls, however, they are located to minimize the effect on surrounding areas. Penetrations are located so that the radiation source cannot "see" through the penetration. When this is not possible, or to provide an added order of reduction, penetrations are located to exit far above floor level in open corridors or in other relatively inaccessible areas. Penetrations that are offset through a shield wall are frequently employed for electrical penetrations to reduce the streaming of radiation through these penetrations.

Where permitted, the annular region between pipe and penetration sleeves, as well as electrical penetrations, are filled with shielding material to reduce the streaming area presented by these penetrations. The shielding materials used in these applications include lead-loaded silicone foam, with a density comparable to concrete, and boron-loaded refractory-type material for applications requiring neutron as well as gamma shielding. There are certain penetrations where these two approaches are not feasible or are not sufficiently effective. In those cases, a shielded enclosure about the penetration as it exits in the shield wall, with a 90-degree bend of the process pipe as it exits the penetration, is employed.

12.3.1.2.2 Sample Stations

Sample stations in the plant provide for the routine surveillance of reactor water quality. These sample stations are located in low radiation areas to reduce the exposure to operating personnel. Flushing provisions are included using demineralized water, and pipe drains to plant sumps are provided to minimize the possibility of spills. Fume hoods are employed for airborne contamination control. Both working areas and fume hoods are constructed of polished stainless steel to ease decontamination if a spill does occur. Grab spouts are located above the sink to reduce the possibility of contaminating surrounding areas during the sampling process.

12.3.1.2.3 HVAC Systems

Major HVAC equipment (blowers, coolers, and the like) is located in dedicated low radiation areas to minimize exposures to personnel maintaining this equipment ALARA. HVAC ducting is routed outside pipe chases and does not penetrate pipe chase walls, which could compromise the shielding. HVAC ducting penetrations through walls of shielded cubicles are located to minimize the effect of the streaming radiation levels in adjoining areas. Additional HVAC design considerations are addressed in Subsection 12.3.3.

12.3.1.2.4 Piping

Piping containing radioactive fluids is routed through shielded pipe chases, shielded equipment cubicles, or embedded in concrete walls and floors, whenever possible. Where possible, "clean" services, such as compressed air and demineralized water, are not routed through shielded pipe chases. For situations in which radioactive piping must be routed through corridors or other low radiation areas, an analysis is conducted to ensure this routing does not compromise the existing radiation zoning.

In some piping, feed-throughs with short sections, the piping may be embedded in concrete. Optimization by short sections with embedded piping to the extent practicable facilitates the dismantlement of the systems and the decommissioning of the facility, as required by 10 CFR 20.1406.

Radioactive services are routed separately from piping containing nonradioactive fluids, whenever possible, to minimize the exposure to personnel during maintenance. When such routing combinations are required, however, drain provisions are provided to remove the radioactive fluid contained in equipment and piping. In such situations, provisions are made for the valves required for process operation to be controlled remotely, without need for entering the cubicle.

Radwaste piping gallery between the Turbine Building and the Radwaste Building contains only nonsafety-related electrical cables that are separated from the radwaste piping by a 20 cm shield wall. Cable replacement, though infrequent, will be performed during shutdown or when no waste transfer operations are occurring in accordance with plant maintenance and radiation protection program procedures that take into account ALARA. The dose rate from the waste piping is negligible.

"Clean" services and radioactive piping are required at times to be routed together in shielded cubicles. In such situations, provisions are made for the valves required for process operation to be controlled remotely, without need for entering the cubicle.

Penetrations for piping through shield walls are designed to minimize the effect on surrounding areas. Approaches used to accomplish this objective are described in Subsection 12.3.1.2.

Piping configurations are designed to minimize the number of "dead legs" and low points in piping runs to avoid accumulation of radioactive crud and fluids in the line. Drains and flushing provisions are employed whenever feasible to reduce the effect of required "dead legs" and low points. Systems containing radioactive fluids are welded to the most practical extent to reduce leakage through flanged or screwed connections. For highly radioactive systems, butt welds are employed to minimize crud traps. Provisions are also made in radioactive systems for flushing with condensate or chemically cleaning the piping to reduce crud buildup.

12.3.1.2.5 Equipment Layout

Equipment layout is designed to reduce the exposure of personnel required to inspect or maintain equipment. "Clean" pieces of equipment are located separately from those which are sources of radiation whenever possible. For systems that have components that are major sources of radiation, piping and pumps are located in separate cubicles to reduce exposure from these components during maintenance. These major radiation sources are also separately shielded from each other.

12.3.1.2.6 Contamination Control

Contaminated piping systems are welded to the most practical extent to minimize leaks through screwed or flanged fittings. For systems containing highly radioactive fluids, drains are hard piped directly to equipment drain sumps, rather than to allow contaminated fluid to flow across the floor to a floor drain. Certain valves in the main steam line are also provided with leakage drains piped to equipment drain sumps to reduce contamination of the steam tunnel. Pump casing drains are employed on radioactive systems whenever possible to remove fluids from the pump prior to disassembly. In addition, provisions for flushing with condensate, and in especially contaminated systems, for chemically cleaning the equipment prior to maintenance, are provided.

The HVAC System is designed to limit the extent of airborne contamination by providing air-flow patterns from areas of low contamination to more contaminated areas. This, in general, is accomplished by pressurizing the main corridor on each floor with the flow directed outward in each cubicle and then to the pipe chases where the flow is directed to the plant stack. Penetrations through outer walls of the building containing radiation sources are sealed to prevent miscellaneous leaks into the environment. The equipment drain sump vents that contain airborne contaminants from discharges to the sump are piped directly to their respective building HVAC System. Wet transfer of both the steam dryer and separator also reduces the likelihood of contaminants on this equipment being released into the plant atmosphere. In areas where the reduction of airborne contaminants cannot be eliminated efficiently by HVAC Systems, breathing air provisions are provided, for example, for CRD removal under the reactor pressure vessel and in the CRD maintenance room.

Appropriately sloped floor drains are provided in shielded cubicles and other areas where the potential for a spill exists to limit the extent of contamination. Curbs are also provided to limit contamination and simplify washdown operations. A cask decontamination vault is located in the reactor building where the spent fuel cask and other equipment may be cleaned. The CRD maintenance room is used for disassembling control rod drives to reduce the contamination potential.

Consideration is given in the design of the plant for reducing the effort required for decontamination. Epoxy-type wall and floor coverings have been selected which provide smooth surfaces to ease decontamination. Expanded metal-type floor gratings are minimized in favor of smooth surfaces in areas where radioactive spills could occur. Equipment and floor drain sumps are stainless steel lined to reduce crud buildup and to provide surfaces easily decontaminated.

The main equipment washed down in the washdown bays is the spent fuel cask and its transporter. The spent fuel cask is decontaminated in the cask pit (room 21P2). After the spent fuel cask is loaded on the transporter, any remaining potential surface contamination is monitored in the washdown bays. If any contamination is detected, the spent fuel cask and the transporter are washed down in the washdown bays. Other equipment leaving the plant will also be decontaminated inside the plant before loaded onto the transporter, monitored, and washed down if required in the washdown bays before leaving the Fuel Building.

The washdown bays include, among others, the following design features to minimize the spread of contamination:

- Walls or curbs located around areas of potential contaminated fluid leakage;
- Floor surfaces sloped to drains, and sumps sized for cleanup water flow rate;
- Concrete surfaces, including floor surfaces, which have the potential of being flooded or sprayed with radioactive liquid, are protected with a non-porous coating. Epoxy-type wall and floor coverings provide smooth surfaces for ease of decontamination; and
- The decontamination fluid will be processed through the liquid radwaste system as necessary, per plant operating procedures.

The washdown bays will be secured for use as per Regulatory Guide 8.8 guidance.

12.3.1.3 Radiation Zoning

Radiation zones are established in all areas of the plant as a function of both the access requirements of that area and the radiation sources in that area. Operating activities, inspection requirements of equipment, maintenance activities, and abnormal operating conditions are considered in determining the appropriate zoning for a given area. The relationship between radiation zone designations and accessibility requirements is presented in the following tabulation:

Zone Designation	**Dose Rate μSv/hr**	**Access Description**
A	D≤ 6 μSv/h	Uncontrolled, unlimited access

B	6 μSv/h <D≤ 10 μSv/h	Controlled and unlimited access. (No or very low radiation sources are present)
C	10 μSv/h<D ≤ 50 μSv/h	Controlled and limited access (20 hr/wk). (Low radiation sources are present)
D	50 μSv/h <D ≤ 250 μSv/h	Controlled and limited access (4 hr/wk). (Low to moderate radiation sources are present)
E	250 μSv/h <D ≤ 1 mSv/h	Controlled and limited access (1 hr/wk). (Moderate radiation sources are present)
F	1 mSv/h <D ≤ 10 mSv/h	Limited and controlled access with special authorization permit required. (High radiation sources are present)
G	10 mSv/h <D ≤ 100 mSv/h	(Same as zone F above)
H	100mSv/hr<D≤ 1 Sv/h	(Same as zone F above)
I	1 Sv/h <D≤ 5 Sv/h	(Same as zone F above)
J	D > 5 Sv/h	Inaccessible during power and shutdown operations. (Very High radiation sources are present)

The dose rate applicable for a particular zone is based on operating experience and represents design dose rates in a particular zone, and should not be interpreted as the expected dose rates which would apply in all portions of that zone, or for all types of work within that zone, or at all periods of entry into the zone. Large BWR plants have been in operation for three decades, and operating experience with similar design basis numbers shows that only a small fraction of the 10 CFR 20 maximum permissible dose is received in such zones from radiation sources controlled by equipment layout or the structural shielding provided. Therefore, on a practical basis, a radiation zoning approach as described above accomplishes the as low as reasonably achievable objectives for doses as required by 10 CFR 20 Subpart C. The radiation zone maps for this plant with zone designations as described in the preceding tabulations are contained in Figures 12.3-1 through 12.3-22.

Access to areas in the plant is controlled and regulated by the zoning of a given area. Areas with dose rates such that an individual would receive a dose in excess of 1000 μSv (100 mRem) in a period of one hour are locked and posted with “High Radiation Area” signs. Areas in which an individual would receive a dose in excess of 5 Sv (500 Rem) within a period of one hour at 1 meter from a radiation source or 1 meter from any surface that the radiation penetrates are posted with “Very High Radiation Area” signs. Controlled access to these areas is provided by the COL holder (COL 12.3-3-A).

12.3.1.4 Implementation of ALARA

In this subsection, the implementation of design considerations to radioactive systems for maintaining personnel radiation exposures as low as reasonably achievable is described for the RWCU/SDC, Main Steam, FAPCS, and Inclined Fuel Transfer System (IFTS).

12.3.1.4.1 Reactor Water Cleanup / Shutdown Cooling System

This system is designed to operate continuously to reduce reactor water radioactive contamination, as well as perform shutdown cooling. Components for this system are located outside the containment and include demineralizers, regenerative (RHX) and nonregenerative (NRHX) heat exchangers, pumps, and associated valves.

The highest radiation level components include the demineralizers and heat exchangers. The demineralizers are located in separate concrete-shielded cubicles that are accessible through shielded hatches. The demineralizer rooms in the Reactor Building are identified in Figure 12.3-72. The radiation source term associated with the demineralizers is provided in Table 12.2-7. Adjacent rooms to the demineralizers are identified in Figures 12.3-71 through 12.3-73. Valves and piping within the cubicles are reduced to the extent that entry into the cubicles is not required during any operational phase. Most of the valves and piping are located in a shielded valve gallery adjacent to the demineralizer cubicles. The valves are remotely operable to the greatest practical extent to minimize entry requirements into this area. The RWCU/SDC heat exchangers are also located in a shielded cubicle with valves operated remotely by use of extension valve stems, or from instrument panels located outside the cubicle. The backwash tank is shielded separately from the resin transfer pump, permitting maintenance of the pump without being exposed to the spent resins contained in the backwash tank. The pump valves are operated remotely from outside the cubicle. The resin pump valves are operated remotely from outside the cubicle.

The RWCU/SDC is provided with chemical cleaning and decontamination connections that can utilize the condensate system to flush piping and equipment prior to maintenance to provide decontamination of pumps, the shell side of the RHX, the tube side of both the RHX and NRHX taken together. The RWCU/SDC demineralizer can be remotely back-flushed to remove spent resins. If additional decontamination is required, chemical addition connections are provided in the piping to clean piping as well as equipment prior to maintenance. The HVAC System is designed to limit the spread of contaminants from these shielded cubicles by maintaining a negative pressure in the cubicles relative to the surrounding areas.

