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CHAPTER 12.0

RADIATION PROTECTION

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12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

12.1.1 POLICY CONSIDERATIONS

The policy of Union Electric, through guidance from Regulatory Guides 1.8, 8.8, and 8.10 and 10 CFR Part 20, includes a commitment to ensure that occupational radiation exposures (ORE) are as low as reasonably achievable (ALARA). Union Electric will establish radiation protection practices for all applicable plant activities and will provide a qualified radiation protection organization to accomplish this goal. Utility management recognizes and will emphasize the importance of each individual's responsibilities to maintain ORE-ALARA.

Section 12.1.1 of the Site Addendum identifies who has the ultimate responsibility for establishing an ALARA philosophy for operating, maintaining, refueling, and testing. The operating organizational structure and responsibilities are discussed in Section 13.1.2. Section 12.1.1 of the Site Addendum also indicates the responsibility of key personnel in implementation of ORE-ALARA policies.

12.1.2 DESIGN CONSIDERATIONS

A major objective in the design of the Callaway power block has been to limit the potential radiation exposures of operating personnel. Union Electric has emphasized this objective to its lead architect engineer (Bechtel) from the outset of the project, through its reviews of project design criteria and frequent design reviews as the detailed design of the power block has evolved. Specific design considerations and the guidelines employed in developing the Callaway design are presented in Sections 12.3.1.1 and 12.3.3.3. Estimated occupational radiation doses are given in Section 12.4. To ensure that ORE are ALARA, significant elements of the design program have been implemented as described in the following sections.

12.1.2.1 Plant Design

The design engineers and first level supervisors assigned by Bechtel to the Callaway project have, in most cases, performed similar design work on other nuclear power plants. Through this experience, they have developed sensitivity to and knowledge of radiation protection aspects of design, which have been applied to Callaway. Bechtel design engineers are also made aware of other operating experience through NRC Licensee Event Reports, NRC Operating Experience Bulletin Information Reports, and Bechtel-generated problem alert reports.

The Bechtel designers have, where practical, followed the design guidelines of NRC Regulatory Guide 8.8. The designers have also followed recommendations of the NSSS supplier (Westinghouse). Many of these recommendations are available in documented form, in the Westinghouse information packages and in Reference 1. Other recommendations developed from discussions with Westinghouse. Westinghouse representatives met with Bechtel designers, as appropriate.

Design of NSSS equipment, within Westinghouse's scope of supply, to ALARA objectives is described in Reference 1.

In the early stages of the plant design, calculations were performed to quantify potential concentrations of radioactivity throughout process systems and buildings of the power block. These calculations made use of measured data from operating plants, and also followed the methodology of NRC Regulatory Guide 1.112. Systems designs and equipment specifications have been influenced by these assessments (see Section 12.1.2.5). Additionally, radiation dose rates have been estimated throughout the power block.

Specifications for equipment and/or systems to be purchased reflect the objective to keep ORE-ALARA. For example, the specifications for the solid radwaste system state that "special consideration shall be given to eliminate points where radioactive materials may tend to accumulate." The specifications also require provisions for remote flushing and rinsing of portions of equipment that contain radioactive material and state that "special design consideration shall be given to minimizing operator exposure to radiation during the maintenance of equipment."

12.1.2.2 Scale-Model Program

The use of scale models is an innovative aspect of the Callaway project that has proven effective for design and design review. Three classes of scale models have been used with respect to radiation protection: (1) preliminary design model, (2) study models, and (3) final design model.

The preliminary design model (3/8 inch = 1 foot) was, in effect, a three-dimensional layout tool. Model builders constructed the main elements of the buildings (walls, floors, columns, etc.) and set in place major equipment (tanks, pumps, motors, etc.). Plant design engineers then routed piping, electrical cable trays, and HVAC ducts, and located valves and valve actuators on the model. The three-dimensional aspects of the model, compared to conventional layout drawings, placed radiation protection, among other considerations, into sharp focus for both design engineers and reviewers and facilitated evaluation of design alternatives, such as choosing the best valve placement. Prior to completion of the preliminary design models, several reviews by utility and SNUPPS staff were held, as discussed further in Section 12.1.2.4. Upon completion of the preliminary design model, the approved layouts were committed to paper as design drawings.

Study models were less formal in concept than the preliminary design and final design models and were constructed to evaluate specific design features or alternatives. For example, a study model was constructed of portions of the auxiliary building in order to evaluate the arrangement of radioactive demineralizers. The results of the review of this model are discussed in Section 12.1.2.5.

The final design model (3/4 inch = 1 foot) was built by model makers from design drawings, but was constructed sufficiently early to permit design changes, e.g., to facilitate maintenance. The final design model comprises the reactor building, auxiliary building, control building, radwaste building, and turbine building. Additional design reviews by Bechtel and by SNUPPS were performed, using the final design model (see Sections 12.1.2.3 and 12.1.2.4). These reviews again focused on such factors as operability, maintainability, and radiation protection.

12.1.2.3 Second-Level Design Reviews by the Lead Architect-Engineer

The term second-level review pertains to reviews beyond that of first-level supervision.

On-project reviews are conducted by engineering and supervisory personnel from groups, other than the group that originated the design, and by higher level supervisory personnel. The reviews are generally interdisciplinary. For example, layout of shielding and valve operators in the radwaste building is performed by plant design personnel and is reviewed by mechanical/nuclear engineering personnel, as well as by the project engineers for plant design and systems. These reviews bring a broader base of experience to bear on all aspects of design, including radiation protection. Participants are professional engineers with from 5 to 20 years of nuclear power plant design experience.

Off-project reviews are performed, when requested by the project organization, by members of the Bechtel technical staff. There have been no off-project reviews specifically addressed to radiation protection. However, shielding calculations have been the responsibility of the chief nuclear engineer and his staff and, through this involvement in the design, the staff personnel have contributed to the ALARA review.

12.1.2.4 Design Reviews by SNUPPS

Utility reviews of all safety-related systems, structures, and equipment have been coordinated through the SNUPPS Technical Committee, which is composed of senior-level utility engineers, one from each SNUPPS utility. It is the responsibility of the Technical Committee to obtain comments from appropriate personnel within their companies and to bring those comments to meetings of the Committee, where decisions are made on the basis of discussion and eventually a vote. Throughout the design and construction phase of the SNUPPS project, the Technical Committee has met or had a telephone conference call on the average of once every week and a half, and total meeting days per year of the Technical Committee have averaged about 50. The SNUPPS technical director participates in Technical Committee meetings.

Assisting the Technical Committee are various plant review groups, which are ad hoc groups of utility personnel selected to review and recommend action on specific aspects of design. The total meeting days per year have averaged about 15. A SNUPPS staff member participates in each meeting.

Radiation protection has been an important aspect of the reviews by the Technical Committee and the plant review groups and has been the subject of numerous specific reviews, as discussed further in Section 12.1.2.5. At least six distinct reviews of the scale models have specifically included consideration of radiation protection during plant operation and maintenance. At these reviews, representatives of SNUPPS have included, in addition to Technical Committee members and SNUPPS staff, the health physics superintendant from Ginna Station, several licensed senior reactor operators, operations and maintenance supervisors from other nuclear power plants, and utility engineering personnel experienced in nuclear plant operation. Several design changes resulted from these reviews. Examples are described in Section 12.1.2.5.

During design and construction of the first SNUPPS unit, SNUPPS staff and qualified personnel from other SNUPPS utilities participated in a review of the effectiveness of design to the ALARA objectives. Periodically during construction, preoperational testing, and start-up of the SNUPPS units, qualified personnel from each SNUPPS utility will participate in ALARA reviews of their respective plants.

12.1.2.5 Examples of Radiation Protection Design Reviews

a. Radiation Zone Drawings

Every location within the power block has been assigned a radiation zone classification. The method of establishing radiation zone classifications has been as follows. Shortly after initiation of the SNUPPS project, Bechtel prepared shielding design criteria which were reviewed by a SNUPPS plant review group. Participants in that review were utility engineering personnel, including one person with an SRO license from Ginna Station. Subsequently, Bechtel prepared radiation zone drawings which define the zone classification of each location in the power block. These drawings were reviewed by the utilities and SNUPPS staff. Participants in those reviews included the Technical Committee, utility engineering personnel, utility operating and health physics personnel, and SNUPPS staff (technical director and civil/structural engineer).

b. Reactor Cavity

SNUPPS learned in the Spring of 1975 that, unless neutron shielding was provided for the reactor cavity, neutron dose rates in the containment would be 10 to 100 times too high to permit operator access to the containment for reasonable periods of time during full-power operation. This conclusion was based on measured dose rates in the Calvert Cliffs plant, which is designed (as is SNUPPS) for access to the outside of the reactor vessel for performance of inservice inspection. SNUPPS had Bechtel undertake a study of possible neutron shield configurations, the effect of the shield on subcompartment pressure and pressure loadings on the reactor vessel, and obtainable dose reduction factors. SNUPPS has had numerous design review meetings with Bechtel. Participants for SNUPPS have included: SNUPPS staff (executive director, technical director, and licensing manager), the Technical Committee, and utility engineering personnel. SNUPPS also contracted for an outside review by the NUS Corporation, which included independent estimates of neutron streaming and the effectiveness of neutron shielding materials. The design of the neutron shield in the containment is described in Sections 3.8.3.1.4 and 12.3.2.2.1. Subsequent to NRC approval of Callaway's revised GDC-4 submittal, a permanent cavity seal/neutron shield was installed to reduce neutron dose rates in containment.

c. Radioactive Demineralizers

As a result of a review of a study model of the auxiliary building, the following design changes were made:

1. A single wall between the corridor and the valve and piping compartments was designed to replace space-consuming overlapping (staggered for radiation protection) access walls. This allows more room in the corridor and more accessibility for maintenance within the compartments.

2. Vertical valve controls were designed to replace horizontal controls. This eliminates the need for 90-degree turns in the valve control fixtures and eliminates the access difficulty, which horizontal valve control rods pose as obstacles to maintenance.

3. A concrete shielding floor was provided above the valve and piping compartments to minimize exposure to the valve control operators.

d. Airborne Radioactivity in Containment

Airborne radioactivity (predominantly noble gases) has limited containment access in operating PWRs. Leakage of noble gases from the reactor coolant through the packing of the pressurizer spray valve has been determined to be a significant contributor to the gaseous activity in containment. To alleviate this situation, the following design provisions have been incorporated in the SNUPPS plants:

1. A packless, low-leakage, ball-type pressurizer spray valve.

2. Provisions for continuous stripping of noble gases from the reactor coolant (see Section 9.3.4).

3. Addition of a mini-purge system to permit purging of the containment during power operation, prior to operator access (see Section 9.4.6).

e. Steam Generator Maintenance

A final example of the results of scale-model reviews by SNUPPS is provision of permanent maintenance platforms below the steam generators to facilitate access and thereby reduce associated personnel radiation doses for maintenance operations, such as eddy-current inspection of tubes and sludge-lancing.