Personnel access to the cubicles for maintenance of these components is on a controlled basis whereby specific restrictions and controls are implemented to minimize personnel exposure.

12.3.1.4.2 Fuel and Auxiliary Pools Cooling System

This system is designed to operate continuously to handle the spent fuel cooling load and to reduce pool water radioactive contamination in all the major pools in the ESBWR. The system components are located in the fuel and reactor building. Included are two independent filter demineralizer units that serve to remove radioactive contamination from the fuel pool and suppression water during cleanup and Low Pressure Coolant Injection (LPCI) mode. These units are the highest radiation level components in the system. Each unit is located in a concrete-shielded cubicle that is accessible through a shielded hatch. The demineralizer rooms in the Fuel Building are identified in Figure 12.3-72. The radiation source term associated with the demineralizers is provided in Table 12.2-8a. Adjacent rooms to the demineralizers are identified in Figures 12.3-71 through 12.3-73. Provisions are made for remotely backflushing the units when filter and resin material are spent. This removal of radioactivity contaminated material reduces the component radiation level considerably and serves to minimize exposures during maintenance. All valves (inlet, outlet, recycle, vent, and drain) to the filter demineralizer units are located outside the shielded cubicles in a separate shielded cubicle together with associated piping, headers, and instrumentation. The radiation level in this cubicle is sufficiently low to permit required maintenance to be performed. Piping potentially containing resin is continuously sloped downward to the backwash tank. The system also includes two low radiation level heat exchangers and two circulation pumps.

All of the shielded system components are consolidated in the same section of the reactor building. Personnel access to shielded system components is controlled to minimize personnel exposure. Shielding for the components is designed to reduce the radiation level to less than 1 mR/hr in adjacent areas where normal access is permitted.

Operation of the system is accomplished from the MCR and local control panels located where designed radiation levels are less than 25μSv/hr (2.5 mR/h) and normal personnel access is permitted.

12.3.1.4.3 Main Steam System

All radioactive materials in the main steam system, located in the main steam-feedwater pipe tunnel of the reactor building, result from radioactive sources carried over from the reactor during plant operation, including high energy short-lived Nitrogen-16. During plant shutdown, residual radioactivity from prior plant operation is the radiation source.

Access to the main steam pipe tunnel in the reactor building is controlled. Entry into the reactor building steam tunnel is through a controlled personnel access door shielded by a concrete labyrinth to attenuate radiation streaming from the steam lines to adjoining areas. During reactor operation, the steam tunnel is not accessible except in the hot standby conditions under regulated access.

Providing valve drains that are piped to equipment drain sumps minimizes leakage from selected valves into surrounding areas. Floor drains are provided to minimize the spread of contamination should a leakage occur.

Penetrations through the steam tunnel walls are minimized to reduce the streaming paths made available by these penetrations. Penetrations through the steam tunnel walls, when they are required, are located so as to exit in controlled access areas or in areas that are not aligned with the steam lines. A lead-loaded silicone foam is employed whenever possible for these penetrations to reduce the available streaming area presented.

12.3.1.4.4 Inclined Fuel Transfer System (IFTS)

The inclined fuel transfer tube transits, through a shielded tube, 21P1, and rooms 18P2 and 1702, with no connection to any other room or area that could be potentially accessible during fuel transfer operations are shown on Figure 9.1-2. Accessible areas and rooms adjacent to the inclined fuel transfer tube and rooms are shielded and with radiation levels lower than 1 Sv per hour as shown on Figures 12.3-1 through 12.3-4 and 12.3-5 through 12.3-8.

Access from any area adjacent to the transfer tube is controlled through a system of physical controls, interlocks and an annunciator.

During IFTS operation or shutdown, personnel are prevented from (a) either reactivating the IFTS while personnel are in a controlled maintenance area, or (b) entering a controlled IFTS maintenance area while irradiated fuel or component are in any part of the IFTS.

Both an audible alarm and flashing red lights are provided inside and outside any IFTS maintenance area indicating IFTS operation.

Radiation monitors with alarms are provided both inside and outside any maintenance area.

A keylock system of a key lock in both the IFTS main operation panel and in the control room is provided to allow access to any IFTS maintenance area.

12.3.2 Shielding

12.3.2.1 General Design Guides

The primary objective of the radiation shielding is to protect operating personnel and the general public from radiation emanating from the reactor, the power conversion systems, the radwaste process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance. The radiation shielding is also designed to keep radiation doses to equipment below levels at which disabling radiation damage occurs.

Specifically, the shielding requirements in the plant are designed to perform the following functions:

- Limit the exposure of the general public, plant personnel, contractors, and visitors to levels that are ALARA and within 10 CFR 20 requirements;
- Limit the radiation exposure of personnel, in the unlikely event of an accident, to levels that are ALARA and which conform to the limits specified in 10 CFR 50, Appendix A, Criterion 19 to ensure that the plant is maintained in a safe condition during an accident; and
- Limit the radiation exposure of critical components within specified radiation tolerances, to assure that component performance and design life are not impaired.

12.3.2.2 Design Description

12.3.2.2.1 General Design Guides

In order to meet the above design objectives, the following design guides are used in the shielding design of the ESBWR:

- All systems containing radioactivity are identified and shielded based on access and exposure level requirements of surrounding areas. The radiation zone maps described in Subsection 12.3.1.3 indicate design radiation levels for which shielding for equipment contributing to the dose rate in the area is designed.
- The source terms used in the shielding calculations are analyzed with a conservative approach. Transient conditions as well as shut down and normal operating conditions are considered to ensure that a conservative source is used in the analysis. Shielding design is based on fission product quantities in the coolant corresponding to the design basis off-gas release, in addition to activation products. This is considered an anticipated operational occurrence, and hence represents conservatism in design. For components where N^{16} is the major radiation source, a concentration based upon operating plant data is used.
- Effort is made to locate processing equipment in a manner that minimizes the shielding requirements. Shielded labyrinths are used to eliminate radiation streaming through access ways from sources located in cubicles.

- Penetrations through shield walls are located so as to minimize the effect on surrounding areas due to radiation streaming through the penetrations. The approaches used to locate and shield penetrations, when required, are discussed in Subsection 12.3.1.2.
- Wherever possible, radioactive piping is run in a manner that minimizes radiation exposure to plant personnel. This involves:
  - Minimizing radioactive pipe routing in corridors;
  - Avoiding the routing of high-activity pipes through low-radiation zones;
  - Use of shielded pipe trenches and pipe chases, where routing of high-activity pipes in low-level areas cannot be avoided; and
  - Separating radioactive and non-radioactive pipes for maintenance purposes.
- To maintain acceptable levels at the valve stations, motor-operated or diaphragm valves are used where practical. For valve maintenance, provision is made for draining and flushing associated equipment so that radiation exposure is minimized. If manual valves are used, provision is made for shielding the operator from the valve by use of shield walls and valve stem extensions, where practicable.
- Shielding is provided to permit access and occupancy of the control room to ensure that plant personnel exposure following an accident does not exceed the guideline values set forth in 10 CFR 50, Appendix A, Criterion 19. The analyses of the doses to Control Room personnel for the design basis accidents are included in Chapter 15.
- The dose at the site boundary as a result of direct and scattered radiation from the turbine and associated equipment is considered.
- In selected situations, provisions are made for shielding major radiation sources during inservice inspection to reduce exposure to inspection personnel. For example, steel platforms are provided for ISI of the RPV nozzle welds and associated piping.
- The primary material used for shielding is concrete at a density of 2.35 gm/cm^3. Concrete used for shielding purposes is designed in accordance with Regulatory Guide 1.69 (Reference 12.3-12). Where special circumstances dictate, steel, lead, water, lead-loaded silicone foam, or a boron-laced refractory material is used.

12.3.2.2.2 Method of Shielding Design

The radiation shield wall thicknesses are determined using basic shielding data and proven shielding codes. A list of the computer programs used is contained in Table 12.3-1. The shielding design methods used also rely on basic radiation transport equations contained in Reference 12.3-1. The sources for basic shielding data, such as cross sections, buildup factors, and radioisotope decay information, are listed in References 12.3-2 through 12.3-10.

The shielding design is based on the plant operating at maximum design power with the release of fission products resulting in a source of noble gas after a 30-minute decay period, and the corresponding activation and corrosion product concentrations in the reactor water listed in Section 11.1. Radiation sources in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions, such as a

Loss of Coolant Accident (LOCA) or a Fuel Handling Accident (FHA), have also been considered in designing shielding for the plant.

The mathematical models used to represent a radiation source and associate equipment and shielding are established to ensure conservative calculation results. Depending on the versatility of the applicable computer program, various degrees of complexity of the actual physical situation are incorporated. In general, cylindrically shaped equipment such as tanks, heat exchangers, and demineralizers are mathematically modeled as truncated cylinders. Equipment internals are sectional and homogenized to incorporate density variations where applicable. For example, the tube bundle section of a heat exchanger exhibits a higher density than the tube bundle clearance circle, due to the tube density, and this variation is accounted for in the model. Complex piping runs are conservatively modeled as a series of point sources spaced along the piping run. Equipment containing sources in a parallelepiped configuration, such as fuel assemblies and fuel racks are modeled as parallelepiped with a suitable homogenization of materials contained in the equipment. The shielding for these sources is also modeled on a conservative basis, with discontinuities in the shielding, such as penetrations, doors, and partial walls accounted for. The dimension of the floor decking is not considered in the shielding calculation as it is part of the effective shield thickness provided by the floor slab.

Pure gamma dose rate calculations, both scattered and direct, are conducted using point kernel codes (QADF/GGG). The source terms are divided into groups as a function of photon energy, and each group is treated independently of the others. Credit is taken for attenuation through all phases of material, and buildup is accounted for using a third-order polynomial buildup factor equation. The more conservative material buildup coefficients are selected for laminated shield configuration to ensure conservative results.

For combined gamma and neutron shielding situations, discrete ordinates techniques (DORT) are applied.

The shielding thicknesses are selected to reduce the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone specified in the zone maps in Subsection 12.3.1.3. By maintaining dose rates in these areas at less than the upper limit values specified in the zone maps, sufficient access to the plant areas is allowed for maintenance and operational requirements.

Where shielded entries to high-radiation areas such as labyrinths are required, a gamma ray scattering code (GGG) is used to confirm the adequacy of the labyrinth design. The labyrinths are designed to reduce the scattered as well as the direct contribution to the aggregate dose rate outside the entry, such that the radiation zone designated for the area is not violated.

12.3.2.2.3 Plant Shielding Description

The general description of the shielding is provided below:

Containment - The major shielding structures located in the drywell area consist of the reactor shield wall and the drywell wall. The reactor shield wall in general consists of 16 cm of steel plate. The primary function served by the reactor shield wall is the reduction of radiation levels in the drywell due to the reactor, to valves that do not unduly limit the service life of the equipment located in the drywell. In addition, the reactor shield wall reduces gamma heating effects on the drywell wall, as well as providing for low radiation levels in the drywell during reactor shutdown. The drywell is an F radiation zone during full power reactor operation and is not accessible during this period.

The containment (drywell) outside wall is a 2 m thick reinforced concrete cylinder that totally surrounds the drywell. A 2.4 m thick reinforced concrete containment top slab tops the drywell. The drywell wall attenuates radiation from the reactor and other radiation sources in the drywell to allow occupancy of the reactor building during full power reactor operation.

The ESBWR plant includes all necessary shielding provisions in the upper drywell in order to reduce the dose ALARA during transfer of irradiate spent fuel assemblies. In such a way ESBWR plant includes all applicable shielding design provisions to minimize dose rates in case of fuel handling mishap resulting in dropping a fuel assembly across the reactor flange.

Reactor Building - In general, the shielding for the reactor building is designed to maintain open areas at dose rates less than 6 μSv/hr (0.6 mR/hr).

Penetrations of the containment wall are shielded to reduce radiation streaming through the penetrations. Localized dose rates outside these penetrations are limited to less than 50 μSv/hr (5 mR/hr). The penetrations through interior shield walls of the reactor building are shielded using a lead-loaded silicone sleeve to reduce the radiation streaming. Penetrations are also located so as to minimize the consequences of radiation streaming into surrounding areas.