12.1.2.6 Decommissioning

The following features of the plant design will assist decommissioning crews to maintain ORE-ALARA during the eventual decommissioning of Callaway.

a. The building arrangements, compartmentation, corridors, doorways, and hatches provide the ability to remove most items of equipment intact or, alternatively, to isolate and entomb specific areas.

b. The design features to maintain ORE-ALARA throughout the plant operating life are also applicable to the eventual decommissioning of the plant. These features include equipment design for ease of accessibility and maintenance, floor and wall coatings, provisions for remote flushing of equipment, ability to use remote handling equipment, decontamination techniques, and component design features to minimize crud buildup.

c. Specifications and limitations on cobalt content in equipment components will serve to limit radiation doses from crud buildup during both operation and subsequent decommissioning.

d. The amount of potentially radioactive buried pipe is limited.

12.1.3 OPERATIONAL CONSIDERATIONS

Refer to Section 12.1.1.3 of the Site Addendum for operational considerations.

12.1.4 REFERENCES

1. "Design, Inspection, Operation, and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures As Low As Reasonably Achievable," WCAP - 8872, April 1977

12.2 RADIATION SOURCES

The sources of radiation that form the basis for shield design calculations and the sources of airborne radioactivity required for the design of personnel protective measures and for dose assessment are discussed and identified in this section.

12.2.1 CONTAINED SOURCES

The shielding design is based on full-power operation with 0.25 percent fuel cladding defects (Ref. 1, 2, 3, 4). The sources were obtained by multiplying the ANSI N237 fission product sources by two (Ref. 5). Sources in the primary coolant include fission products released from fuel clad defects and activation and corrosion products. The sources in the primary coolant are discussed in Section 11.1 and listed in Table 11.1-4. Throughout most of the primary coolant system, activation products, principally nitrogen-16 during reactor operation, are the primary radiation sources for shielding design. For all systems transporting radioactive materials, conservative allowance is made for transit decay, while, at the same time, providing for daughter product formation.

In this section, the design sources are presented by building location and system. General location of the equipment discussed in this section is shown in the general arrangement drawings provided in Section 1.2.

12.2.1.1 Containment

12.2.1.1.1 Reactor Core

The primary radiations within the containment during normal operation are neutrons and gamma rays emanating from the reactor core. Tables 12.2-1 and 12.2-2 list neutron and gamma multigroup fluxes at the core centerline location outside the reactor vessel. The tables are based on nuclear parameter values discussed in Chapter 4.0. Table 12.2-4 lists core gamma fluxes at the core centerline location outside the reactor vessel after shutdown, for shielding requirements during shutdown and inservice inspection.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the reactor coolant system are fission products released from fuel and activation and corrosion products which are circulated in the reactor coolant. These sources are listed in Table 11.1-4, and their bases are discussed in Section 11.1.

During operation, the activation product nitrogen-16 is the predominant activity in the reactor coolant pumps, steam generators, and reactor coolant piping. The contained source of radiation within the pressurizer is comprised of a liquid volume activity, a vapor volume activity, and a deposited activity. These activities are identified in Table 12.2-3.

12.2.1.1.3 Secondary Coolant Cycle

Under normal operating conditions, there is insignificant radioactive contamination present within the steam and power conversion system. It is possible to spread contamination to this cycle via steam generator tube leakage. Based on the primary-to-secondary leak rate given in Table 11.1A-1, the equilibrium secondary system activities are developed in Section ll.l and provided in Table 11.1-4. The condensate demineralizers and steam generator blowdown system further reduce the radioactivity level in the secondary cycle, as described in Section 11.1.

An evaluation of the secondary coolant activity in Tables 11.1-6 (Sheet 4) and 11.1-4 verifies that shielding is not required for the steam and power conversion system, with the exception of the components that could potentially concentrate the radioactivity. The condensate demineralizers and the steam generator blowdown demineralizers are the only components which could potentially concentrate the radioactivity. These are discussed in Sections 12.2.1.4 and 12.2.1.5, respectively.

12.2.1.1.4 Auxiliary Systems

Residual heat removal system - see Section 12.2.1.2.1.

Chemical and volume control system - see Section 12.2.1.2.2.

Steam generator blowdown system - see Section 12.2.1.5.

12.2.1.1.5 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant source of radiation in the containment after plant shutdown for refueling. A reactor operating time necessary to establish fission product buildup near equilibrium for the reactor at rated power is used in determining the source strength. Shielding requirements for spent fuel transfer are based on the fission product activity present 72 hours after shutdown to conservatively take credit for the time elapsed prior to the initiation of refueling operations. Source terms for spent fuel are listed in Table 12.2-4.

12.2.1.2 Auxiliary Building

12.2.1.2.1 Residual Heat Removal System

The pumps, heat exchangers, and associated piping of the residual heat removal (RHR) system contain radioactive materials. For plant shutdown, the RHR pumps and heat exchanger sources result from the radioactive isotopes carried in the reactor coolant, discussed in Section 12.2.1.1.2, considering 4 hours of decay following shutdown. The radiation source terms for the RHR system are listed in Table 12.2-5.

12.2.1.2.2 Chemical and Volume Control System

The CVCS source activity is the reactor coolant inventory which is provided in Table 11.1-4. More than 1 minute of N-16 coolant activity decay is provided before the letdown line exits the containment, and, therefore, is not significant in determining shielding requirements for the CVCS equipment outside the containment.

Major equipment items include the letdown and excess letdown heat exchangers, mixed bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The seal water subsystem for the reactor coolant pumps includes the injection and return filters and the seal water heat exchanger. The design activities of the CVCS components are listed in Table 12.2-6. Heat exchanger and piping activities are derived from primary coolant activities. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

12.2.1.2.3 Nuclear Sampling System

The major radiation sources in the nuclear sampling system originate from the RCS, RHR, and CVCS systems. The greatest radiation exposure would be to personnel taking the samples. To minimize this exposure, an integral shield has been incorporated into the sampling station design.

Plant procedures provide contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere during the recovery phase following plant transient or accident.

The sampling systems, which provide the capability to make required analyses are listed on Tables 9.3-3, 9.3-4, 9.3-5, and 9.3-6. Figures 9.3-2, 9.3-3, and 9.3-4 show the piping and instrumentation diagrams for the sampling systems. Process radioactivity monitors for the sampling systems are indicated in the tables and figures mentioned above. The area and airborne radioactivity monitors for worker protection are given in Section 12.3.4 and are shown in Figure 12.3-2.

12.2.1.3 <u>Fuel Building</u>

12.2.1.3.1 Spent Fuel Storage and Transfer

The predominant radioactivity sources in the spent fuel storage and transfer areas in the fuel building are the spent fuel assemblies. Spent fuel assembly sources are discussed in Section 12.2.1.1.5. For shielding design, the fuel storage pool assumptions are given in Section 9.1.2. The major radionuclide concentrations in the water are provided in Table 12.2-7.

The fuel transfer tube is completely shielded with permanent shielding to within radiation zone limits. No special access control, radiation monitoring, or posting is required. The expansion bellows for the fuel transfer tube are under water in the fuel transfer canal in the fuel building (see Figure 3.8-48). There is no bellows inspection room or opening. The fuel transfer tube is completely surrounded by concrete or water, with the exception of the seismic gaps, so that no personnel access is possible. The fuel transfer tube in the seismic gap between the containment wall and the internal containment structure and in the seismic gap between the containment wall and the fuel building is shielded, using permanently installed lead loaded silicone foam rubber, to meet the radiation zone limits (see Figure 12.3-2). Therefore, there is no unshielded portion of the fuel transfer tube.

Shielding during dry fuel storage, transfer, and transport operations is provided by the HI-STORM UMAX Multi-Purpose Canister (MPC), the HI-TRAC VW transfer cask, the Vertical Ventilated Module (VVM), and the VVM closure lid. In addition to this shielding, the HI-STORM UMAX Certificate of Compliance (CoC) requires a minimum decay time of 3 years for fuel assemblies selected for dry fuel storage.

12.2.1.3.2 Fuel Pool Cooling and Cleanup System

Sources in the fuel pool cooling and cleanup system (FPCCS) are the result of the transfer of radioactive isotopes from the reactor coolant into the fuel storage pool during refueling operations. The reactor coolant activities for fission, corrosion, and activation products (Table 11.1-4) are decayed for the amount of time required to remove the reactor vessel head following shutdown, are reduced by operation of the letdown system filters and demineralizers following shutdown, and are diluted by the total volumes of the water in the reactor vessel, refueling pool, and fuel storage pool. This activity then undergoes subsequent decay and accumulation on the FPCC filters and demineralizers.

12.2.1.4 Turbine Building

12.2.1.4.1 Main Steam Supply and Power Conversion Systems

Potential radioactivity in the main steam supply and power conversion systems is a result of steam generator tube leaks and fuel cladding defects, as discussed in Section 12.2.1.1.3.

This radioactivity is sufficiently low that no radiation shielding for equipment in the turbine building is required in order to meet the radiation zone requirements. The isotopic concentrations for a condensate demineralizer bed and other secondary system sources are listed in Table 12.2-8.

12.2.1.5 Radwaste Building

12.2.1.5.1 Boron Recycle and Liquid Radwaste Systems

The system sources are radioisotopes, including fission and activation products, present in the reactor coolant. The components of the systems contain varying degrees of activity, depending on the detailed system and equipment design.

The concentrations of radionuclides in the process fluids at various locations in the radwaste systems, such as pipes, tanks, filters and demineralizers are discussed in Section 11.1 and are listed in Tables 11.1-4 and 11.1-6. These nuclide concentrations for 0.25 percent failed fuel have been used in the final shielding design. Shielding for each component of the radwaste systems is based on maximum activity conditions, as given in Sections 11.1 and 11.2.

12.2.1.5.2 Gaseous Radwaste System

Radiation sources for each component of the waste gas system are based on operation under the conditions given in Sections 11.1 and 11.3. Tabulation of the activities is shown in Table 12.2-9.

12.2.1.5.3 Solid Radwaste System

Radiation sources for each component providing influent to the solid radwaste system are based on operation under the conditions given in Sections 11.1 and 11.4. Tabulation of the activities is shown in Table 11.4-3.

12.2.1.6 Sources Resulting from Design Basis Accidents

The radiation sources from design basis accidents include the design basis inventory of radioactive isotopes in the reactor coolant, plus postulated fission product releases from the fuel. Accident parameters and sources are discussed and evaluated in Chapters 11.0 and 15.0.

12.2.1.7 Stored Radioactivity

The principal sources of activity not stored inside the plant structures are the original steam generators, the original reactor vessel closure head, spent fuel assemblies stored within the Independent Spent Fuel Storage Installation (ISFSI), the Forced Helium Dehydration (FHD) skid stored within the ISFSI Support Building, the reactor makeup water storage tank (RMWST), the refueling water storage tank (RWST), the condensate storage tank (CST), and two radwaste discharge monitor tanks.