The components of the RWCU/SDC are located in the reactor building. Both the RWCU regenerative and nonregenerative heat exchangers are located in shielded cubicles separated from the other components of the system. Neither cubicle needs to be entered for system operation.

Process piping between the heat exchangers and the demineralizers is routed through shielded areas or embedded in concrete to reduce the dose rate in surrounding areas. The RWCU/SDC demineralizers are located in separate shielded cubicles. This arrangement allows maintenance of one unit while operating the other. The dose rate in the adjoining demineralizer cubicle from the operating unit is less than 60 μSv/hr (6 mR/hr). Entry into the demineralizer cubicle, which is required infrequently, is via a labyrinth entryway. The bulk of the piping and valves for the filter demineralizers is located in an adjacent shielded valve gallery. Backfilling and resin application of the filter demineralizers are controlled from an area where dose rates are less than 10 μSv/hr (1 mR/hr).

The ESBWR employs a passive cooling system in addition to the RWCU/SDC for cooling the core and vessel. Access into the cubicles is not required to operate the systems. All such components that could become contaminated in the event of an accident are located in the containment except those components that would be used as part of the RWCU/SDC.

Fuel Storage - The fuel storage pool is designed to ensure the dose rate around the pool area is less than 25 μSv/hr (2.5 mR/hr). In the event of an anticipated operational occurrence where the fuel sustains significant damage, such as a fuel drop accident, airborne dose rates in the pool area may significantly exceed this dose rate.

Control Room - The dose rate in the control room is limited to 6 μGy/h during normal reactor operating conditions. The outer walls of the Control Building are designed to attenuate radiation from radioactive materials contained within the Reactor Building and from possible airborne radiation surrounding the Control Building following a LOCA. The walls provide sufficient shielding to limit the direct-shine exposure of control room personnel following a LOCA to a fraction of the 5 Rem limit as is required by 10 CFR 50 Appendix A, Criterion 19.

Main steam tunnel - The main steam tunnel extends from the primary containment boundary in the Reactor Building up to the turbine stop valves. The primary purpose of the steam tunnel is to shield the plant complex from N-16 gamma shine in the main steam lines. The tunnel walls provide sufficient shielding to limit the direct-shine exposure from the main steam lines in any point that may be inhabited during normal operations.

12.3.3 Ventilation

The HVAC systems for the various buildings in the plant are discussed in Section 9.4, including the design bases, system descriptions, and evaluations with regard to the heating, cooling, and ventilating capabilities of the systems. This Subsection discusses the radiation control aspects of the HVAC systems.

12.3.3.1 Design Objectives

The following design objectives apply to all building ventilation systems:

- The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.
- The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance will be below the concentrations that define an airborne radioactive area in 10 CFR 20 during normal power operation. This is accomplished by establishing in each area a reasonable compromise between specifications on potential airborne leakages in the area and HVAC flow through the area. Appendix 12.A to this chapter outlines the methodology by which such calculations are made.

12.3.3.2 Design Description

In the following subsections, the design features of the various ventilation systems that achieve the radiation control design objectives are discussed. For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

12.3.3.2.1 Control Room Ventilation

The control building atmosphere is maintained at a slightly positive pressure (up to 0.5 in. Hg) at all times in order to prevent infiltration of contaminants. When offsite power is available, fresh air may be taken in via the single inlet system, which has its intake structure on the side of the building. During an isolation event if offsite or backup power is not available, bottled air can be supplied by a redundant supply system for up to 72 hours prior to requiring recharging. Under conditions when offsite or backup power is available, either bottled or filtered air may be used. The operator has manual control in the event filtered air is used to either run under filtered air or bottled air.

Outside a particulate filter normally filters air that enters the intake. Under contamination conditions, however, if external air is selected, the airflow is diverted through an adsorber system having:

- a particulate filter
- a HEPA filter
- a charcoal filter
- another HEPA filter

The outdoor cleanup unit is located in a closed room that helps prevent the spread of any radiation during maintenance. Adequate space is provided for maintenance activities. The particulate and HEPA filters can be bagged when being removed from the unit. Before removing the charcoal, any radioactivity is allowed to decay to minimal levels, and is then removed through a connection in the bottom of the filter by a pneumatic transfer system. Air used in the transfer system goes through a HEPA filter before being exhausted. Facemasks can be worn during maintenance activities, if necessary.

For a complete description of the control room HVAC system see Subsection 9.4.1.

12.3.3.2.2 Containment

Access into the containment drywell is not permitted during normal operation. The ventilation system inside merely circulates, without filtering, the air. The only airflow out of the drywell into accessible areas is minor leakage through the wall. During maintenance, the drywell air is purged before access is allowed.

12.3.3.2.3 Reactor Building

The reactor building HVAC system is divided into two major components: the contaminated and the clean areas. The clean area system conditions and circulates air through all the clean areas of the reactor building. The contaminated area system conditions and circulates air through the contaminated areas of the building. Flow into both areas is directed from the corridors (point of highest pressure) to the equipment alcove rooms, then to the rooms themselves, and finally to the external wall pipe chases and from the pipe chases back to the HVAC system. The clean area system dumps circulated air to the environment through building vents, while the contaminated air system directs flow through the HVAC system to the plant stack. Under isolation conditions, the HVAC system isolates to localize any contamination until operations and health physics personnel determine the best decontamination method.

For a description of the reactor building HVAC system, see Subsection 9.4.6.

12.3.3.2.4 Radwaste Building

The radwaste building is divided into two zones for ventilation purposes. The control room is one zone, and the remainder of the building is the other zone. The air pressure in the first zone is maintained slightly above atmospheric, while the air pressure in the second zone is maintained slightly below atmospheric. Air in the second zone is drawn from outside the building and distributed to various work areas within the building. Air flows from the work areas and is then discharged via the reactor building stack. An alarm sounds in the control room if the exhaust fan fails. The exhaust flow is monitored for radioactivity, and if a high activity level is detected, the potentially radioactive cells are automatically isolated, but airflow through the work areas continues.

If the exhaust flow high-radiation alarm continues to annunciate after the tank and pump rooms are isolated, the work area branch exhaust ducts are selectively manually isolated to locate the involved building area. Should this technique fail, because the airborne radiation has spread throughout the building, the control room air conditioning continues, but the air conditioning for the balance of the building is shut down.

The work area's exhaust air is drawn through a filter unit consisting of a particulate filter and a HEPA filter before being discharged to the reactor building stack. The air is monitored for radioactivity, and if a high level is detected, supply and exhaust is terminated.

Maintenance provisions for the filters are similar to those for the control building HVAC System.

See Subsection 9.4.3 for a detailed discussion of the radwaste building HVAC System.

12.3.3.2.5 Fuel Building

The Fuel Building HVAC System (FBVS) consists of the Fuel Building General Area HVAC Subsystem (FBGAVS) and the Fuel Building Fuel Pool Area HVAC Subsystem (FBFPVS). The FBGAVS serves the general area. The FBFPVS serves the refueling floor and pool areas. The FBVS operates during normal plant operation, plant startup, and plant shutdown.

The FBGAVS consists of two 100% capacity AHUs with two 100% capacity supply fans, two 100% capacity exhaust fans, recirculation AHUs, and unit heaters. The FBGAVS incorporates a common supply and return duct system that distributes conditioned air to the general area of the Fuel Building and exhaust air to the outside atmosphere. During normal operation, air travels through the AHU's stages where particulates are removed from the air by low and high efficiency filters; heat is transferred between the mixed air and the hot/chilled water coils; and the conditioned air is distributed to the clean areas by the supply fan. Exhaust air is ducted to the exhaust fan and exhausted to the outside atmosphere.

The FBFPVS consists of two 100% capacity AHUs with two 100% capacity supply fans, two 100% capacity exhaust fans, and redundant bubble-tight isolation dampers. The FBFPVS is a once-through ventilation system that distributes conditioned air to the refueling area of the reactor and spent fuel pool area of the Fuel Building. During normal operation, outside air travels through the AHU's stages where particulates are removed from the air by low and high efficiency filters; heat is transferred between the air and the hot/chilled water coils; and the conditioned air is distributed to the refueling area and spent fuel pool surfaces. Air is ducted to the exhaust fan and exhausted to the outside atmosphere through the plant vent stack. The exhaust system has the manual capability to divert the exhaust for filtration by the purge exhaust filter unit, prior to discharge to the plant vent stack. FBFPVS exhaust fans are used for smoke removal.

The common plant vent stack provides monitoring and discharging of FBGAVS and FBFPVS exhausts. See Subsection 9.4.2 for a detailed discussion of the FBVS.

12.3.3.3 Accident Conditions

The ventilation systems filter units designed to operate during accident conditions are the Reactor Building HVAC Filter Units and the Control Building Emergency Filter Unit (EFU).

To determine the radiation level in the HVAC filters in accident conditions, the LOCA (Loss-of-Coolant Accident) event is postulated.

The source term of the Reactor Building HVAC filter for accident dose assessment is the LOCA Inventory in Reactor Building obtained following the assumptions of Regulatory Guide 1.183 (Reference 12.3-16). The source term of the Control Building EFU for accident dose assessment is the LOCA inventory in the environment obtained following the assumptions of Regulatory Guide 1.183. The activity retained in the filters over 30 days is shown in Table 12.3-9.

In order to maintain the exposure from filter maintenance ALARA, the shielding wall thickness between RB HVAC filter cubicles is sized so that the dose contribution in any cubicle from the filter in the adjacent one does not exceed 250 mSv/hr.

For the Control Building EFU, the dose rates in the filter and adjacent rooms in accident conditions are shown in Table 12.3-10.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The following systems are provided to monitor area radiation and airborne radioactivity within the plant:

- The Area Radiation Monitoring System (ARMS) continuously measures, indicates and records the gamma radiation levels at strategic locations throughout the plant except within the primary containment, and activates alarms in the main control room as well as in local areas to warn operating personnel to avoid unnecessary or inadvertent exposure to radiation. This system is classified as nonsafety-related.
- The Containment Monitoring System (CMS) continuously measures, indicates, and records the gamma radiation levels within the primary containment (drywell and suppression chamber), and activates alarms in the main control room on high radiation level. As described in Subsection 7.5.2, four gamma sensitive ion chambers are provided within the primary containment to monitor gamma rays during normal, abnormal and accident conditions. Two redundant sensors are located in the drywell and two in the wetwell. The monitors are located such that they are widely separated to provide independent measurements, with a large fraction of the containment volume considered in both the wetwell and drywell. Further, the selection of the location considers reasonable access for personnel to allow for replacement, maintenance and calibration of equipment. The range of each monitor covers 7 decades from 0.01 Gy/hr (1R/hr) to 10^5 Gy/hr (10^7 R/hr) as required by RG 1.97 (Reference 12.3-13). The CMS is classified as safety-related. The radiation monitors have been designed in accordance with NUREG-0737, Item II.F.1 (Reference 12.3-17).
- Airborne radioactivity in effluent releases and ventilation air exhausts is continuously sampled and monitored by the Process Radiation Monitoring System (PRMS) for noble gases, air particulates and halogens. As described in Section 11.5, airborne contamination is sampled and monitored at the stack common discharge, in the off-gas

releases, and in the ventilation exhaust from the reactor, radwaste and turbine buildings. Samples are periodically collected and analyzed for radioactivity. In addition to this instrumentation, portable air samplers are used for compliance with 10 CFR 20 restrictions to check for airborne radioactivity in work areas prior to entry where potential radiation levels may exit that exceed the allowable limits.

- The in-plant airborne radiation monitoring instrumentation is located so that selected local areas and ventilation paths are monitored. Each location monitored is supplied with a local audible alarm and the monitor has variable alarm set points. When appropriate, selected airborne radioactivity sampling points are located upstream of any ventilation filter trains to monitor representative radioactivity concentrations from the areas being sampled. Plant operating personnel are supplied with continuous information about the airborne radioactivity levels throughout the plant. The instruments used for monitoring airborne radioactivity are specified to detect the time integrated change of the most limiting particulate and iodine species equivalent to those concentrations specified in Appendix B of 10 CFR Part 20 (one derived air concentration (DAC)) in each monitored plant area within 10 hours). Locations are selected based on the potential for leakage into rooms and areas that contain radioactive processes that become airborne and for which personnel occupancy is required for operation of the reactor plant.
- The radiation instrumentation that monitors airborne radioactivity is classified as nonsafety-related. Airborne radiation monitoring operational considerations such as the procedures for operations and calibration of the monitors, as well as the placement of the portable monitors, are the COL applicant's responsibility (COL 12.3-2-A).