The original steam generators and original reactor vessel closure head are stored in the Old Steam Generator Storage Facility (OSGSF). As shown in Figure 12.2-1, the OSGF contains the following 3 rooms or areas: the Old Steam Generator (OSG) bay, the Old

Reactor Vessel Head (ORVH) room, and miscellaneous radioactive materials (RAM) storage room. Contaminated materials in the RAM storage room are limited to those not exceeding the threshold of Category 2 radioactive materials as defined in 10 CFR 37.

The walls and ceiling of the OSGSF are constructed of reinforced concrete with a minimum thickness of 2 feet. Shielding calculations that conservatively assume 100% of the activity in the OSGSF is from high energy Co-60 have been performed to confirm the external dose rate will not exceed 2 mrem/hr for the OSGSF. The OSGSF is expected to yield an external surface dose rate of 0.6 mrem/hr or less.

Personnel access to the OSGSF is controlled by two labyrinth entryways with locked steel doors. One entryway leads into the OSG bay, and one entryway leads into the RAM storage room. A locked access door is also provided between the OSG bay and ORVH room. Additionally, a lockable steel rollup door is also provided for exterior access to the RAM storage room. The OSG bay and ORVH room are treated as confined spaces due to emergency egress requirements associated with Building Officials and Code Administrators (BOCA) National Building Code, 1999. Personnel access to the facility is infrequent. The annual dose to an individual at the site boundary due to normal operation of the ISFSI is limited to the values specified in 10 CFR 72.104. The RMWST and CST are expected to contain concentrations of radionuclides which yield a surface dose rate of 0.5 mrem/hr or less. The RWST is expected to have a maximum contact dose of less than 10 mrem/hr when the water is returned from the refueling pool. This will be rapidly reduced by processing through the fuel storage pool purification filter and demineralizer. The radwaste discharge monitor tanks will also have a dose rate of less than 10 mrem/hr at contact. Tabulations of the activities within these tanks are provided in Table 11.1-6 (Sheets 1, 2, and 3).

All spent fuel is stored in the fuel storage pool until it is placed in either a spent fuel storage canister for storage in the ISFSI, or a spent fuel shipping cask for transport offsite. Storage space is allocated in the radwaste building for the storage of spent filter cartridges and dewatered spent resins, and chemical/hazardous wastes. Radioactive wastes stored inside the plant structures are shielded so that there is Zone A access outside the structure (see Figure 12.3-2). If it becomes necessary to temporarily store radioactive wastes outside the plant structures, adequate radiation protection measures will be taken by the radiation protection staff.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

This section identifies the models, parameters, and sources required to evaluate airborne radionuclide concentrations during plant operations in various plant radiation areas where personnel occupancies are expected.

An analysis of operating plant measurements of average airborne radionuclide concentrations (Ref. 6) and their respective MPCS for various stations in the auxiliary building (these stations include the waste handling area and the waste gas decay tank rooms, which are located in the radwaste building) indicates that the concentration to

MPC ratios in all stations is well within the limits of Note 5 of Appendix B to 10 CFR 20.1-20.601 to consider these airborne activities as insignificant. It is expected that the SNUPPS units would not have airborne radioactivity concentrations significantly greater than the operating plant data for corresponding locations. However, it is possible that, within these stations, there may be rooms where maximum airborne concentrations can occur due to localized leakage of radioactive fluids, but these rooms house equipment and components that handle highly radioactive fluids and consequently are D or E zones and, therefore, will be normally inaccessible. Also the ventilation systems in the auxiliary and radwaste buildings are designed in such a manner that airborne contamination from high radiation zones will not generally spread into low radiation zones, since the airflow is from regions of lower potential for contamination to those with higher potential for contamination. Consequently, negligible airborne radioactivity concentrations are expected in those areas of the auxiliary and radwaste buildings which are accessible. Airborne radioactivity concentrations in the turbine building are also expected to be negligible, since possible leaks into the turbine building are only from the secondary side, and, also, the turbine building ventilation exhaust is high (at least 90,000 cfm). For example, airborne concentrations are calculated to be 2.3×10^{-3} μCi/cc and 6.9×10^{-12} μCi/cc for I-131 and Xe-133, respectively in the turbine building.

Higher airborne concentrations can, however, occur in the containment, both during power operation and refueling--the former due to coolant leakage and the latter primarily due to the evaporation of the refueling pool. Likewise, airborne concentrations can also occur in the fuel building both during power operation and refueling due to the evaporation of the spent fuel pool. During power operation, the airborne radioactivity in the fuel building will be almost all due to tritium, since the continuous operation of the fuel pool cleanup system will effectively remove other isotopes from the pool. The assumptions and parameters required to evaluate the airborne radionuclide concentrations in the containment and fuel building both during power operation and refueling are listed in Table 12.2-11. The concentrations in these buildings are listed in Table 12.2-12. Even though some of these airborne concentrations may be high, limited occupancies in these areas ensures that the doses from airborne radioactivity to an individual will be a small fraction of the 10 CFR 20 limits for occupation exposures.

Airborne radioactivity is monitored inside the plant, as described in Section 12.3.4, and in process equipment and effluents, as described in Section 11.5.

12.2.2.1 <u>Model For Calculating Airborne Concentrations</u>

For those regions which are characterized by a constant leakrate of the radioactive source at constant source strength and a constant exhaust rate of the contaminant, the peak or equilibrium airborne concentration of the radioisotope in the regions can be calculated, using the following equation:

$$Ci(t) = \frac{(LR)_i Ai(PF)_i(1 - e^{-\lambda Ti^t})}{V\lambda_{Ti}} \tag{1}$$

where

$(LR)_i$	=	Leak or evaporation rate of the i^{th} radioisotope in gm/sec, in the applicable region

and

A_i	=	activity concentration of the i^{th} leaking or evaporating radioisotope in μCI/gm
$(PF)_i$	=	partition factor or the fraction of the leaking activity that is airborne for the i^{th} radioisotope
λ_{Ti}	=	Total removal rate constant for the i^{th} radioisotope in sec^{-1} from the applicable region
	=	$(\lambda di + \lambda e)$
		$\lambda di + \lambda e$ are the removal rate constants in sec^{-1} due to radioactive decay for the i^{th} radioisotope and the exhaust from the applicable region
t	=	time interval between the start of the leak and the time at which the concentration is evaluated in seconds
V	=	free volume of the region in which the leak occurs in cc
$C_i(t)$	=	airborne concentration of the i^{th} radioisotope at time t in mCi/cm3 in the applicable region

From the above equation, it is evident that the peak or equilibrium concentration, C_{Eqi} , of the i^{th} radioisotope in the applicable region will be given by the following expression:

$$C_{Eqi} = (LR)_i\, A_i\, (PF)_i\, /V\lambda_{Ti} \tag{2}$$

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

12.2.3 REFERENCES

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2. Reactor Physics Constants, Argonne National Laboratory, ANL-5800 (July, 1963).

3. H. Soodak, Reactor Handbook, Vol. III, Part A, Physics, second edition (1962).

4. D. A. Klopp, NAP - Multigroup Time-dependent Neutron Activation Prediction Code, IITRI-A6088-21 (January, 1966), conditions as given in Sections 11.1, 11.2, and 11.4.

5. ANSI N 237, "Source Term Specification," Final Draft, 1977.

6. NUREG/CR-0140, In-Plant Source Term Measurements at Fort Calhoun Station - Unit 1, Prepared for USNRC by EG & Idaho, Inc., July 1978.

7. NUREG-0017, Calculation of Releases of Radioactive Materials in Gaseous & Liquid Effluents from Pressurized Water Reactors, USNRC, April 1976.

TABLE 12.2-1 NEUTRON FLUXES ON INSIDE SURFACE OF THE PRIMARY SHIELD WALL AT THE CORE CENTERLINE (100% POWER)

Energy Group	Neutron Flux (neutrons/cm^2-sec)
ϕ_1(E > 1.0 Mev)	7.6×10^8
ϕ_1(5.53 Kev < E ≤ 1.0 Mev)	1.2×10^{10}
ϕ_3(0.625 ev ≤ E ≤ 5.53 Kev)	7.1×10^9
ϕ_4(E < 0.625 ev)	1.8×10^9

TABLE 12.2-2 GAMMA FLUXES ON INSIDE SURFACE OF THE PRIMARY SHIELD WALL AT THE CORE CENTERLINE (100% POWER)

Group	Flux (Mev/cm^2-sec)	Group Energy (Mev/γ)
1	3.7×10^9	7.5
2	3.3×10^9	4.0
3	1.7×10^9	2.5
4	1.0×10^9	0.8

TABLE 12.2-3 PRESSURIZER SHIELDING SOURCE TERMS

Liquid Volume

Isotope	Activity (µCi/gm)
N-16	1.8 (max)
Cr-5l	1.68E-03
Mn-54	3.10E-04
Fe-55	1.61E-03
Fe-59	9.30E-04
Co-58	1.53E-02
Co-60	2.02E-03
Br-83	2.53E-04
Br-84	3.08E-05
Br-85	3.39E-07
Rb-86	1.48E-04
Rb-88	1.32E-03
Sr-89	6.84E-04
Sr-90	2.11E-05
Sr-91	1.29E-04
Y-89M	6.16E-08
Y-90	1.35E-05
Y-91M	8.44E-05
Y-91	1.33E-04
Y-93	7.05E-06
Zr-95	5.72E-05
Nb-95	4.84E-05
Nb-95M	2.92E-05
Mu-99	7.34E-02
Tc-99	3.68E-09
Ru-103	4.14E-05
Ru-106	1.00E-05
Te-125M	2.75E-05
Te-127M	2.73E-04
Te-127	3.26E-04
Te-129M	1.27E-03
Te-129	8.23E-04
Te-131M	6.13E-04
Te-131	1.15E-04
Te-132	1.24E-02
I-129	8.55E-13

Liquid Volume

Isotope	Activity (µCi/gm)
I-130	5.17E-04
I-131	3.83E-01
I-132	1.71E-02
I-133	1.46E-01
I-134	9.09E-04
I-135	2.68E-02
Cs-134	5.24E-02
Cs-135	1.04E-10
Cs-136	2.10E-02
Cs-137	3.79E-02
Ba-137M	3.58E-02
Ba-140	3.55E-04
La-140	3.43E-04
Ce-141	1.32E-04
Ce-143	6.86E-05
Ce-144	6.86E-05
Pr-143	8.67E-05
Pr-144	6.86E-05

Steam Volume

Isotope	Activity (µCi/gm)
Kr-83M	9.74E-04
Kr-85M	1.14E-02
Kr-85	1.13E+00
Kr-87	1.92E-03
Kr-88	1.36E-02
Kr-89	6.89E-06
Xe-131M	1.31E-01
Xe-133M	1.35E-01
Xe-133	1.59E+01
Xe-135M	9.03E-05
Xe-135	6.79E-02
Xe-137	1.49E-05
Xe-138	2.69E-04

TABLE 12.2-4 SPENT FUEL SHUTDOWN SOURCES (FULL CORE)

Photon Energy	Time After Shutdown					
(Mev)	4 Hours	12 Hours	1 Day	1 Week	1 Month	3 Months
0.4	3.1×10^{11}	2.3×10^{11}	1.9×10^{11}	9.2×10^{10}	3.8×10^{10}	1.3×10^{10}
0.8	1.3×10^{12}	9.8×10^{11}	8.0×10^{11}	4.0×10^{11}	2.3×10^{11}	1.2×10^{11}
1.3	3.9×10^{11}	2.9×10^{11}	2.5×10^{11}	1.6×10^{11}	1.2×10^{11}	5.8×10^{10}
1.7	5.1×10^{11}	3.8×10^{11}	3.3×10^{11}	2.3×10^{11}	6.2×10^{10}	2.9×10^{9}
2.2	7.2×10^{10}	2.6×10^{10}	1.5×10^{10}	8.5×10^{9}	6.7×10^{9}	5.0×10^{9}
2.5	8.9×10^{10}	4.7×10^{10}	3.7×10^{10}	2.5×10^{10}	7.9×10^{9}	3.5×10^{8}
3.5	8.2×10^{9}	2.0×10^{9}	1.3×10^{9}	9.6×10^{8}	2.0×10^{8}	1.5×10^{7}

Values given in Mev/cc-sec.

TABLE 12.2-5 RADIATION SOURCES RESIDUAL HEAT REMOVAL SYSTEM

Isotope	Activities (µCi/gm)	Isotope	Activities (µCi/gm)
Cr-51	1.51E-03	Te-127M	2.24E-04
Mn-54	2.48E-04	Te-127	5.63E-04
Fe-55	1.28E-03	Te-129M	1.12E-03
Fe-59	7.98E-04	Te-129	7.66E-04
Co-58	1.28E-02	Te-131M	1.82E-03
Co-60	1.60E-03	Te-131	3.34E-04
Br-83	2.49E-03	Te-132	2.08E-02
Br-84	2.28E-05	I-129	2.63E-14
Br-85	negligible	I-130	2.75E-03
Kr-83M	1.34E-02	I-131	4.37E-01
Kr-85M	1.20E-01	I-132	6.36E-02
Kr-85	1.68E-02	I-133	5.46E-01
Kr-87	1.48E-02	I-134	3.17E-03
Kr-88	1.57E-01	I-135	2.06E-01
Kr-89	negligible	Xe-131M	3.94E-02
Rb-86	1.39E-04	Xe-133M	2.07E-01
Rb-88	1.75E-01	Xe-133	1.06E+01
Rb-89	5.28E-08	Xe-135M	3.56E-02
Sr-89	5.73E-04	Xe-135	5.42E-01
Sr-90	1.64E-05	Xe-137	negligible
Sr-91	8.01E-04	Xe-138	8.11E-07
Y-89M	5.15E-08	Cs-134	4.10E-02
Y-90	2.58E-06	Cs-135	4.48E-13
Y-91M	5.30E-04	Cs-136	2.11E-02
Y-91	1.06E-04	Cs-137	2.95E-02
Y-93	4.25E-05	Cs-138	4.47E-04
Zr-95	4.79E-05	Ba-137M	2.79E-02
Nb-95	4.00E-05	Ba-140	3.58E-04
Nb-95M	1.45E-06	La-140	2.54E-04
Mo-99	1.32E-01	Ce-141	1.14E-04
Tc-99	2.00E-10	Ce-143	6.03E-05
Ru-103	3.59E-05	Ce-144	5.41E-05
Ru-106	8.00E-06	Pr-143	8.18E-05
Te-125M	2.32E-05	Pr-144	5.41E-05

TABLE 12.2-5 (Sheet 2)

Isotope	Activities (µCi/gm)
I-131	4.37E-01
I-132	6.36E-02
I-133	5.46E-01
I-134	3.17E-03
I-135	2.06E-01
XE-131M	3.94E-02
XE-133M	2.07E-01
XE-133	1.06E+01
XE-135M	3.56E-02
XE-135	5.42E-01
XE-137	2.78E-21
XE-138	8.11E-07
CS-134	4.10E-02
CS-135	4.48E-13
CS-136	2.11E-02
CS-137	2.95E-02
CS-138	4.47E-04
BA-137M	2.79E-02
BA-140	3.58E-04
LA-140	2.54E-04
CE-141	1.14E-04
CE-143	6.03E-05
CE-144	5.41E-05
PR-143	8.18E-05
PR-144	5.41E-05

TABLE 12.2-6 CHEMICAL AND VOLUME CONTROL SYSTEM SOURCES

LETDOWN MIXED BED DEMINERALIZER

Isotope	Activity (μCi/cc)
Cr-51	3.30E+01
Mn-54	3.36E+01
Fe-55	2.23E+02
Fe-59	2.80E+01
Co-58	6.94E+02
Co-60	2.96E+02
Br-83	6.26E-01
Br-84	7.47E-02
Br-85	8.17E-04
Rb-86	1.15E+00
Rb-88	1.78E+00
Sr-89	2.30E+01
Sr-90	3.25E+00
Sr-91	3.42E-01
Y-89M	2.07E-03
Y-90	3.21E+00
Y-91M	2.25E-01
Y-91	5.13E+00
Y-93	1.88E-02
Zr-95	2.40E+00
Nb-95	3.44E+00
Nb-95M	2.40E+00
Mo-99	3.02E+02
Tc-99	9.73E-04
Ru-103	1.12E+00
Ru-106	1.14E+00
Te-125M	1.04E+00
Te-127M	1.72E+01
Te-127	1.74E+01
Te-129M	2.99E+01
Te-129	1.92E+01
Te-131M	1.95E+00
Te-131	3.64E-01
Te-132	5.49E+01
I-129	1.12E-06

TABLE 12.2-6 (Sheet 2)

Isotope	Activity (μCi/cc)
I-130	1.41E+00
I-131	2.83E+03
I-132	6.73E+01
I-133	4.30E+02
I-134	2.21E+00
I-135	6.92E+01
Cs-134	3.88E+03
Cs-135	1.73E-05
Cs-136	1.22E+02
Cs-137	3.26E+03
Ba-137M	3.08E+03
Ba-140	3.66E+00
La-140	3.98E+00
Ce-141	2.96E+00
Ce-143	7.16E-02
Ce-144	7.20E+00
Pr-143	9.56E-01
P4-144	7.20E+00

Reactor Coolant Filter

Gamma Energy (Mev/γ)	Specific Source Strength (Mev/cc-sec)
0.8	5.7×10^7
1.3	1.5×10^7

NOTE:

All other demineralizers and filters throughout the plant are shielded with these same source terms, since these are the most radioactive.

TABLE 12.2-6 (Sheet 3)

Regenerative Heat Exchanger Shell Side and Excess Letdown and Letdown Heat Exchanger Tube Side		Volume Control Tank Liquid Volume and Seal Water Heat Exchanger		Volume Control Tank Gaseous Volume	
Isotope	Activities (μCi/gm)	Isotope	Activities (μCi/cc)	Isotope	Activity (μCi/cc)
Cr-51	1.90E-03	Cr-51	1.73E-04	Kr-83M	2.69E-01
Mn-54	3.10E-04	Mn-54	2.82E-05	Kr-85M	2.01E+00
Fe-55	1.60E-03	Fe-55	1.46E-04	Kr-85	2.40E-01
Fe-59	1.00E-03	Fe-59	9.10E-05	Kr-87	6.11E-01
Co-58	1.60E-02	Co-58	1.46E-03	Kr-88	3.11E+00
Co-60	2.00E-03	Co-60	1.82E-04	Kr-89	3.19E-03
Br-83	1.00E-02	Br-83	1.00E-03	Xe-131M	5.65E-01
Br-84	5.42E-03	Br-84	5.42E-04	Xe-133M	2.96E+00
Br-85	6.25E-04	Br-85	6.25E-05	Xe-133	1.52E+02
Kr-83M	4.54E-02	Rb-86	8.08E-05	Xe-135M	3.88E-02
Kr-85M	2.25E-01	Rb-88	1.90E-01	Xe-135	7.16E+00
Kr-85	1.67E-02	Sr-89	6.65E-05	Xe-137	6.88E-03
Kr-87	1.32E-01	Sr-90	1.90E-06	Xe-138	1.17E-01
Kr-88	4.23E-01	Sr-91	1.24E-04		
Kr-89	1.13E-02	Y-90	2.28E-07		
Rb-86	1.77E-04	Y-91M	6.84E-05		
Rb-88	4.17E-01	Y-91	1.22E-05		
Sr-89	7.29E-04	Y-93	6.46E-06		
Sr-90	2.08E-05	Zr-95	5.46E-06		
Sr-91	1.35E-03	Nb-95	4.55E-06		
Y-90	2.50E-06	Mo-99	1.60E-02		
Y-91M	7.50E-04	Ru-103	4.10E-06		
Y-91	1.33E-04	Ru-106	9.10E-07		

TABLE 12.2-6 (Sheet 4)

Regenerative Heat Exchanger Shell Side and Excess Letdown and Letdown Heat Exchanger Tube Side		Volume Control Tank Liquid Volume and Seal Water Heat Exchanger		Volume Control Tank Gaseous Volume	
Isotope	Activities (μCi/gm)	Isotope	Activities (μCi/cc)	Isotope	Activity (μCi/cc)
Y-93	7.08E-05	Te-125M	2.64E-06	-----	-----
Zr-95	6.00E-05	Te-127M	2.55E-05		
Nb-95	5.00E-05	Te-127	7.74E-05		
Mo-99	1.75E-01	Te-129M	1.27E-04		
Ru-103	4.50E-05	Te-129	1.46E-04		
Ru-106	1.00E-05	Te-131M	2.28E-04		
Te-125M	2.90E-05	Te-131	1.00E-04		
Te-127M	2.80E-04	Te-132	2.46E-03		
Te-127	8.50E-04	I-130	4.37E-04		
Te-129M	1.40E-03	I-131	5.62E-02		
Te-129	1.60E-03	I-132	2.08E-02		
Te-131M	2.50E-03	I-133	7.92E-02		
Te-131	1.10E-03	I-134	9.79E-03		
Te-132	2.70E-02	I-135	3.96E-02		
I-130	4.37E-03	Cs-134	2.38E-02		
I-131	5.62E-01	Cs-136	1.24E-02		
I-132	2.08E-01	Cs-137	1.71E-02		
I-133	7.92E-01	Ba-137M	3.04E-03		
I-134	9.79E-02	Ba-140	4.18E-05		
I-135	3.96E-01	La-140	2.85E-05		
Xe-131M	3.98E-02	Ce-141	1.33E-05		
Xe-133M	2.17E-01	Ce-143	7.60E-06		
Xe-133	1.08E+01	Ce-144	6.27E-06		
Xe-135M	3.00E-02	Pr-143	9.50E-06		
Xe-135	6.44E-01	Pr-144	6.27E-06		
Xe-137	2.04E-02				

TABLE 12.2-6 (Sheet 5)

Regenerative Heat Exchanger Shell Side and Excess Letdown and Letdown Heat Exchanger Tube Side		Volume Control Tank Liquid Volume and Seal Water Heat Exchanger		Volume Control Tank Gaseous Volume	
Isotope	Activities (µCi/gm)	Isotope	Activities (µCi/cc)	Isotope	Activity (µCi/cc)
Xe-138	9.89E-02	-----	-----	-----	-----
Cs-134	5.21E-02				
Cs-136	2.71E-02				
Cs-137	3.75E-02				
Ba-137M	3.33E-02				
Ba-140	4.58E-04				
La-140	3.12E-04				
Ce-141	1.46E-04				
Ce-143	8.33E-05				
Ce-144	6.87E-05				
Pr-143	1.04E-04				
Pr-144	6.87E-05				

NOTE :

The moderating heat exchanger, chiller heat exchanger, and letdown reheat heat exchanger are the same as the combined liquid and gaseous sources for the volume control tank.