12.3.4.1 ARM System Description

Every ARM channel consists of a gamma sensitive detector and a digital area radiation processor; all channels are provided with local visual and audible alarms and local readouts. Where appropriate, additional readouts and alarms, provided by local auxiliary units, will be utilized. The output signals from the detectors are digitized and multiplexed for transmission to digital radiation monitors for measurement and display. Also, the radiation signals are transmitted to the process computer for recording. Each radiation monitoring channel has two adjustable trip alarm circuits, one for high radiation and the other for downscale indication (loss of sensor input). Also, each area radiation monitor has a built-in self test capability that checks for gross failures and activates an alarm on power failure or inoperative monitor. Auxiliary units with local audible alarms are provided in selected local areas to provide for immediate warning in order to minimize occupational exposure. Each area radiation monitor is powered from non-1E vital 120 VAC power source, which is continuously available during loss of off-site power.

12.3.4.2 ARM Detector Location and Sensitivity

The detector locations are shown on plant layout drawings for each building (Figures 12.3-23 through 12.3-42). The area radiation channels for each building are listed in Tables 12.3-2 through 12.3-6, along with reference to the figure that shows the detector location, the channel monitoring range, and the local area alarms assignment. The monitoring range of each area radiation channel is shown in Table 12.3-7.

12.3.4.3 Pertinent Design Parameters and Requirements

Two high-range radiation channels are provided in the fuel transfer and storage area to monitor radiation that may result from a fuel handling accident. Criticality detection monitors are not needed to satisfy the criticality accident requirements of 10 CFR 70.24 since the ESBWR design utilizes high-density fuel storage racks that are designed to be subcritical under normal, abnormal and accident conditions. The new fuel bundles are stored under water in storage racks that are located in the fuel vault adjacent to the reactor cavity, while the spent fuel bundles are stored in racks that are placed at the bottom of the fuel storage pool. A full array of loaded new or spent fuel storage racks is designed to be subcritical as defined in Subsections 9.1.1 and 9.1.2, respectively.

Both Process Radiation Monitors and Area Radiation Monitors, are located in the fuel storage and associated handling areas in order to detect excessive radiation levels, and are used to demonstrate compliance with 10 CFR 50.68(b)(6) (Reference 12.3-18).

Process Radiation Monitors, described in Subsection 11.5.3, monitor ventilation paths from the fuel storage area and, in addition to isolating the appropriate ventilation path upon receipt of high radiation, provide indication and alarms to the operator. Area Radiation Monitors, listed in Table 12.3-2, are provided in fuel storage areas to detect high radiation levels and provide visual and audible indication to operating personnel.

The detectors and radiation monitors are responsive to gamma radiation over an energy range of 80 keV to 7 MeV. The energy dependence does not exceed 20% of point from 100 keV to 3 MeV. The overall system design accuracy is within 10% of equivalent linear full-scale output for any decade.

The alarm setpoints are established in the field following equipment installation at the site. The exact settings are based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures. The radiation alarm setpoint for each channel is set slightly above the background radiation level that is normal in the area where the monitor is located.

Each channel is calibrated based on a pseudo input signal to verify monitor response. Each detector is calibrated using a radioactive source traceable to the National Institute of Standards and Technology. The area radiation monitors are checked and calibrated periodically.

The ARMS is designed to provide early detection and warning for personnel protection to ensure occupational radiation exposures are as low as reasonably achievable (ALARA) in accordance with guidelines stipulated in RG 8.2 (Reference 12.3-14) and RG 8.8. Also, the ARMS include instrumentation in crucial areas of the reactor building where access may be required to service safety-related equipment following a LOCA event.

12.3.5 Post-Accident Access Requirements

The locations requiring access to mitigate the consequences of an accident during the 100-day post-accident period are the main control room, the technical support center, the remote shutdown panel, the primary containment sampling locations, the health physics facility (counting room), the isolation condenser (IC) pool refill nozzles, and the nitrogen gas supply bottles. Each area has low post LOCA radiation levels. The dose evaluations are within regulatory guidelines.

Access to vital areas through out the reactor building/control building/turbine building complex is controlled via the service building. Entrance to the service building and access to the other areas are controlled via double locked secured entryways. Access to the reactor building is via two specific routes, one for clean access and the second for controlled access. During an event such as a design basis accident, the control building is maintained under filtered HVAC at positive pressure with respect to the environment. Air infiltration is minimized by positive flow via double entry ways. Therefore, radiation exposure is limited to gamma shine from the reactor building, turbine building, main steam line access corridor, and skyline.

During a design basis accident event, access to nitrogen bottles and monitor systems is controlled from the service building via the controlled access way. These corridors are not maintained under filtered positive pressure so personal protection equipment (radiation protection suits, breathing gear, etc.) is required in the access corridor. Primary contamination would occur from leakage through primary containment sampling locations. This pathway is considered minimal and minor contamination under even the most adverse conditions is expected. Access to the IC pool refill nozzles is from outside and would not likely require special breathing gear, etc.

The reactor building vital areas are all located off the controlled access way and contamination is limited to air infiltration from the accident environment and penetration leakage from primary containment sampling locations. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from primary containment sampling locations . These sources are considered minimal including the stack monitor room, which contains only instrumentation with their associated penetrations for monitoring stack effluent.

12.3.6 Post-Accident Radiation Zone Maps

The post-accident radiation zone maps for the areas in the reactor building are presented in Figures 12.3-43 through 12.3-51. The zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period. These dose rates do not include the airborne contribution in the reactor building. The zone maps are designed to reflect the criteria established in Subsection 3.1.2.

12.3.7 COL Information

12.3-1-H Facility Design Features (Deleted)

12.3-2-A Operational Considerations

Airborne radiation monitoring operational considerations such as the procedures for operations and calibration of the monitors, as well as the placement of the portable monitors, are the COL applicant’s responsibility (Subsection 12.3.4).

12.3-3-A Controlled Access

Controlled access to “Very High Radiation Areas” is provided by the COL applicant (Subsection 12.3.1.3).

12.3.8 References

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Table 12.3-1

Computer Programs Used in Shielding Design Calculations

Computer Code	Description
QADF	A multi-group, multi-region, point kernel gamma radiation code for calculating the flux and dose rate at discrete locations within a complex source geometry configuration.
GGG	A multi-group, multi-region, point kernel code for calculating the contributions due to gamma ray scattering in a heterogeneous three-dimensional space.
DORT	A discrete ordinates two-dimensional transport code. Multi-group, multi-region neutron or gamma transport.
QAD CGGP 1.0	"Quick and Dirty Combinatorial Geometry –Geometric Progression". A multi-group, multi-region, point kernel gamma radiation code for calculating the flux and dose rate at discrete locations within a complex source geometry configuration
SKYIII-PC	A Monte Carlo skyshine code designed to aid in the evaluation of the effects of structure geometry on the gamma-ray dose rate at given detector positions outside of a building housing N-16 gamma-ray sources.

Table 12.3-2

Area Radiation Monitors for Reactor Building

No. [1]	Description & Location	Figure No.	Monitoring Range
1.	Refueling Floor Area #1, EL 34000	12.3-31	H
2.	Refueling Floor Area # 2, El 34000	12.3-31	H
3.	New Fuel Storage Pool, EL 27000	12.3-30	H
4.	New Fuel Storage Pool, EL 27000	12.3-30	H
17.	RWCU/SDC Pump, EL -11500	12.3-23	H
18	RB Sump Pumps, EL -11500	12.3-23	H
19.	RWCU/SDC Train A Heat Exchanger EL -11500	12.3-23	H
20.	RWCU/SDC Train B Heat Exchanger EL -11500	12.3-23	H
21.	Equipment Hatch Pathway, EL -6400	12.3-24	M
22.	Personnel Hatch Pathway, EL -6400	12.3-24	H
23	FMCRD HCU Area #1, EL -6400	12.3-24	M
25.	FMCRD HCU Area # 3, EL -6400	12.3-24	M
27.	RWCU/SDC Filter Demineralizer Area (Near Equip. Hatch), EL -1000	12.3-25	H
28.	Radiological Control Area Entrance, EL 17500	12.3-29	M
29	Hydrogen/Oxygen Monitoring (CMS), Skid EL 13570	12.3-28	H
30.	Hydrogen/Oxygen Monitoring (CMS), Skid EL 13570	12.3-28	H
31	Instrument Rack Area #1, EL -11500	12.3-23	H
32.	Instrument Rack Area #2, EL -11500	12.3-23	H
33	Instrument Rack Area #3, EL -11500	12.3-23	H
34	Instrument Rack Area #4, EL -11500	12.3-23	H
35	Instrument Rack Area #5, EL -11500	12.3-23	H
36	Instrument Rack Area #6, EL -11500	12.3-23	H
37	Instrument Rack Area #7, EL -11500	12.3-23	H
38	Instrument Rack Area #8, EL -11500	12.3-23	H
39 [2]	Fuel Transfer System (FTS) Maintenance Room (Multiple), EL 17500	12.3-29	H
40	Fuel Handling Machine (IFTS), EL 34000	12.3-31	H
41	Remote Shutdown Panel A Area, El. –1000	12.3-25	H
42	Remote Shutdown Panel B Area, El. –1000	12.3-25	H

[1] Notes #5 through 16, 24 and 26 not used

[2] Utilizes auxiliary units

Table 12.3-3

Area Radiation Monitors for Fuel Building

No.[1]	Description & Location	Figure No.	Monitoring Range
1.	Spent Fuel Floor, EL 4650	12.3-26	H
2.	Fuel Handling Machine, EL 4650	12.3-26	M
3.	Fuel Transfer Cask Area, EL 4650	12.3-26	H
5	FAPCS Heat Exchangers, EL -11500	12.3-23	H
6	FAPCS System Transfer Pumps, EL -11500	12.3-23	H
9	Sump Pumps, EL -11500 H	12.3-23	H
10	Ground Grade Access Pathway, EL 4650	12.3-26	M
11	Wash Down Bay Entry Door, EL 4650 (Truck)	12.3-26	H
12	Fuel Transfer System (FTS) Maintenance Rooms (Multiple) EL 4650	12.3-26	H

[1] Notes #4, 7 & 8 not used

Table 12.3-4

Area Radiation Monitors for Radwaste Building

No.	Description & Location	Figure No.	Monitoring Range
1.	Electrical Board Room, El - 9350	12.3-39	H
2.	Control Room	12.3-39	H
3.	High Activity Resin Recirculation Pump Room, El - 9350	12.3-39	H
4.	High Activity Resin Transfer Pump Room, El - 2350	12.3-39	H
5.	Trailer Access Area, El - 4650	12.3-41	H
6.	Liquid Radioactive Waste Treatment Area (Hallover Fiber deep-Bed Demineralizer, Reverse Osmosis System, etc.) El - 4650	12.3-41	H
7.	Wet Solid Radioactive Waste Treatment Area (Dewatering Equipment, Concentrate Treatment System, etc.), El - 4650	12.3-41	H
8.	Dry Solid Waste Treatment Area (High Dose Rate Waste Storage Area, etc.) El - 4650	12.3-41	H
9.	Packaged Waste Staging Area, El - 4650	12.3-41	H

Table 12.3-5

Area Radiation Monitors for Turbine Building

No.[1]	Description & Location	Figure No.	Monitoring Range
1.	Main Condenser Floor Area EL -1400	12.3-32	M
2.	Drain Cooler Area EL 4650	12.3-33	M
3.	Offgas Sampling Area EL 4650	12.3-33	M
4.	Condensate Pumps Area EL -1400	12.3-32	M
5.	Low Pressure Heater Area EL 20000	12.3-35	M
6	Deaerator Area, EL 28000	12.3-36	M
7	SRV/MSIV Maintenance Area EL 20000	12.3-35	M
8	Steam Jet Air Ejector (SJAE) B Area EL 4650	12.3-33	M
9	SJAE A Area EL 4650	12.3-33	M
10	High Pressure Heater Area EL 20000	12.3-35	M
11	Filters and Demineralizers Area EL 4650	12.3-34	M
12	Turbine Operating Floor Area EL 28000	12.3-36	M
13.	Turbine Operating Floor Area EL 28000	12.3-36	M
14	Crane Travel Area (Various)	12.3-38	M
15	Equipment Main Access Area, EL 4650	12.3-33	M
16.	RCCW System Area Entrance EL 4650	12.3-33	M
17	Offgas Charcoal Adsorber Room Entrance Area EL -1400	12.3-32	M
18	Backwash Transfer [1]Pumps Entrance Area EL -1400	12.3-32	M
19	Condensate Hollow Fiber Filter Valve Room EL -1400	12.3-32	M
20	Sample Room Area EL -1400	12.3-32	M
22	Condensate D/B Demineralizer Entrance Area, EL 4650	12.3-33	
23	Offgas Hydrogen Recombiner A, EL 12000	12.3-34	M
24	Offgas Hydrogen Recombiner B, EL 4650	12.3-33	M
25	Instrument Air Compressor Area, EL 12000	12.3-34	M
26	MCC Water Chiller Room A, EL 28000	12.3-36	M
27	MCC Water Chiller Room B, EL 28000	12.3-36	M
28	Turbine Building Exhaust Duct Area EL 33000	12.3-37	M
29	RCCWS Area Entrance EL 4650	12.3-33	M