TABLE 12.2-6 (Sheet 6)

Boric Acid Tanks

Isotope	Activity (μCi/cc)
Cr-51	4.16E-06
Mn-54	8.56E-07
Fe-55	4.49E-06
Fe-59	2.41E-06
Co-58	4.08E-05
Co-60	5.63E-06
Br-83	2.54E-08
Br-84	1.92E-12
Br-85	negligible
Rb-86	8.40E-06
Rb-88	1.87E-09
Rb-89	5.94E-14
Sr-89	7.32E-06
Sr-90	5.90E-08
Sr-91	6.39E-08
Y-89M	6.58E-10
Y-90	5.00E-08
Y-91M	4.27E-08
Y-91	3.54E-07
Y-93	3.61E-09
Zr-95	1.52E-07
Nb-95	1.35E-07
Nb-95M	1.18E-07
Mo-99	8.72E-05
Tc-99	1.45E-11
Ru-103	1.06E-07
Ru-106	2.77E-08
Te-125M	7.23E-08
Te-127M	7.40E-07
Te-127	7.68E-07
Te-129M	3.21E-06
Te-129	2.05E-06
Te-131M	5.16E-07
Te-131	9.41E-08
Te-132	1.59E-05
I-129	5.08E-15
I-130	3.06E-07

TABLE 12.2-6 (Sheet 7)

Boric Acid Tanks

Isotope	Activity (μCi/cc)
I-131	7.46E-04
I-132	1.67E-05
I-133	1.09E-04
I-134	2.26E-09
I-135	1.06E-05
Cs-134	3.56E-03
Cs-135	3.71E-09
Cs-136	1.11E-03
Cs-137	2.59E-03
Cs-138	7.90E-08
Ba-137M	2.45E-03
Ba-140	7.64E-07
La-140	8.17E-07
Ce-141	3.31E-07
Ce-143	1.92E-08
Ce-144	1.90E-07
Pr-143	1.93E-07
Pr-144	1.90E-07

TABLE 12.2-7 FUEL STORAGE POOL WATER ACTIVITIES

Isotope*	Activities (µCi/cc)
Co-58	1.3E-04
Co-60	4.0E-04
Cs-134	2.0E-05
Cs-137	2.0E-04

*   Other isotopes will be present in much lower concentrations.

TABLE 12.2-8 SECONDARY SYSTEM ACTIVITIES

Condensate Demineralizer		High TDS Regenerant Collector Tank		Secondary Liquid Waste Drain Collector Tank		Secondary Liquid Waste Monitor Tank	
Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)
Cr-51	1.85E-06	Cr-51	2.04E-07	Cr-51	3.98E-11	Cr-51	2.38E-12
Mn-54	4.96E-07	Mn-54	6.43E-08	Mn-54	8.99E-12	Mn-54	6.43E-13
Fe-55	2.01E-06	Fe-55	2.61E-07	Fe-55	3.60E-11	Fe-55	2.61E-12
Fe-59	1.33E-06	Fe-59	1.73E-07	Fe-59	2.67E-11	Fe-59	1.72E-12
Co-58	1.86E-05	Co-58	2.41E-06	Co-58	3.57E-10	Co-58	2.41E-11
Co-60	2.27E-06	Co-60	2.94E-07	Co-60	4.05E-11	Co-60	2.94E-12
Br-83	4.96E-07	Br-83	6.43E-08	Br-83	8.83E-11	Br-83	5.83E-13
Br-84	2.94E-08	Br-84	3.81E-09	Br-84	5.23E-12	Br-84	7.03E-18
Br-85	4.27E-11	Br-85	5.54E-12	Br-85	7.62E-15	Br-85	6.31E-66
Rb-86	1.52E-07	Rb-86	1.96E-08	Rb-86	3.95E-12	Rb-86	1.94E-13
Rb-88	7.73E-08	Rb-88	1.00E-08	Rb-88	1.53E-11	Rb-88	3.08E-22
Sr-89	9.40E-07	Sr-89	1.22E-07	Sr-89	1.86E-11	Sr-89	1.21E-12
Sr-90	2.10E-08	Sr-90	2.73E-09	Sr-90	3.75E-13	Sr-90	2.73E-14
Sr-91	4.21E-08	Sr-91	5.46E-09	Sr-91	7.13E-12	Sr-91	3.01E-14
Y-89M	8.46E-11	Y-89M	1.10E-11	Y-89M	1.67E-15	Y-89M	1.09E-16
Y-90	1.73E-08	Y-90	2.25E-09	Y-90	1.36E-13	Y-90	2.29E-14
Y-91M	2.86E-08	Y-91M	3.70E-09	Y-91M	4.84E-12	Y-91M	2.01E-14
Y-91	1.50E-07	Y-91	1.95E-08	Y-91	2.90E-12	Y-91	1.94E-13
Y-93	2.19E-09	Y-93	2.84E-10	Y-93	3.68E-13	Y-93	1.61E-15
Zr-95	9.26E-08	Zr-95	1.20E-08	Zr-95	1.79E-12	Zr-95	1.20E-13
Nb-95	9.43E-08	Nb-95	1.22E-08	Nb-95	1.78E-12	Nb-95	1.23E-13
Nb-95M	6.62E-08	Nb-95M	8.59E-09	Nb-95M	2.61E-13	Nb-95M	8.80E-14
Nb-99	4.89E-05	Nb-99	6.35E-06	Nb-99	3.14E-09	Nb-99	5.82E-11
Tc-99	5.99E-12	Tc-99	7.77E-13	Tc-99	2.59E-17	Tc-99	7.96E-18
Ru-103	4.37E-08	Ru-103	5.66E-09	Ru-103	8.89E-13	Ru-103	5.63E-14
Ru-106	9.93E-09	Ru-106	1.29E-09	Ru-106	1.79E-13	Ru-106	1.29E-14
Te-125M	2.29E-08	Te-125M	2.96E-09	Te-125M	4.47E-13	Te-125M	2.95E-14
Te-127M	2.39E-07	Te-127M	3.10E-08	Te-127M	4.48E-12	Te-127M	3.10E-13
Te-127	2.61E-07	Te-127	3.39E-08	Te-127	8.06E-12	Te-127	3.25E-13

TABLE 12.2-8 (Sheet 2)

Condensate Demineralizer		High TDS Regenerant Collector Tank		Secondary Liquid Waste Drain Collector Tank		Secondary Liquid Waste Monitor Tank	
Isotope	Activity (µCi/cc)	Isotope	Activity (µCi/cc)	Isotope	Activity (µCi/cc)	Isotope	Activity (µCi/cc)
Te-129M	1.28E-06	Te-129M	1.66E-07	Te-129M	2.66E-11	Te-129M	1.65E-12
Te-129	8.26E-07	Te-129	1.07E-07	Te-129	1.83E-11	Te-129	1.06E-12
Te-131M	2.78E-07	Te-131M	3.61E-08	Te-131M	3.08E-11	Te-131M	2.98E-13
Te-131	5.17E-08	Te-131	6.70E-09	Te-131	5.77E-12	Te-131	5.44E-14
Te-132	6.79E-06	Te-132	8.81E-07	Te-132	3.86E-10	Te-132	8.18E-12
I-129	1.44E-15	I-129	1.87E-16	I-130	2.34E-10	I-129	1.88E-20
I-130	1.45E-06	I-130	1.88E-07	I-131	7.85E-08	I-130	1.18E-11
I-131	2.45E-03	I-131	3.18E-04	I-132	2.68E-09	I-131	3.09E-08
I-132	1.98E-05	I-132	2.57E-06	I-133	6.17E-08	I-132	2.82E-11
I-133	4.60E-04	I-133	5.97E-05	I-134	2.03E-10	I-133	4.52E-09
I-134	1.14E-06	I-134	1.48E-07	I-135	1.19E-08	I-134	1.91E-14
I-135	6.77E-05	I-135	8.77E-06	Cs-134	1.20E-09	I-135	3.71E-10
Cs-134	6.00E-05	Cs-134	7.78E-06	Cs-135	7.31E-18	Cs-134	7.77E-11
Cs-135	7.70E-13	Cs-135	9.99E-14	Cs-136	5.94E-10	Cs-135	6.54E-19
Cs-136	2.04E-05	Cs-136	2.65E-06	Cs-137	8.66E-10	Cs-136	2.60E-11
Cs-137	4.37E-05	Cs-137	5.67E-06	Ba-137M	8.21E-10	Cs-137	5.67E-11
Ba-137M	4.14E-05	Ba-137M	5.36E-06	Ba-140	9.02E-12	Ba-137M	5.36E-11
Ba-140	3.42E-07	Ba-140	4.44E-08	La-140	7.56E-12	Ba-140	4.36E-13
La-140	3.63E-07	La-140	4.71E-08	Ce-141	3.70E-12	La-140	4.67E-13
Ce-141	1.77E-07	Ce-141	2.29E-08	Ce-143	6.63E-13	Ce-141	2.27E-13
Ce-143	6.34E-09	Ce-143	8.23E-10	Ce-144	1.88E-12	Ce-143	6.91E-15
Ce-144	1.03E-07	Ce-144	1.34E-08	Pr-143	1.84E-12	Ce-144	1.34E-13
Pr-143	7.36E-08	Pr-143	9.55E-09	Pr-144	1.90E-12	Pr-143	9.40E-14
Pr-144	1.03E-07	Pr-144	1.34E-08			Pr-144	1.34E-13

TABLE 12.2-9 CONSERVATIVE BASIS ACCUMULATED RADIOACTIVITY IN THE GASEOUS WASTE PROCESSING SYSTEM AFTER FORTY YEARS OPERATION

Isotope	Activity at Plant Shutdown (curies)
Kr-85	15317.5
Kr-85m	10.9
Kr-87	1.05
Kr-88	10.6
Xe-131m	128.5
Xe-133	16536
Xe-133m	145
Xe-135	82
Xe-135m	0.035
Xe-138	0.04
I-131	0.207
I-132	0.00095
I-133	0.04
I-134	0.000187
I-135	0.0065

This table is based on 40 years' continuous operation with 0.25 percent fuel defect. Power is assumed to be 3,565 MWt. The data are based on a volume control tank purge rate of 0.7 scfm and 100 percent stripping efficiency.