[1] Note #21 not used; #1, #3, #8, #11, #12, and #14 utilize auxiliary units

Table 12.3-6

Area Radiation Monitors for Control Building

No.	Description & Location	Figure No.	Monitoring Range	Local Alarms
1.	Main Control Room, EL -1000	12.3-25	H	

Table 12.3-7

Area Radiation Channel Monitoring Range

Low Setting	High Setting	Descriptor
1E-4 mSv/h; 1E-4 mGy/hr (1E-5 R/hr)	1E0 mSv/h, 1E0 mGy/hr (1E-1 R/hr)	H (High Sensitivity)
1E-3 mSv/h; 1E0 µGy/hr (1E-4 mR/hr)	1E1 mSv/h; 1E1 mGy/hr (1E0 R/hr)	M (Medium Sensitivity)
1E-2 mSv/h; 1E1 µGy/hr (1E0 mR/hr)	1E2 mSv/h; 1E2 mGy/hr (1E1 R/hr)	L (Low Sensitivity)
1E0 mSv/h; 1E0 mGy/hr (1E2 mR/hr)	1E4 mSv/h; 1E1 Gy/hr (1E6 mR/hr)	LL (Low-Low Sensitivity)
1E-4 Sv/h; 1E-1 mGy/hr (1E-2 R/hr)	1E2 Sv/h; 1E2 Gy/hr (1E4 R/hr)	VL (Very Low Sensitivity)

Table 12.3-8
Shielding Geometry in Centimeters

Elev.	Room	Room Name	North	East	South	West	Floor	Ceiling
Nuclear Island								
-11500	1151	RWCS Heat Exchanger Room A	75	111	100	100/75	Ground	70
-11500	1152	RWCS Pump Room A	60	55	55	60/40	Ground	90
-11500	1161	RWCS Heat Exchanger Room B	75	100	100/75	111	Ground	70
-11500	1162	RWCS Pump Room B	60	60	70	35	Ground	70
-11500	2102	FAPCS Backwash Tank Room	70	80	90	110	Ground	90
-11500	2150	FAPCS Pump/Heat Exchanger Room A	35	70	60	30	Ground	70
-11500	2151	FAPCS Filter/Demineralizer Vault 1	90	105	70	95	Ground	70
-11500	2160	FAPCS Pump/Heat Exchanger Room B	35	30	60	35	Ground	70
-11500	2161	FAPCS Filter/Demineralizer Vault 2	70	105	70	95	Ground	70
-6400	1250	RWCS Heat Exchanger Room A	111	111	100	100	70	70
-6400	1251	RWCS Filter/Demineralizer Vault A1	135	150	40	135	110	90
-6400	1252	RWCS Filter/Demineralizer Vault A2	40	150	40	135	110	90
-6400	1260	RWCS Heat Exchanger Room B	111	100	100	100	70	70
-6400	1261	RWCS Filter/Demineralizer Vault B1	135	40	150	40	110	90
-6400	1262	RWCS Filter/Demineralizer Vault B2	135	40	150	70	110	90
-6400	2251	Backwash Transfer Pump Room A	90	70	30	90	70	70
-6400	2261	FAPCS Filter/Demineralizer Vault 2	30	70	115	90	70	70
Radwaste Building								
-9350	6104	Equipment Drain Collection Tank Room B	70	60	60	80	Ground	80
-9350	6105	Equipment Drain Collection Tank Room C	60	60	80	80	Ground	80
-9350	6161	Low Activity Sludge Phase Separator Room B	100	60	70	40	Ground	80
-2350	6103	Equipment Drain Collection Tank Room A	70	60	60	60	80	80

Table 12.3-8

Shielding Geometry in Centimeters

Elev.	Room	Room Name	North	East	South	West	Floor	Ceiling
-2350	6106	Condensate Resin Holdup Tank Room	90	40	80	60	80	80
-2350	6107	Low Activity Resin Holdup Tank Room	100	80	60	100	80	80
-2350	6108	High Activity Resin Holdup Tank Room	70	90	70	100	80	80
-2350	6109	Concentrated Waste Tank Room	70	90	90	100	80	80
-2350	6150	Floor Drain Collection Tank Room A	70	80	60	60	80	80
-2350	6151	Low Activity Sludge Phase Separator Room A	100	90	70	40	80	80
-2350	6160	Floor Drain Collection Tank Room B	60	80	80	60	80	80
-2350	6171	Floor Drain Sample Tank Room	30	35	30	30	80	80
-2350	6172	Equipment Drain Sample Tank Room	30	35	30	30	80	80
Turbine Building								
-1400	4108	Off-Gas Charcoal Absorber Vessel Vault	150	150	120	120	Ground	-
-1400	4186	Main Condenser Area	60	40	70	120	Ground	-
-1400	4199	Pipe Chase	50	50	50	60	Ground	65
-1400	41F1A	Condensate Hollow Fiber Filter Vault A	50	50	30	60	Ground	65
-1400	41F1B/E	Condensate Hollow Fiber Filter Vault B/E	30	50	30	60	Ground	65
-1400	41F1F	Condensate Hollow Fiber Filter Vault F	30	50	50	60	Ground	65
-1400	41F2	Condensate Hff Backwash Receiving Vault	50	50	60	60	Ground	60
-1400	41F6	Condensate D/B Resin Receiving Tank Vault	50	100	50	80	Ground	60
-1400	41F8/9	Condensate D/B Demin. Resin Storage Tank Vault A/B	50	100	65	80	Ground	60
4650	4204	Feedwater Heaters 1A/B/C Drain Cooler Room	65	120	110	85	70	90/130
4650	4205	Condensate Drains Tank Room	100	100	110	120	100	120
4650	4206	Steam Jet Air Ejector Room A	120	100	100	120	70	30
4650	4207	Steam Jet Air Ejector Room B	100	100	100	120	100	120
4650	4293	Steam And Feedwater Piping Area	110	150	150	150	150	120/150

Table 12.3-8

Shielding Geometry in Centimeters

Elev.	Room	Room Name	North	East	South	West	Floor	Ceiling
4650	42F1A	Condensate Deep Bed Demineralizer Vault A	65	65	50	55	60	70
4650	42F1B/G	Condensate Deep Bed Demineralizer Vault B/G	50	75	50	55	60	70
4650	42F1H	Condensate Deep Bed Demineralizer Vault H	50	75	65	55	60	70
12000	4303	Feedwater Heaters 6A/B & 7A/B (Channel Side) Room	150	150	110	90	150	-
12000	4381	H2 Recombiner And Cooler Room A	120	80	60	120	30	120
12000	4382	H2 Recombiner And Cooler Room B	60	120	90	30	30	120
12000	4390	Main Steam Piping Area	0	115	150	115	-	-
12000	4391	Turbine Extr. Steam And Cross Around Piping Area A	110	90	110	0	-	120/-
12000	4392	Turbine Extr. Steam And Cross Around Piping Area B	110	-	110	150	-	120/-
12000	4393	Turbine Building Steam Tunnel	150	150	150	150	-	-
20000	4401	Feedwater Heaters 3A/B/C & 4A/B/C Shell Side Room	60	150	100	90	-	115
28000	4503	Feedwater Heater 5 & Feedwater Storage Tank Room	90	120	80	80	115	115
28000	4504	Turbine Gland Steam Seal Evaporator Room	80	60	60	80	115	115
28000	4506	Moisture Separator And Reheater Room A	135	80	115	-	-	130
28000	4507	Moisture Separator And Reheater Room B	135	-	115	150	-	130

Table 12.3-9

Activity Accumulated in the HVAC Filters in Accident Conditions After 30 Days

Isotope	Reactor Building-HVAC filter (MBq)	Control Building EFU (MBq)
Co-58	5.85E+03	1.99E+01
Co-60	7.08E+03	2.05E+01
Rb-86	1.38E+05	5.34E+02
Sr-89	8.36E+06	2.51E+04
Sr-90	1.13E+06	2.86E+03
Sr-91	6.41E+05	3.90E+03
Sr-92	1.31E+05	9.45E+02
Y-90	2.88E+05	4.14E+02
Y-91	1.36E+05	3.83E+02
Y-92	1.32E+05	5.91E+02
Y-93	8.45E+03	5.11E+01
Zr-95	1.61E+05	4.67E+02
Zr-97	1.59E+04	9.17E+01
Nb-95	1.77E+05	4.70E+02
Mo-99	5.21E+05	2.55E+03
Tc-99m	2.57E+05	9.42E+02
Ru-103	1.46E+06	4.56E+03
Ru-105	2.60E+04	1.75E+02
Ru-106	7.22E+05	1.87E+03
Rh-105	1.91E+05	9.99E+02
Sb-127	6.79E+05	3.18E+03
Sb-129	1.58E+05	1.06E+03
Te-127	4.33E+05	1.53E+03
Te-127m	3.51E+05	9.49E+02
Te-129	1.30E+05	5.42E+02
Te-129m	8.48E+05	2.74E+03
Te-131m	4.70E+05	2.57E+03
Te-132	8.53E+06	4.08E+04
I-131	6.20E+07	2.81E+05
I-132	4.66E+06	2.34E+04
I-133	3.16E+07	1.79E+05
I-134	4.46E+05	3.35E+03
I-135	1.01E+07	6.51E+04
Cs-134	2.30E+07	6.15E+04

Table 12.3-9

Activity Accumulated in the HVAC Filters in Accident Conditions After 30 Days

Isotope	Reactor Building-HVAC filter (MBq)	Control Building EFU (MBq)
Cs-136	3.42E+06	1.43E+04
Cs-137	1.52E+07	4.02E+04
Ba-139	5.37E+04	3.85E+02
Ba-140	8.20E+06	3.30E+04
La-140	3.29E+06	6.09E+03
La-141	2.91E+03	1.99E+01
La-142	5.84E+02	4.24E+00
Ce-141	3.04E+05	9.90E+02
Ce-143	5.55E+04	3.01E+02
Ce-144	3.72E+05	9.68E+02
Pr-143	8.68E+04	3.24E+02
Nd-147	2.88E+04	1.20E+02
Np-239	9.70E+05	4.85E+03
Pu-238	9.73E+02	2.45E+00
Pu-239	1.18E+02	2.97E-01
Pu-240	1.52E+02	3.82E-01
Pu-241	4.38E+04	1.11E+02
Am-241	2.35E+01	5.27E-02
Cm-242	4.17E+03	1.11E+01
Cm-244	2.25E+02	5.67E-01
Total	**1.90E+08**	**8.16E+05**

Table 12.3-10

Dose Rates in the Control Building EFU and Adjacent Rooms in Accident Conditions

Position	Room	Thickness (cm)	Dose rate (mSv/hr)
Inside, 30 cm below EFU	3406 (or 3407)	-	7.87E+01
Lower Slab	3302	50	1.81E-01
Upper Slab	Roof	40	1.40E-01
Lateral walls	Corridor	30	2.59E-01
	3404 (or 3403)	30	1.20E+00
	3407 (or 3406)	30	2.59E-01
	Outside	40	4.46E-02

Figure 12.3-1. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 11500 mm

Figure 12.3-2. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 6400 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-3. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 1000 mm

Figure 12.3-4. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-5. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 9060 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-6. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 13570 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-7. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 17500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-8. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 27000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-9. Nuclear Island Radiation Zones for Full Power and Shutdown Operation at Elevation - 34000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-10. Nuclear Island Radiation Zones for Full Power and Shutdown Operation Section A-A