TABLE 12.2-10 DELETED

Table 12.2-10, Sheets 1 through 3, Deleted.

TABLE 12.2-11 PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVE CONCENTRATIONS (REF. 7)

a. Leak Rates	Pounds/Day
Equivalent reactor coolant leak into containment during power for noble gases	5,300
Equivalent reactor coolant leak into containment for halogens	5.3
Equivalent reactor coolant leak into containment for other isotopes	240
b. Evaporation Rates	Pounds/Hr
From refueling pool into containment atmosphere (based on pool temperature of 120°F, and building air temperature of 70°F and 50 percent relative humidity and pool surface area of 1,500 square feet and 30 ft/minute flow parallel to the pool surface)	499.5
From fuel storage pool transfer canal and connecting slots into fuel building atmosphere during refueling (based on pool temperature of 137°F and building air temperature of 110°F and 95 percent relative humidity and with pool surface area of 2111.5 ft^2	1024
From fuel storage pool transfer canal and connecting slots into fuel building atmosphere during power (based on pool temperature of 95°F and building air temperature of 70°F and 50 percent RH and pool surface area of 2111.5 ft^2	409

c. Partition Factors	Halogens	Particulates	Tritium
Containment during power	1	.001	0.35
Fuel storage and refueling pool surfaces	1	Negligible	1

TABLE 12.2-11 (Sheet 2)

d.	Ventilation Rates	CFM	
	Containment during power	4,000	
	Containment during refueling	20,000	
	Fuel building during power	20,000	
	Fuel building during refueling	20,000	
e.	Volumes of the Regions	CF	
	Containment	2.5×10^6	
	Fuel building	8.2×10^5	
f.	Maximum Annual Individual Occupancy	Hrs/yr	
	Containment during power 5 hr/wk for 50 wks/year	250	
	Containment during fuel handling 10 hr/day for ~ 6.25 days when the refueling pool is full of water	62.5	
	Fuel building during power 5 hr/week for 50 weeks/year	250	
	Fuel building during refueling 10 hrs/day for ~ 10 days and 8 hrs/day for ~ 3 days	125	
g.	Miscellaneous Information		
	1. Failed fuel percentage for fission products	0.12	
	2. Reactor coolant specific activities	Table 11.1-1	
	3. Refueling and fuel storage pool concentration (μCi/gm). (These are the maximum concentrations during refueling)	I-131: 3.21×10^{-5}	I-133: 2.32×10^{-6}
	4. Fuel storage pool cleanup rate during power (gpm) (for conservatism, no cleanup of fuel storage pool or refueling pool is assumed during refueling)	300	

TABLE 12.2-11 (Sheet 3)

5. Decay of isotopes in the pools during refueling	Not included for conservatism
6. Duration of refueling pool evaporation (hrs)	150
7. Duration of fuel storage pool evaporation during refueling (hrs)	320
8. Duration of fuel storage pool evaporation during power (hours/year)	8440
9. Tritium release to environment via containment ventilation exhaust during refueling (Ci) (based on a total tritium release of 1000 Ci via gaseous effluents, durations of evaporation quoted in items 6, 7 and 8, and evaporation rates from pools given in Item b)	24
10. Tritium release to environment via fuel building ventilation exhaust during refueling (Ci) (same bases as given for Item 9)	104
11. Tritium release to environment via fuel building ventilation exhaust during power (Ci) (same bases as given for Item 9)	1094

TABLE 12.2-12 AIRBORNE RADIOACTIVITY CONCENTRATIONS (μCi/cc)

Nuclide	Containment (power)	Containment (refueling)	Fuel Building (power)	Fuel Building (refueling)
H-3	2.34E-7	4.7E-6	3.81E-06	9.55E-06
Cr-51	1.25E-12	negligible	negligible	negligible
Mn-54	2.07E-13	negligible	negligible	negligible
Fe-55	1.07E-12	negligible	negligible	negligible
Fe-59	6.63E-13	negligible	negligible	negligible
Co-58	1.06E-11	negligible	negligible	negligible
Co-60	1.33E-12	negligible	negligible	negligible
Br-83	1.76E-11	negligible	negligible	negligible
Br-84	2.61E-12	negligible	negligible	negligible
Kr-83m	6.60E-8	negligible	negligible	negligible
Kr-85m	6.02E-7	negligible	negligible	negligible
Kr-85	1.18E-7	negligible	negligible	negligible
Kr-87	1.39E-7	negligible	negligible	negligible
Kr-88	8.34E-7	negligible	negligible	negligible
Kr-89	5.83E-10	negligible	negligible	negligible
Rb-88	8.01E-7	negligible	negligible	negligible
Rb-89	5.63E-10	negligible	negligible	negligible
Sr-89	3.54E-12	negligible	negligible	negligible
Sr-91	2.49E-13	negligible	negligible	negligible
Mo-99	5.06E-11	negligible	negligible	negligible
Te-127m	1.86E-13	negligible	negligible	negligible
Te-127	4.01E-13	negligible	negligible	negligible
Te-129m	9.26E-13	negligible	negligible	negligible
Te-129	6.58E-13	negligible	negligible	negligible
Te-131m	1.34E-12	negligible	negligible	negligible
Te-131	2.69E-13	negligible	negligible	negligible
Te-132	1.65E-11	negligible	negligible	negligible
I-130	1.95E-11	negligible	negligible	negligible
I-131	3.82E-9	2.14E-10	negligible	7.6E-10
I-132	3.66E-10	negligible	negligible	negligible
I-133	4.14E-9	1.55E-11	negligible	5.5E-11
I-134	7.41E-11	negligible	negligible	negligible
I-135	1.34E-9	negligible	negligible	negligible
Xe-131m	2.75E-7	negligible	negligible	negligible
Xe-133m	1.35E-6	negligible	negligible	negligible
Xe-133	7.24E-5	negligible	negligible	negligible
Xe-135m	7.64E-9	negligible	negligible	negligible

TABLE 12.2-12 (Sheet 2)

Nuclide	Containment (power)	Containment (refueling)	Fuel Building (power)	Fuel Building (refueling)
Xe-135	2.55E-6	negligible	negligible	negligible
Xe-137	1.26E-9	negligible	negligible	negligible
Xe-138	2.22E-8	negligible	negligible	negligible
Cs-134	1.67E-11	negligible	negligible	negligible
Cs-136	8.48E-12	negligible	negligible	negligible
Cs-137	1.20E-11	negligible	negligible	negligible
Cs-138	2.07E-8	negligible	negligible	negligible
Ba-137m	1.14E-11	negligible	negligible	negligible
Ba-140	1.43E-13	negligible	negligible	negligible
La-140	1.07E-13	negligible	negligible	negligible

NOTES:

1. Iodine airborne concentrations during refueling are calculated very conservatively, assuming no purification by fuel pool cleanup system and no decay in the pool and a partition factor of 1 at the water-air interface at pool surface.
2. Continuous pool cleanup, decay in the pool, and lower evaporation rates are expected to reduce iodine air concentrations to negligible levels during power operation in the fuel building.
3. Xe-133 and I-131 air concentrations in the containment ventilation exhaust duct are expected to be $\leq 2 \times 10^{-2}$ and 1.6×10^{-6} μCi/cc, respectively during reactor head venting. However, the containment airborne concentrations for these isotopes are expected to be significantly less during head venting since the radioactivity is directly piped to the exhaust duct. The maximum value for Xe-133 is based on operating plant measurements normalized for a reactor system which has continuous stripping of the noble gases in the volume control tank. The maximum value for I-131 is based on operating plant measurements.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

In this section, specific design features for maintaining personnel exposures ALARA, commensurate with the guidance given in Regulatory Guide 8.8, are discussed.

12.3.1.1 Plant Design Description for as Low as is Reasonably Achievable (ALARA)

The equipment and plant design features employed to maintain radiation exposures as low as is reasonably achievable are based upon the design considerations of Section 12.1.2 and are outlined in this section for several general classes of equipment (Section 12.3.1.1.1) and several typical plant layout situations (Section 12.3.1.1.2).

12.3.1.1.1 Common Equipment and Component Designs for ALARA

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

FILTERS - Liquid systems containing radioactive filters are provided with a semi-remote filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the drumming station for packaging and shipment from the site for burial. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. Each filter vessel is located within individual shielded compartments, provided with a ventilation supply and return, external remotely operated (via reach rods) vent and drain valving, and compartment drainage capabilities. Ventilation return ducts are equipped with a removable access panel through which portable radiation monitoring instruments can be inserted to check radiation levels within the compartment. The filter handling system with semi-remote tools has also been designed to be simple with a minimum of components susceptible to malfunction. Care has been taken to ensure that adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste area.

DEMINERALIZERS - Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin storage tanks prior to solidification and the fresh resin can be loaded into the demineralizer remotely. Underdrains and downstream strainers are designed for full system pressure drop. The demineralizers and piping are designed with provisions for being flushed with demineralized water. Strainers are installed in the vent lines to prevent the entry of spent resin into the exhaust duct. Each demineralizer compartment is equipped with a removable shield plug in the ceiling, offset from the centerline of the demineralizer vessel. Removal of the shield plug provides access for insertion of portable radiation monitoring instruments to check radiation levels within the compartment.

EVAPORATORS - Evaporators are provided with chemical addition connections to allow chemicals to be used for descaling operations. Space is provided to allow the removal of heating tube bundles. The more radioactive components are separated by shield walls from those that are less radioactive. All instruments and controls are located on the accessible side of the shield wall. Valves in nonradioactive lines are located remote from the radioactive components.

PUMPS - Wherever practicable, pumps have mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal, if necessary. All pumps in the radioactive waste systems are provided with flanged connections for ease in removal. Piping or pump casing drain connections are provided for draining the pump for maintenance.

TANKS - Whenever practicable, tanks are provided with sloped, dished heads or conical bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to control any contamination within the plant structures.

HEAT EXCHANGERS - Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials with tube-to-tube sheet joints welded to minimize leakage. Impact baffles are provided, and tube side and shell side velocities are limited to minimize erosive effects. Flushing connections are provided.

INSTRUMENTS - Instrument devices are located in low radiation zones and away from radiation sources, whenever practicable. Primary instrument devices which, for functional reasons, are located in high radiation zones are designed for easy removal to a lower radiation zone for calibration. Transmitters and readout devices are located in low radiation zones, such as corridors and the control room, for servicing. Some instruments (such as thermocouples) in high radiation zones are provided in duplicate to reduce the required access and service time, since immediate repair will not be necessary due to the backup instrument.

Instrument and sensing line connections are located in such a way as to avoid corrosion product and radioactive gas buildup.

VALVES - To minimize personnel exposures from valve operations, motor-operated, diaphragm, or other remotely actuated valves are used to the maximum extent practicable.

Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual valves are used on frequently operated process lines, either valve stem extenders or shielding are provided so that personnel need not enter the radiation area for valve operation.

For infrequently operated equipment in Zone E, manual valves associated with safe operation, shutdown, and draining of the equipment may be provided with remote-manual operators or reach rods. For a definition of the radiation zones, see Figure 12.3-2. All other valve operations are performed with equipment in the shutdown mode. To the maximum practicable extent, simple straight reach rods have been used to allow the operators to retain the feel of whether the valves are tightly closed or not. Valves with reach rods are installed either with their stems horizontal, with the reach rods also horizontal but above the heads of personnel, to allow ready access or with both the stem and reach rod orientation vertical, again to permit easy access. For valves located in the radiation areas, provisions are made to drain the adjacent radioactive components when maintenance is required.

Wherever practicable, valves for clean, nonradioactive systems are separated from radioactive sources and are located in readily accessible areas.

Manually-operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown may be equipped with reach rods extending through or over the valve gallery wall. The valve gallery shield walls are designed for maximum expected filter activities.

For most larger valves (2 1/2 inch and larger) in lines carrying radioactive fluids, either a double set of packing with a lantern ring and leakoff connection or a five ring set of packing with a carbon spacer are provided. Full ported valves are used in systems expected to contain radioactive solids.

Valve designs with minimum internal crevices are used where crud trapping could become a problem, especially for piping carrying spent resin or evaporator bottoms.

PIPING - The piping in pipe chases is designed for the lifetime of the unit. There are no valves or instrumentation in the pipe chase. Wherever radioactive piping is routed through areas where routine maintenance is required, pipe chases are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection requirements. Wherever practicable, piping containing radioactive material is routed to minimize the radiation exposure to the unit personnel.

FLOOR DRAIN - Floor drains and properly sloped floors are provided for each room or cubicle having serviceable components containing radioactive liquids.

Local gas traps or porous seals are not used on radwaste floor drains. Gas traps are provided at the common sump or tank.

LIGHTING - Multiple electric lights are provided for each cell or room containing highly radioactive components so that the burnout of a single lamp will not require entry and immediate replacement of the defective lamp, since sufficient illumination will still be available. Normally, incandescent lights are provided which require less time for servicing, and hence the personnel exposure is reduced. The fluorescent lights which are used in some areas do not require frequent service, due to the increased life of the tubes. However, when the system in that room is secured and flushed out, the burned out lamps in the room can be replaced rapidly so as to minimize the exposure of the personnel.

HVAC - The HVAC system design provides for the rapid replacement of the filter elements and housings.

HYDROGEN RECOMBINERS - The more radioactive components are separated by a shield wall from those that are less radioactive. All instruments and controls are located on the accessible side of the shield wall. All valves in the radioactive lines are also located on the accessible side of the shield wall. Valves in the nonradioactive lines are located outside of the room.

SAMPLE STATIONS - Sample stations for routine sampling of process fluids are located in the accessible areas. Shielding is provided at the local sample stations, as required, to maintain radiation zoning in proximate areas and minimize personnel exposure during sampling. The counting room and laboratory facilities are described in Section 12.5.

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

12.3.1.1.2 Common Facility and Layout Designs for ALARA

This section describes the design features utilized for standard type plant processes and layout situations. These features are employed in conjunction with the general equipment designs described in Section 12.3.1.1.1 and include the features discussed in the following paragraphs.

VALVE GALLERIES - Valve galleries are provided with shielded entrances for personnel protection. The valve galleries are divided into subcompartments which service only two or three components and are further subdivided by stud walls so that the personnel are only exposed to the valves and piping associated with one component at any given location. Threshold berms and floor drains are provided to control radioactive leakage. To facilitate decontamination in the valve galleries, concrete floors are covered with a smooth surfaced coating which will permit easy decontamination.

PIPING - Pipes carrying radioactive materials are routed through controlled-access areas properly zoned for that level of activity. Radioactive piping runs are analyzed to determine the potential radioactivity level and area dose rate. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways are provided. Valves and instruments are not placed in the radioactive pipeways. Whenever practicable, equipment compartments are used as pipeways only for those pipes associated with equipment in that compartment.

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

Potentially radioactive piping is always located in appropriately zoned and restricted areas. Process piping is monitored to ensure that access is controlled to limit exposure (Section 12.5).

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Thermal expansion loops are raised rather than dropped, where possible. In radioactive systems, the use of nonremovable backing rings in the piping joints is prohibited to eliminate a potential crud trap for radioactive materials. Butt welds are used in lieu of socket welds for all resin slurry and evaporator bottoms piping. Piping carrying resin slurries or evaporator bottoms is run vertically as much as possible, and large radius bends are utilized instead of elbows.

Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

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

CONTAMINATION CONTROL - Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All welded piping systems are employed on contaminated systems, to the maximum extent practicable, to reduce system leakage and crud buildup at joints. The valves in radioactive systems 2-1/2 inch and larger are provided with leak-off connections piped directly to the collection system.

Decontamination of potentially contaminated areas within the plant is facilitated by the application of suitable smooth surfaced coatings to the concrete floors.

Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

In controlled access areas where contamination is expected, radiation monitoring equipment is provided (Section 12.3.4). Those systems which become highly radioactive, such as the radwaste slurry transport system, are provided with flush and drain connections. Certain systems have provisions for chemical and mechanical cleaning prior to maintenance.

A decontamination facility, used for decontamination of removable components, is located adjacent to the auxiliary building. The decontamination system, shown in Figure 12.3-4, consists of wash tanks, pumps, filters, spray booth, ultrasonic generator, turbulator, and associated piping.

EQUIPMENT LAYOUT - In those systems where process equipment is a major radiation source (such as fuel pool cleanup, radwaste, etc.), pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. Control panels are located in low radiation zones (Zones A or B).

Major components (such as tanks, demineralizers, and filters) in the radioactive systems are isolated in individual shielded compartments, insofar as practicable.

Provision is made on some major plant components for the removal of these components to lower radiation zones for maintenance.

Labyrinth entranceway shields or shielding doors are provided for each compartment from which radiation could stream to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as filters and demineralizers), completely enclosed shielded compartments with hatch openings or removable wall sections are used.

Equipment in the nonradioactive systems which requires maintenance is located outside the radiation areas.

Figure 12.3-1 (Sheets 1-5) provides typical layout arrangements for demineralizers, filters, spent resin storage tanks, waste gas compressors, hydrogen recombiners, and sample racks and their associated valve compartments or galleries.

Exposure from routine in-plant inspection is controlled by locating, whenever possible, inspection points in properly shielded low background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is provided. In high radiation areas where routine inspection is required, remote viewing devices are provided, as needed. When this is not practicable, written procedures which reduce radiation exposure by reducing the total time exposed to the radiation field are used, and access to the high radiation areas is under the direct supervision of the plants radiation protection personnel.

FIELD RUN PIPING - All radioactive process piping, large and small, has been run and shielded by the architect/engineer. Scale models have been used to ensure that any possible interferences are taken into account.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yards is regulated and controlled by radiation zoning and access control (Section 12.5.2). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding. All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal operation, shutdown, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Radiation zone categories employed and their descriptions are given in Figure 12.3-2 (Sheet 1), and the specific zoning for each plant area is shown in Figure 12.3-2 (Sheets 1-6). All frequently accessed areas, i.e., corridors, are shielded for Zone A or Zone B access.

The control of ingress or egress of plant operating personnel to controlled access areas and procedures employed to ensure that radiation levels and allowable working time are within the limits prescribed by 10 CFR 20 are discussed in Section 12.5.

Any area having a radiation level which could result in a dose equivalent in excess of 5 mrem in one hour at 30 centimeters from the source or from any surface that the radiation penetrates, is posted with signs bearing the radiation symbol and the words, "CAUTION, RADIATION AREA." Access alert barriers (e.g. signs, chain, rope, door, etc.) are provided for all radiation areas. Locations of these barriers are shown on Figure 12.3-2. Any areas having a radiation level which could result in a dose equivalent in any one hour in excess of 100 mrem at 30 centimeters from the source or from any surface that the radiation penetrates is posted with the radiation symbol and the words, "CAUTION (or DANGER), HIGH RADIATION AREA." High radiation areas are kept locked or barricaded if greater than 1,000 mrem/hr except during periods when access to the area is required in which case positive control is exercised over each individual entry. Any area in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at 1 meter from a radiation source or from any surface that the radiation penetrates is posted with the radiation symbol and the words, "GRAVE DANGER, VERY HIGH RADIATION AREA."

Radiological surveys shall be posted or available at the entrance to the RCA and may be posted at the entrance to a radiological posted area requiring a RWP for entry. Hot spot tags are used to identify the highest source of radiation within a designated radiation, high radiation or very high radiation area.

The locations of all radioactive equipment and the shield wall thicknesses are given on the general arrangement drawings in Section 1.2.

The normal radiation level in the counting room will be the natural background level. Fly ash was specifically excluded from all concrete used for the counting room.

The facilities and equipment used for handling and shielding radioactive equipment calibration sources are described in Section 12.5.

12.3.2 SHIELDING

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

12.3.2.1 Design Objectives

The basic objective of the plant radiation shielding is to reduce personnel and population exposures, in conjunction with a program of controlled personnel access to and occupancy of radiation areas, to levels that are ALARA within the dose regulations of 10 CFR 20. Shielding and equipment layout and design are considered in ensuring that exposures are kept ALARA during all anticipated personnel activities in all areas of the plant containing radioactive materials, in accordance with Regulatory Guide 8.8.

Two basic plant conditions are considered in the nuclear radiation shielding design: normal, full-power operation, and plant shutdown.

The shielding design objectives for the plant during normal operation, including anticipated operational occurrences, and for shutdown operations are:

a. To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and proximate site boundary occupants are ALARA and within the limits of 10 CFR 20.

b. To assure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

c. To reduce potential equipment neutron activation and mitigate the possibility of radiation damage to materials.

d. The control room will be sufficiently shielded such that the direct dose plus the inhalation dose (calculated in Chapter 15.0) will not exceed the limits of GDC-19 (see Section 6.4 for a more detailed discussion).