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-11. Nuclear Island Radiation Zones for Full Power and Shutdown Operation Section B-B

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-12. Turbine Building Radiation Zones at Elevation - 1400 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-13. Turbine Building Radiation Zones at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-14. Turbine Building Radiation Zones at Elevation - 12000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-15. Turbine Building Radiation Zones at Elevation - 20000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-16. Turbine Building Radiation Zones at Elevation - 28000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-17. Turbine Building Radiation Zones at Elevations - 33000 and 38000 mm

Figure 12.3-18. Turbine Building Radiation Zones at Elevation Various

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-19. Radwaste Building Radiation Zones at Elevation - 9350 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-20. Radwaste Building Radiation Zones at Elevation - 2350 mm

Figure 12.3-21. Radwaste Building Radiation Zones at Elevation - 4650 mm

Figure 12.3-22. Radwaste Building Radiation Zones at Elevation - 10650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-22a. Radiation Zones in the Access Tunnel to the Electrical Building

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-22b. Radiation Zones in the Access Tunnel to the Electrical Building and Radwaste Building

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-23. Nuclear Island Area Radiation Monitors at Elevation - 11500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-24. Nuclear Island Area Radiation Monitors at Elevation - 6400 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-25. Nuclear Island Area Radiation Monitors at Elevation - 1000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-26. Nuclear Island Area Radiation Monitors at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-27. Nuclear Island Area Radiation Monitors at Elevation - 9060 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-28. Nuclear Island Area Radiation Monitors at Elevation - 13570 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-29. Nuclear Island Area Radiation Monitors at Elevation - 17500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-30. Nuclear Island Area Radiation Monitors at Elevation - 27000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-31. Nuclear Island Area Radiation Monitors at Elevation - 34000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-32. Turbine Building Area Radiation Monitors at Elevation - 1400 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-33. Turbine Building Area Radiation Monitors at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-34. Turbine Building Area Radiation Monitors at Elevation - 12000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-35. Turbine Building Area Radiation Monitors at Elevation - 20000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-36. Turbine Building Area Radiation Monitors at Elevation - 28000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-37. Turbine Building Area Radiation Monitors at Elevation - 33000 and 38000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-38. Turbine Building Area Radiation Monitors at Elevation Various

Figure 12.3-39. Radwaste Building Area Radiation Monitors at Elevation - 9350 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-40. Radwaste Building Area Radiation Monitors at Elevation - 2350 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-41. Radwaste Building Area Radiation Monitors at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-42. Radwaste Building Area Radiation Monitors at Elevation - 10650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-43. Nuclear Island Post Accident Radiation Zones at Elevation - 11500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-44. Nuclear Island Post Accident Radiation Zones at Elevation - 6400 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-45. Nuclear Island Post Accident Radiation Zones at Elevation - 1000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-46. Nuclear Island Post Accident Radiation Zones at Elevation - 4650 mm

Figure 12.3-47. Nuclear Island Post Accident Radiation Zones at Elevation - 9060 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-48. Nuclear Island Post Accident Radiation Zones at Elevation - 13570 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-49. Nuclear Island Post Accident Radiation Zones at Elevation - 17500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-50. Nuclear Island Post Accident Radiation Zones at Elevation - 27000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-51. Nuclear Island Post Accident Radiation Zones at Elevation - 34000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-52. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation – 11500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-53. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 6400 mm

Figure 12.3-54. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 1000 mm

Figure 12.3-55. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-56. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 9060 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-57. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 13570 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-58. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 17500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-59. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 27000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-60. Reactor, Fuel, & Control Buildings Personnel Egress Routes at Elevation - 34000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-61. Radwaste Building Personnel Egress Routes at Elevation - 9350 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-62. Radwaste Building Personnel Egress Routes at Elevation - 2350 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-63. Radwaste Building Personnel Egress Routes at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-64. Radwaste Building Personnel Egress Routes at Elevation - 10650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-65. Turbine Building Personnel Egress Routes at Elevation - 1400 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-66. Turbine Building Personnel Egress Routes at Elevation - 4650 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-67. Turbine Building Personnel Egress Routes at Elevation - 12000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-68. Turbine Building Personnel Egress Routes at Elevation - 20000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-69. Turbine Building Personnel Egress Routes at Elevation - 28000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-70. Turbine Building Personnel Egress Routes at Elevations - 33000 and 38000 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-71. Reactor Building Rooms Adjacent to the RWCU/SDC and FAPCS Demineralizers at Elevation - 11500 mm

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 12.3-72. Reactor Building RWCU/SDC and FAPCS Demineralizer Rooms and Adjacent Rooms at Elevation - 6400 mm

Figure 12.3-73. Reactor Building Rooms Adjacent to the RWCU/SDC and FAPCS Demineralizers at Elevation - 1000 mm

12.4 DOSE ASSESSMENT

This section discusses the radiological dose assessment (person-Sievert) for the ESBWR facility. Subsections 12.4.1 through 12.4.5 discuss the various factors involved with dose assessment within the different plant radiation areas. The resulting annual radiation dose estimate is summarized in Table 12.4-1.

Dose assessment is a significant element in determining that the facility design and methods of operation ensure occupational radiological exposures are as low as reasonably achievable (ALARA). Dose assessment depends on estimates of occupancy, dose rates in various occupied areas, frequency of operations, and number of personnel participating in reactor operations and surveillance, routine maintenance, waste processing, refueling, in-service inspection and special (unscheduled) maintenance. Facility personnel include station and utility employees, as well as contract workers. The occupations for these personnel include maintenance, operations, health physics, supervision and engineering.

These dose estimates are based on operation with a 24-month fuel cycle.

To estimate the total annual radiological dose to personnel, five facility areas are specified, and the annual person-Sievert dose for each area/task is evaluated separately. These designations are listed in Table 12.4-1. Where job functions and expected radiation levels are predictable or clearly defined, analytical methods were employed for the person-Sievert estimates. The resulting dose estimates for the ESBWR are contained in Table 12.4-1. Subsections 12.4.1 through 12.4.6 discuss the various factors involved in the evaluations and their related elements.

The analytical method used for the person-Sievert assessment is based on the product of the estimated occupancy time (i.e., person-hours per year) and the estimated average dose rate. Estimates of the occupancy time required for operations associated with equipment in facility radiation areas (e.g., maintenance, testing or surveillance time) are first determined. An applicable frequency of occurrence is also incorporated in the resulting annual occupancy time for each operation. Areas with insignificant radiation sources or occupancy are not included in the exposure estimate. Where radiation sources are present, a design maximum dose rate of 10 μSv/hr (1 mrem/hr) is assumed for Radiation Zone B areas and 50 μSv/hr (5 mrem/hr) for Radiation Zone C areas. Other estimated dose rates are based on either calculations or extrapolated from radiation levels reported at operating plants.

The primary purpose for dose assessment is to aid in reducing the radiological exposure associated with all phases of plant operation consistent with practical considerations for performing each task. To achieve this ALARA objective, the ESBWR design includes numerous significant design improvements to reduce occupational exposures from past BWR experience. For example, facility design improvements include the elimination of recirculation piping and valves, improved water chemistry and low cobalt alloys for the reactor cooling water boundary, a simplified Control Rod Drive (CRD) system, reduced equipment maintenance and improved access, increased use of live-load valve packing to mitigate stem leakage, overhaul handling and refueling devices, multiple main steam line plugs, improved MSIV seat grinding system and a reactor vessel stud tensioner. In assessing the collective occupational radiological dose, each potentially significant dose-causing activity was evaluated. The major activities are provided in

Table 12.4-1. Examples of significant design improvements that affect dose assessment in different plant areas are discussed below.

12.4.1 Drywell Dose

For the ESBWR drywell, design simplicity is the key to reduced occupational doses. Reactor systems are simpler with more passive safety features. The recirculation piping and pumps have been eliminated and a steel cylindrical shield has been provided around the reactor vessel to reduce drywell radiation fields.

Significant dose-causing activities identified for the drywell primarily involve maintenance tasks. The drywell is inaccessible during full power operations. Testing, maintenance, and surveillance activities will be performed after shutdown.

Projected ESBWR annual radiation exposures are shown in Table 12.4-1.

The major drywell activities identified in Table 12.4-1 are:

- Main Steam Isolation Valve (MSIV) Repair;
- Safety Relief Valve (SRV) Work;
- Fine Motion Control Rod Drive (FMCRD)/Automated Fixed In-Core Probe (AFIP) Work;
- Local Power Range Monitor (LPRM) Work;
- In-Service Inspection (ISI);
- Misc. Valves; and
- Misc. Instrumentation.

The Nuclear Boiler System (NBS) supplies steam to the main turbine. The MSIVs are located in the upper drywell area (4 valves) and in the reactor building outboard of the primary containment isolation wall (4 valves).

The ESBWR design incorporates three specific features to reduce occupational exposures in the MSIV maintenance areas:

- Improved MSIV leakage rate test procedures;
- Improved maintenance procedures with some procedures automated; and
- Reduced radiation fields, primarily due to the absence of the recirculation piping.

The MSIVs require periodic testing and maintenance to ensure proper action and leak tightness. Maintenance operations incorporate an improved seat grinding system and other special tools. Overall maintenance is reduced by use of the MSIV overhauling devices, use of main steamline plugs and the improved MSIV grinding system. Use of these automatic systems results in an additional overall reduction in maintenance times of approximately 50%. This, along with improved drywell access, significantly reduces the maintenance time necessary for MSIV repair.

Beginning in the early 1980's, the BWR Owner's Group began an extensive study of the causes for failure of MSIVs to meet the leakage rate limits and the extensive person-hours required to maintain these valves. As a result of these studies, the ESBWR uses the improved leakage rate test procedures and latest technology for valve maintenance to reduce the personal exposures. As a result of these aids, there is an estimated overall maintenance person-hour reduction to the value shown in Table 12.4-1.

Early studies on dose rates during MSIV maintenance showed increases in dose rate directly proportional to recirculation line activity. The ESBWR does not have these recirculation lines, thus removing the most significant shutdown source of radiation in the drywell. Additionally, the ESBWR is designed to limit the use of cobalt bearing materials on moving components that have historically been identified as major sources of in-water contamination. Overall, the feedwater line radiation is expected to be a factor of two lower than current BWRs.

The estimated dose rate for SRV work is shown in Table 12.4-1. In the ESBWR, the primary source of radiation exposure, the recirculation lines, has been removed. Overall, the reduction in drywell dose level for these types of maintenance is shown in Table 12.4-1. Overhead tracks and in-place removal equipment is provided in the ESBWR to reduce dose rates for maintenance.

A design improvement for the Neutron Monitoring System (NMS) involves replacing the conventional Traversing In-core Probe (TIP) system with fixed in-core detectors (AFIP) for calibrating the Local Power Range Monitors (LPRMs). Eliminating the complex drive and indexer mechanism with associated controls, which are required to insert and withdraw the TIPs from the core region, substantially improves operability, maintainability and reduces occupational radiological exposures.

LPRM design has been improved and currently each monitor lasts up to seven years. The estimated annual time and dose rates are shown in Table 12.4-1.

The drywell design includes many features to accommodate in-service inspection (ISI). Some of these include the use of stand-off mirror type insulation around the reactor vessel, use of remote-operated mechanical devices for inspection of the RPV body and nozzle welds, removable pipe insulation, and provisions for additional ISI operations laydown space. The use of natural circulation simplifies the design within the drywell by eliminating the recirculating loops, pumps, pipe supports, hangers, and shock suppressors. This also results in reduced ISI maintenance and personnel exposures.

ISI primarily consists of NDE examination of vessel and piping systems and welds. Dose resulting from ESBWR ISI is estimated based on the following:

- Elimination of recirculation lines and pumps with savings of annual time and dose.
  - Elimination of 14 nozzle inspections at 2 per year;
  - Elimination of shield penetration and shield plug removal; and
  - Reduction in drywell dose by 50% with the provision that the feedwater line dose is more than half the recirculation line dose and general drywell dose level and therefore, removal of recirculation line inspection may reduce the general drywell dose rate by 50%.
- Overall, it is estimated that by use of automated inspection, person-hour expended in ISI is reduced by a factor of two.

- The total vessel weld length inspection is reduced and the total weld inspection is reduced from the inspection required for conventional BWRs; and
- The ESBWR design incorporates specific access into inspection areas past insulation areas with additional reductions in annual time.

The overall person-hours and typical effective dose rate for the ESBWR are shown in Table 12.4-1.

Simplified systems result in a significant reduction in the total number of valves and instrumentation located in the drywell with an accompanying decrease in maintenance time. Valve design is also enhanced. For example, operation of the Gravity-Driven Cooling System (GDCS) requires reactor depressurization. This depressurization utilizes eight depressurization valves (DPVs) with four located on the steam pipes and four located on stub tubes off the RPV shared with the IC lines in the upper drywell. Squib valves were selected for the DPV function because they are simple, reliable, eliminate all leakage concerns and have low maintenance requirements. The DPVs are a non-leak, non-simmer, and non-maintenance design. They also simplify the Automatic Depressurization System (ADS) by reducing the total number of relief valve discharge lines and steam quenchers mounted in the suppression pool. The estimated annual time to perform maintenance on miscellaneous drywell valves is shown in Table 12.4-1. The estimated annual time to perform maintenance on miscellaneous drywell instrumentation is also shown in Table 12.4-1.

Improved materials of construction and design for the reactor core fuel reduces the probability of fuel failure, thus, fuel leakage is significantly limited. This results in a reduced source term throughout the facility's radiation areas. Dose-reduction improvements also include improved water chemistry and the use of special materials for the reactor cooling water boundary. The design employs materials and processes that prevent intergranular stress corrosion and corrosion cracking by adopting resistant materials, limiting sensitizing operations, incorporating heat treatment techniques and eliminating crevice conditions. In addition, a significant reduction in the drywell radiation levels result by restricting the cobalt content of selected vessel internals, using materials or cladding with corrosion resistance, and designing for erosive conditions.

Other drywell work includes items such as minor valve maintenance and instrumentation work. Overall reduction, in this effort, to the values shown in Table 12.4-1 is estimated due to the following ESBWR design improvements:

- Significant savings in total hours are estimated due to removal of the recirculation lines with miscellaneous recirculation line work such as line snubbers, fewer drywell cooling units, and less assembly/disassembly work on insulation.
- Overall reduction in the drywell radiation due to removal of the recirculation system results in the reduction of the overall upper drywell dose rate to the values shown in Table 12.4-1 because the components involved, such as drywell coolers, typically do not carry radioactive inventory.

12.4.2 Reactor Building Dose

The reactor building surrounds the Reinforced Concrete Containment Vessel (RCCV) and provides holdup and decay of radionuclides during an accident. It has been arranged to take advantage of the reduced quantity of equipment associated with the simpler reactor systems. The building arrangement features numerous dose-reducing benefits and improved equipment maintenance times. Equipment is more accessible which facilitates improved access control and maintenance. The building features enhanced accessibility on all floors. Equipment access is provided for all surveillance, maintenance and replacement activities with local service areas and laydown space for periodic inspections. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.

Projected ESBWR annual radiation exposures are shown in Table 12.4-1.

The major reactor building activities identified in Table 12.4-1 are:

- Reactor Pressure Vessel (RPV) Access/Reassembly
- Refueling
- Fine Motion Control Rod Drive (FMCRD) Work
- Control Rod Drive Hydraulic Control Unit (CRD HCU) Work
- Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System
- Instrumentation
- Other

Refueling operations involves all work with fuel and reactor components performed above the reactor and in pool area. Reactor vessel access and reassembly exposure times are reduced by use of a special stud tensioner for the RPV head bolts. The projected time to remove the RPV bolts with this equipment and reassemble, and average estimated exposures are shown in Table 12.4-1. Underwater transfer for the dryer and separator decreases exposures during refueling operations.

Refueling exposures are also decreased by use of an automated refueling platform. The improved fuel, inspection equipment and increased remote operations significantly reduce the refueling floor exposure. Also, fuel sipping is not required based upon the improved fuel design.

The RWCU/SDC purifies reactor coolant during normal operation and shutdown. Two 100% redundant trains are provided in the ESBWR design that uses state-of-the-art water treatment technology to significantly reduce the concentration of radionuclide material in the coolant. In addition, the material of construction for the system is stainless steel for those portions in contact with the reactor coolant. For system piping, smooth bends are used instead of welds and the nuclear grade pipes are electro-polished to reduce corrosion product buildup.

RWCU/SDC maintenance work consists of inspection for two pumps per year in each train. The ESBWR uses canned pumps in both trains with an estimated reduction in maintenance time to the values shown in Table 12.4-1. With improved water chemistry and overall reductions in reactor water concentrations due to the two percent cleanup system, the effective dose rate is estimated at the value shown in Table 12.4-1.

With significant reductions in instrumentation due to reduced emphasis on active safety systems in lieu of passive systems, and combining systems such as the FAPCS, or deleting systems such as the TIP system, instrumentation work is reduced. With these changes, combined with improvements in water chemistry systems, the anticipated ESBWR effective dose rate is as shown in Table 12.4-1.

Simplifying systems in the reactor building result in a significant reduction in the total number of valves and an accompanying decrease in maintenance time. This work includes all valve work, minor maintenance, and CRD hydraulic line work. Use of live-load valve packing to control stem leakage reduces maintenance and worker radiation exposure for valve repair. The major task in this area involves the Hydraulic Control Units. With the use of the FMCRD units, an additional reduction of maintenance is anticipated. In addition, the ESBWR reactor building has been designed to provide for ease of maintenance with overhead lifts, coordinated hatchways and ample space to maintain in-place equipment. In addition, most of the equipment in the reactor building is removable. Because of these factors, an additional reduction in work is anticipated, resulting in the final value shown in Table 12.4-1. Because of the improved water chemistry, the overall effective dose rate is anticipated at the reduced value shown in Table 12.4-1.

12.4.3 Fuel Building Dose

The Fuel building houses the Spent Fuel Pool and the FAPCS that purifies spent fuel pool water during normal operation and shutdown. Simplified systems result in a significant reduction in the total number of valves with an accompanying decrease in maintenance time.

ESBWR refueling is performed as described above and expected doses in the Fuel building during refueling are shown in Table 12.4-1.

12.4.4 Turbine Building Dose

The steam and power conversion system includes the turbine main steam system, the main turbine generator, main condenser, main condenser air removal system, turbine gland seal system, turbine bypass system, extraction steam system, condensate purification system, and the condensate and feedwater pumping and heating system. The heat rejected to the main condenser is removed by a circulating water system and discharged to the normal power heat sink.

Steam, generated in the reactor, is supplied to the high-pressure turbine and the second stage reheater of the steam moisture separators/reheaters. Steam leaving the high-pressure turbine passes through a combined moisture separator/reheater prior to entering the low pressure turbines. The moisture separator drains, steam reheater drains, and the drains from the two high pressure feedwater heaters are drained to the open feedwater heater which is combined with a feedwater storage tank. The reactor feedwater pumps take suction from the open feedwater heater storage tank. The low pressure feedwater heater drains are cascaded to the condenser.

Steam exhausted from the low-pressure turbines is condensed and deaerated in the condenser. The condensate pumps take suction from the condenser hotwell and deliver the condensate through filters and demineralizers, gland steam condenser, steam jet air ejector condenser, off-gas recombiner condensers, and through the low-pressure feedwater heaters to the open feedwater heater storage tank. The reactor feed pumps discharge through the high pressure feedwater heaters to the reactor.

Projected ESBWR annual radiation exposures are shown in Table 12.4-1.

The major turbine building activities identified in Table 12.4-1 are:

- Turbine Overhaul
- Valves/Pumps
- Condensate Treatment
- Other

With additional operational improvements in automating and a simpler overall system design, the expected overall turbine maintenance (overhaul) work is reduced to the value shown in Table 12.4-1.

The condensate system in the ESBWR uses hollow-fiber filled filters that require approximately half the maintenance of typical systems, resulting in an estimated annual maintenance time shown in Table 12.4-1. The condenser tube material (with compatible tubesheet material) is corrosion resistant (titanium or stainless steel) which reduces leakage of corrosion products into the Condensate and Feedwater System. Low pressure feedwater drains from the feedwater heaters are cascaded back to the condenser; thus, all corrosion products from these drains are filtered via condensate filter/demineralizers before returning to the RPV. The overall effective dose rate is estimated at the value shown in Table 12.4-1.

12.4.5 Radwaste Building Dose

Radwaste Building work consists of pump and valve maintenance, shipment handling, radwaste management and general clean up activity. Radwaste building doses result from routine surveillance, testing, and maintenance of the solid and liquid waste treatment equipment. The liquid treatment system collects liquid wastes from equipment drains, floor drains, filter backwashes and other sources within the facility. The solid treatment system processes resins, backwash slurries and sludge from the phase separator. It also processes dry active waste from the plant. Some examples of radwaste activities include movement of casks and liners, filter handling, resin moving and installation and removal of mobile radwaste processing skids. Both waste treatment systems are based on current mobile radwaste processing technology and avoid complex permanently installed components. All radwaste tankage and support systems are permanently installed. More of the radwaste operations involve remote handling than in a typical BWR. This, as well as improved maintenance procedures and a more flexible radwaste system and building design, leads to the estimated value shown in Table 12.4-1 for maintenance tasks in the Radwaste Building. The average dose rate shown in Table 12.4-1 is estimated for all operations.

12.4.6 Work at Power Doses

Routine work at power represents various tours of operators through the plant each shift, inspections and miscellaneous maintenance in radiation zones, as necessary. It covers all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment and repair. Overall, the ESBWR is designed using more automated and remote handling equipment. It is estimated there is a reduction in the total hours for work at power to the value shown in Table 12.4-1.

Exposure from these miscellaneous surveillance, testing and maintenance activities at power is due to N-16 as well as reactor coolant corrosion and fission products. Additional shielding is provided to reduce radiation levels in routinely occupied areas during power operation from N-16 sources. The ESBWR is expected to have lower general radiation levels as compared to the typical BWR due to more stringent water chemistry controls, a full cleanup condensate flow system, a 1% reactor water clean up program, titanium condenser tubes, and low cobalt usage. The overall estimated effective dose for work at power is shown in Table 12.4-1.

12.4.7 COL Information

None.

12.4.8 References

None.

Table 12.4-1

Projected ESBWR Annual Radiation Exposure

Facility Area/Task	Estimated Annual Time, person-hour	Estimated Average Dose Rate, µSv/hr (mrem/hr)	Projected Annual Collective Dose, person-Sv (person-rem)
Drywell			
MSIV Repair	720	40 (4.0)	0.029 (2.9)
SRV Work	240	75 (7.5)	0.018 (1.8)
FMCRD/AFIP	405	80 (8.0)	0.032 (3.2)
LPRM Work	78	500 (50.0)	0.039 (3.9)
ISI	720	55 (5.5)	0.040 (4.0)
Misc. Valves	600	40 (4.0)	0.024 (2.4)
Misc. Instrumentation	500	50 (5.0)	0.025 (2.5)
Reactor Building			
RPV Access/Reassembly	1,200	30 (3.0)	0.036 (3.6)
Refueling	250	5 (0.5)	0.001 (0.1)
CRD HCU	284	45 (4.5)	0.013 (1.3)
FMCRD	324	100 (10.0)	0.032 (3.2)
RWCU/SDC	200	40 (4.0)	0.008 (0.8)
Instrumentation	600	30 (3.0)	0.018 (1.8)
Other	3,400	18 (1.8)	0.061 (6.1)
Fuel Building			
Refueling	500	5 (0.5)	0.003 (0.3)
FAPCS	200	40 (4.0)	0.008 (0.8)
Turbine Building			
Turbine Overhaul	12,000	3 (0.3)	0.036 (3.6)
Valves/Pumps	910	39 (3.9)	0.035 (3.5)
Condensate Treatment	1,000	35 (3.5)	0.035 (3.5)
Other	6,000	1 (0.1)	0.006 (0.6)
Radwaste Building	1,000	25 (2.5)	0.025 (2.5)
Miscellaneous Work at Power	2,000	40 (4.0)	0.080 (8.0)
Total	33,131		0.604 (60.4 person-rem)

12.5 OPERATIONAL RADIATION PROTECTION PROGRAM

12.5.1 Objectives

The ESBWR design includes health physics facilities and features providing capabilities for administrative control of:

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

12.5.2 Equipment, Instrumentation, and Facilities

The health physics facilities are located in the Service building. Access to the radiologically controlled areas of the Reactor, Fuel, Turbine, and Radwaste buildings is normally through the entry/exit area of the health physics facilities of the Service building. Exit from the radiologically controlled areas is at the same location.

The health physics area contains the personnel contamination monitoring equipment, decontamination shower facilities, changing rooms and first-aid equipment. The changing rooms are provided with lockers, wash sinks, showers and toilet facilities.

Portable radiation survey instrumentation is stored at the access control and health physics room and at in-plant control points. This instrumentation allows plant personnel to perform radiation, contamination and neutron surveys, as needed, as well as to collect samples for airborne analysis. Shielded rooms are provided in the health physics area for radioactivity analysis and for calibration of survey instruments.

The non-portable airborne radiation monitoring equipment is described in Subsection 12.3.4. The COL applicant will provide a description of plant health physics equipment, instrumentation, and facilities (COL 12.5-1-A). The COL applicant will provide a description of the portable instruments that accurately measure radio-iodine concentrations in plant areas under accident conditions and of the training and procedures on the use of these instruments in compliance with Paragraph 50.34 (f) (xxvii) of 10 CFR 50 and NUREG-0737 Item III.D.3.3. (COL 12.5-2-A).

12.5.3 Operational Considerations

The COL applicant will provide a description of the operational radiation protection program (COL 12.5-3-A).

12.5.4 COL Information

12.5-1-A Equipment, Instrumentation, and Facilities

The COL applicant will provide a description of plant health physics equipment, instrumentation, and facilities (Subsection 12.5.2).

12.5-2-A Compliance with Paragraph 50.34 (f) (xxvii) of 10 CFR 50 and NUREG-0737 Item III.D.3.3

The COL applicant will provide a description of the portable instruments that accurately measure radio-iodine concentrations in plant areas under accident conditions and of the training and procedures on the use of these instruments (Subsection 12.5.2).

12.5-3-A Radiation Protection Program

The COL applicant will provide a description of the operational radiation protection program (Subsection 12.5.3).

12.5.5 References

12.5-1 USNRC, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.70, Revision 3, November 1978.

12.6 MINIMIZATION OF CONTAMINATION AND RADWASTE GENERATION

This section discusses how the ESBWR design procedures for operation will minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive waste, in compliance with 10 CFR 20.1406.

12.6.1 Minimization of Contamination to Facilitate Decommissioning

Examples of ESBWR design features that minimize contamination and facilitate decommissioning include the following:

- Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of crud traps;
- Provisions for design features to plant systems such as the Reactor Water Cleanup/Shutdown Cooling System, liquid and solid radwaste systems and the condensate demineralizer to minimize crud buildup;
- Provisions for draining, flushing, and decontaminating equipment and piping;
- Penetrations through outer walls of a building containing radiation sources are sealed to prevent miscellaneous leaks to the environment;
- Equipment drain sump vents are piped directly to the radwaste HVAC system to remove airborne contaminants evolved from discharges to the sump;
- Appropriately sloped floor drains are provided in areas where the potential for a spill exists to limit the extent of contamination. The floor drains are designed to be of monolithic construction to minimize possibility of liquid penetrating at embedment boundaries. No grout is used in the installation of the floor drains. Periodic visual inspections of the installation around the floor drains will be performed to ensure that no bypass exists in these floor drain areas;
- Provisions for decontaminable epoxy-type wall and floor coverings which provide smooth surfaces to ease decontamination;
- Equipment and floor drain sumps are stainless steel lined to reduce crud buildup and to provide surfaces easily decontaminated;
- For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination;
- The reactor building HVAC system is divided into two major components: the contaminated and clean areas. The clean area system conditions and circulates air through all the clean areas of the reactor building; the contaminated area system conditions and circulates air through the contaminated areas of the building;
- The Fuel and Auxiliary Pools Cooling System (FAPCS), equipped with two independent filter demineralizer units, is designed to reduce pool water radioactive contamination in the major pools in the ESBWR;

- The ESBWR is designed to limit the use of cobalt bearing materials on moving components that have historically been identified as major sources of in-water contamination;
- To facilitate decommissioning, the Reactor, Fuel, Turbine, and Radwaste Buildings are designed for large equipment removal, consisting of entry doors from the outside and numerous equipment hatches within the buildings;
- To facilitate decommissioning and ease of access, the mobile radwaste systems are skid-mounted and located in the radwaste building to allow truck access, and mobile system skid loading and unloading;
- For some piping, feed-throughs with short sections, the piping may be embedded in concrete as discussed in subsection 12.3.1.2.4. Minimization of short sections with embedded piping to the extent practicable facilitates the dismantlement of the systems and the decommissioning;
- In consideration of minor leaks over long periods of time, the liquid radwaste system (tanks, piping, etc.), Radwaste Building, and the radwaste tunnels are designed to conform to Regulatory Guide 1.143. The spent fuel pool has a leak detection system to monitor any leakage during plant operation, as discussed in Subsection 3.8.4.2.5. The concrete in the underground tunnels containing radwaste piping to and from the Radwaste Building is sealed for ease of decontamination during operation. The tunnels have floor drains to remove any fluid that potentially could leak from the piping. Plant procedures require periodic visual inspection of the radwaste piping in the tunnels; and
- There is no concrete block wall construction in ESBWR. Holes provided for removal of components are filled with interlocking metallic blocks filled with concrete for shielding purposes, as discussed in the last two paragraphs of Subsection 3.8.4. Therefore, there is no exposed porous concrete that could be contaminated, which provides for easier decommissioning.

12.6.2 Minimization of Radioactive Waste Generation

Examples of ESBWR design features that minimize the generation of radioactive waste include the following:

- The Liquid Waste Management System (LWMS) is divided into several subsystems, so that the liquid wastes from various sources can be segregated and processed separately, based on the most efficient process for each specific type of impurity and chemical content. This segregation allows for efficient processing and minimization of overall liquid waste.
- During liquid processing by the LWMS, radioactive contaminants are removed and the bulk of the liquid is purified and either returned to the condensate storage tank or discharged to the environment, minimizing overall liquid waste. The radioactivity removed from the liquid waste is concentrated in filter media ion exchange resins and concentrated waste. The filter sludge, ion exchange resins and concentrated waste are sent to the Solid Waste Management System (SWMS) for further processing.

- The SWMS is designed to segregate and package the wet and dry types of radioactive solid waste for off-site shipment and burial. This segregation allows for efficient processing and minimization of overall solid waste.
- For management of gaseous radioactive waste, the Offgas System (OGS) minimizes and controls the release of radioactive material into the atmosphere by delaying release of the offgas process stream initially containing radioactive isotopes of krypton, xenon, iodine, nitrogen, and oxygen.

The LWMS, OGS, and SWMS are discussed and described in more detail in Sections 11.2, 11.3, and 11.4, respectively.

Examples of ESBWR design features that minimize the generation of radioactive waste during decommissioning operations include the following:

- Reduction of cobalt content in structural and bearing materials;
- Minimization of crud buildup in drains by use of stainless steel linings, improving drainage, and facilitating flushing; and
- Easing surface decontamination by providing epoxy-type wall and floor coverings.

12.6.3 COL Information

None

12.6.4 References

None

12A. CALCULATION OF AIRBORNE RADIONUCLIDES

This appendix presents a simplified methodology to calculate the airborne concentrations of radionuclides in a compartment. This methodology is conservative in nature and assumes that diffusion and mixing in a compartment is basically instantaneous with respect to those mitigating mechanisms such as radioactive decay and other removal mechanisms. For each analyzed compartment, the following calculations are performed on an isotope-by-isotope basis to verify airborne concentrations are within the limits of 10 CFR 20.

- For the compartment, all sources of airborne radionuclides are identified such as:
  - Flow of contaminated air from other areas;
  - Gaseous releases from equipment in the compartment; and
  - Evolution of airborne sources from water leaking from equipment or sumps.
- Second, the primary sinks of airborne radionuclides are identified. This is primarily outflow from the compartment but may also take the form of condensation onto room coolers.
- Given the above information, the following equation calculates a conservative concentration:

$$C_i = \frac{1}{V}\sum_j \frac{S_{ij}}{\left(\lambda_i + \sum_k R_{ijk}\right)}$$

where:

C_i	=	concentration of the ith radionuclide in the room
V	=	volume of room
S_{ij}	=	the jth source (rate) of the ith radionuclide to the room (these sources are discussed below)
R_{ijk}	=	the kth removal constant for the jth source and the ith radionuclide as discussed below.
λ_i	=	radionuclide decay constant

12A.1 EVALUATION PARAMETERS

The following parameters require evaluation on a case-by-case basis dictated by the physical parameters and processes germane to the modeling process.

S_{ij} is defined as the source rate for radionuclide i into the compartment. Typically, these source rates take the form of:

- Inflow of contaminated air from an upstream compartment. Given the concentration of radionuclide i, C_i, in this air and a flow rate of “r”, the source rate then becomes $S_{ij} = r\ C_i$.

- Production of airborne radionuclides from equipment. This typically takes two forms: gaseous leakage and liquid leakage.

For gaseous leakage sources, the source rate is equal to the concentration of radionuclide i, C_i, and the leakage rate, 'r', or $S_{ij} = r\ C_i$.

For liquid sources, the source rate is similar but more complex. Given a liquid concentration ci and a leakage rate, 'r', the total release from the leak is r C_i. The fraction of this release that then becomes airborne is typically evaluated by a partition factor, P_f that may be conservatively estimated from:

Noble Gases

$P_f = 1$

All others

$$P_f = \frac{h_t - h_f}{h_s - h_f}$$

where:

h_t	=	saturated liquid enthalpy
h_f	=	saturated liquid enthalpy at one atmosphere = 180.17 Btu/lb
h_s	=	saturated vapor enthalpy at one atmosphere = 1150.5 Btu/lb

Therefore, the liquid release rate becomes, r C_i P_f.

R_{ijk} is defined as the removal rate constant and typically consists of:

- Exhaust rate from the compartment. This term considers not only the exhaust of any initially contaminated air but also any clean air that may be used to dilute the compartment air.
- In compartment filter systems. Such filter systems are treated by the equation:

  $R_{ijk} = (1-F_i) \cdot r_i$

  where:

r_i	=	filter system flow rate
F_i	=	filter efficiency for radionuclide i

- Other removal factors on a case-by-case basis that may be deemed reasonable and conservative.

12A.2 EXAMPLE CALCULATION

Values used below are examples only and should not be used in any actual evaluation. This example will look at I^{131} in a compartment of 283 $m^3 = V$. First, all primary sources of radionuclides need to be identified and categorized.

Flow into the compartment equals 424 m^3 per hour with the input I^{131} concentration equal to 7.4 x 10^{-6} Bq/ml (from upstream compartments) or 0.90 Bq/sec. No other sources of air either contaminated or clean air are assumed.

The compartment contains a pump carrying reactor coolant with a maximum specified leakage rate of 5.7x10^{-7} m^3 per minute at 288°C.

- Conservatively, it can be estimated based upon properties from steam tables that under these conditions 44% of the liquid will flash to steam and become airborne. The assumption of 44% flashing at 288°C is extremely conservative. See Reference 12A-1 for a discussion of fission product transport. Along with the flashing liquid, it is assumed that a proportional amount of I^{131} will become airborne; therefore, $P_f = 0.44$.
- Assuming iodine concentrations for reactor water of 5.3x10-4 MBq/gm of I131, it is calculated that the pump is providing a source of I131 to the air of 1.6x10-6 MBq/sec to the air. Water density assumed at 0.743 gm/cm3 based upon standard tables for water at 288°C.

Second, the sinks for airborne material need to be identified, which in this example include only exhaust that is categorized as flow out of the compartment at 150% per hour or 4.2 x 10^{-4} per second.

Therefore, for an equilibrium situation, the I^{131} airborne concentration from this liquid source would be calculated from the equation

$$A=1/V(S_1 / (\lambda + R_1) + S_2 / (\lambda + R_2))$$

where:

V	=	volume of compartment = 283 m^3
S_1	=	source rate in Bq per second = 1.6 Bq/sec from liquid
S_2	=	source rate from inflow = 0.9 Bq/sec
λ	=	isotopic decay constant in units of per second = 9.977 x 10^{-7}/sec
$R_1 = R_2$	=	removal rate constant per second (exfiltration) = 4.2 x 10^{-4} per second.
A =		2.117x10^{-11} MBq/ml of I^{131}.

12A.3 COL INFORMATION

None

12A.4 REFERENCES

12A-1 Paquette, et al, “Volatility of Fission Products During Reactor Accidents,” Journal of Nuclear Materials, Vol. 130 Pg. 129-138, 1985.