12.3.2.2 General Shielding Design

Shielding is provided, as necessary, to attenuate direct radiation and scattered radiation through walls and penetrations to less than the upper limit of the radiation zone for each area shown in Figure 12.3-2. The minimum shielding requirements for all plant areas are given on scaled layout drawings in Section 1.2. General locations of the plant areas and equipment discussed in this section are also shown in the general arrangement drawings of Section 1.2. Design criteria for penetrations comply with the intent of Regulatory Guide 8.8 and are discussed in Section 12.3.1.1.2.

The material used for most of the plant shielding is ordinary concrete with a minimum bulk density of 147 lb/ft^3. Whenever poured-in-place concrete has been replaced by concrete blocks or other material, design assures protection on an equivalent shielding basis, as determined by the characteristics of the material selected (Ref. 1). Compliance of concrete radiation shield design with Regulatory Guide 1.69 is discussed in Appendix 3A. Water is used as the primary shield material for areas above the spent fuel transfer and storage areas.

For design basis accidents, the reactor building reduces the plant radiation intensities from fission products inside the containment to acceptable emergency levels, as defined by GDC-19, for the control room (see Sections 12.3.2.2.6, 6.4, and 15.6.5).

12.3.2.2.1 Reactor Building Interior Shielding Design

During reactor operation, most areas inside the reactor building are Zone E and normally inaccessible.

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

The reactor cavity seal/neutron shield, at the RPV seal ledge, attenuates neutron streaming. This seal/shield has been designed to provide equivalent neutron attentuation to 12 inches of water as was previously provided by water bag shielding. MORSE analyses have been performed which predict that limited personnel access may be allowed at the operating floor during power operation. See the response to NRC Question 331.1 for a discussion of the Morse analyses.

Components of the letdown system are located in shielded areas which are normally Zone E restricted access areas. Shielding is provided for N-16 delay piping, the excess letdown heat exchanger, and the regenerative heat exchanger.

After shutdown, most of the containment is accessible for limited periods of time, and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1,000 mrem/hr, depending on the location inside the containment (excluding reactor cavity). These dose rates result from residual fission products, neutron-activated materials, and corrosion products in the reactor coolant system.

Spent fuel is the primary source of radiation during refueling due to its high buildup of fission product activity. However, radiation levels are limited in areas outside the refueling pool by the shielding effects of the thick structural walls of the refueling pool. The operators involved in the refueling operations are shielded from the spent fuel by the depth of water maintained above the fuel assemblies.

12.3.2.2.2 Auxiliary Building Shielding Design

During normal operation, the major components in the auxiliary building containing potentially high radioactivity are those in the chemical and volume control system. These include the letdown lines, the volume control tank, purification filters and demineralizers, and the charging pumps.

Shielding is provided for each piece of equipment consistent with the access and zoning requirements of adjacent areas (Figure 12.3-2).

Depending on the equipment in the compartments, the access varies from Zones B through E. Corridors are shielded to allow Zone B access, and operator areas for valve compartments are limited to Zone C access.

Removable sections of block shield walls and concrete plugs are utilized to replace worn-out equipment and spent filter cartridges, respectively. Partial shield walls are placed between equipment in compartments with more than one piece of equipment to permit maintenance access.

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

12.3.2.2.3 Fuel Building Shielding Design

Spent fuel is the primary source of radiation in the fuel building. Because of the extremely high activity of the fission products contained in the spent fuel elements and the proximity of Zone B and C areas, extensive shielding has been provided for areas surrounding the fuel storage pool, cask loading pit and the fuel transfer canal to ensure that radiation levels remain below zone levels specified for adjacent areas. Water provides the shielding above the spent fuel assemblies during fuel handling operations.

During spent fuel loading operations, the HI-TRAC VW transfer cask provides shielding through a design employing water jackets, steel barriers and lead shielding which minimizes personnel exposure. For a detailed discussion of shielding during ISFSI operations, refer to UMAX FSAR Section 5.

The fuel pool cooling and cleanup system (FPCCS) (Section 9.1.3) shielding is based on the maximum activity discussed in Section 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the FPCC system to be shielded includes the FPCC heat exchangers, pumps, piping, filters, and demineralizers.

12.3.2.2.4 Radwaste Building Shielding Design

Shielding is provided, as necessary, around the following equipment in the radwaste building to ensure that the radiation zone and access requirements are met for surrounding areas.

a. Liquid waste collection tanks and pumps

b. Liquid waste monitor tanks and pumps

c. Chemical drain tank and pump

d. Liquid waste boron recycle and secondary liquid waste evaporators

e. Solid radwaste solidification equipment

f. Solid radwaste drumming and storage areas

g. Evaporator bottoms tanks

h. Liquid radwaste piping

i. Radwaste filters and demineralizers

j. Spent resin storage tank and pump

k. Gaseous radwaste surge tank, recombiners, and compressors

l. Gas decay tanks

Shielding is based upon operation with maximum activity conditions, as discussed in Sections 11.1, 11.2, 11.3, and 11.4.

Depending on the equipment in the compartments, the access varies from Zones B through E. Corridors are shielded to allow Zone B access, and operator areas for valve compartments are limited to Zone C access.

Removable sections of block shield walls and concrete plugs are utilized to replace worn-out equipment and spent filter cartridges, respectively. Partial shield walls are placed between equipment in compartments with more than one piece of equipment to permit maintenance access.

12.3.2.2.5 Turbine Building Shielding Design

Radiation shielding is not required for any process equipment located in the turbine building. All areas in the turbine building are classified Zone A.

12.3.2.2.6 Control Room Shielding Design

The design basis LOCA dictates the shielding requirements for the control room. Shielding is provided to permit access and occupancy of the control room under LOCA conditions with radiation doses limited to 5 rem whole body from all contributing modes of exposure for the duration of the accident, in accordance with GDC-19. A complete discussion of control room habitability during a LOCA is provided in Section 6.4. Figure 12.3-3 provides an isometric view of the control room shielding.

12.3.2.2.7 Diesel Generator Building Shielding Design

There are no radiation sources in the diesel generator building. Therefore, no shielding is required within the building.

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for all plant buildings containing radiation sources so that radiation levels at the accessible outside surfaces of the buildings are maintained below Zone A levels. Plant yard areas which are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public. Access to outside storage tanks which have a contact dose rate greater than 0.5 mrem/hr is restricted by a fence located at a distance at which the dose rate is less than 0.5 mrem/hr.

12.3.2.2.9 Independent Spent Fuel Storage Installation (ISFSI) Shielding Design

The ISFSI is designed for interim dry storage of spent nuclear fuel. The shielding is sufficient to comply with the requirements of 10 CFR 72.104 and 10 CFR 72.106. The dose rate on the VVM closure lid is calculated to be 0.25 mrem/hr neutron and 0.69 mrem/hr gamma for a total dose rate of 0.94 mrem/hr. The dose rate on the outlet duct screen is calculated to be 0.80 mrem/hr neutron and 1.32 mrem/hr gamma for a total dose rate of 2.12 mrem/hr. The dose rate on the side of the HI-TRAC VW transfer cask is calculated to be 237 mrem/hr neutron and 1,611 mrem/hr gamma, for a total dose rate of 1,848 mrem/hr. The regulatory dose limit at the owner-controlled area boundary is 25 mrem annual whole body dose for Normal and Off-Normal conditions and 5 rem whole body for accident conditions.

For a detailed discussion of the VVM and HI-TRAC VW transfer cask shielding properties, refer to the HI-STORM UMAX FSAR Chapter 5.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in Section 12.2.1. The thickness of each shield wall surrounding the radioactive equipment is determined by approximating, as closely as possible, the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources, using data from the Table of Isotopes (Ref. 2).

The geometric model (Ref. 3-11), assumed for the shielding evaluation of pipe tanks, heat exchangers, filters, demineralizers, evaporators, and the containment is a finite cylindrical volume source. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source (Ref. 3-11). Typical computer codes that are used for shielding analysis are listed in Table 12.3-1 (Ref. 12-21).

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall. Therefore, the actual anticipated radiation levels in the greater region of each plant areas are less than this maximum dose rate and, therefore, less than the radiation zone upper limit.

Where shielded entryways to compartments containing high radiation sources are necessary, labyrinths or mazes are used. The mazes are constructed so that the scattered dose rate plus the transmitted dose rate through the shield wall from all contributing sources are below the upper limit of the radiation zone specified for each plant area.

12.3.3 VENTILATION

The plant heating, ventilating, and air-conditioning (HVAC) systems are designed to provide a suitable environment for personnel and equipment during normal operation and anticipated operational occurrences. Parts of the plant HVAC systems perform safety-related functions.

12.3.3.1 Design Objectives

The plant HVAC systems for normal plant operation and anticipated operational occurrences are designed to meet the requirements of 10 CFR 20, "Standards for Protection Against Radiation," and 10 CFR 50, "Licensing of Production and Utilization Facilities."

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

a. During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in the restricted areas of the plant are ALARA and within the limits specified in 10 CFR 20.

b. During normal operation and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary will be ALARA and within the limits specified in 10 CFR 20 and 10 CFR 50.

c. The plant siting dose guidelines of 10 CFR 100 will be satisfied, following those hypothetical accidents described in Chapter 15.

d. The dose to control room personnel shall not exceed the limits specified in GDC-19, following those hypothetical accidents described in Chapter 15.0 and Section 6.4.

12.3.3.3 Design Guidelines

In order to accomplish the design objectives, the following guidelines are followed, wherever practicable.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

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

b. Equipment vents and drains are piped directly to a collection device connected to the collection system, instead of allowing any contaminated fluid to flow across the floor to the floor drain.

c. All-welded piping systems are employed on contaminated systems, to the maximum extent practicable, to reduce system leakage. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.

d. Suitable coatings are applied to the concrete floors of potentially contaminated areas to facilitate decontamination.

e. To minimize the amount of airborne radioactivity as a result of valve leakage, all valves 2-1/2 inches and larger in the radioactive systems are provided with double sets of packing with lantern rings and with leak-off connections that are piped to drain headers. Diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated.

f. Contaminated equipment has design features that minimize the potential for airborne contamination during maintenance operations. These features include flush connections for draining and flushing the pump prior to maintenance and flush connections on piping systems that could become highly radioactive.

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

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

b. In building compartments with a potential for contamination, a greater volumetric flow is exhausted from the area that is supplied to the area to minimize the amount of uncontrolled exfiltration from the area.

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

d. The air cleaning system's design, maintenance, and testing criteria are discussed in detail in the response to Regulatory Guides 1.52 and 1.140 found in Section 9.4. An illustrative example of the air cleaning system design is given in Section 12.3.3.5.

e. Air being discharged from potentially contaminated areas is passed through HEPA filters and charcoal absorbers to remove particulates and halogens or means are provided to isolate these areas upon indication of contamination to prevent the discharge of contaminants to the environment.

f. Suitable containment isolation valves are installed in accordance with GDC-54 and 56, including valve controls, to assure that the containment integrity is maintained.

g. Redundant seismic Category I systems and/or components are provided for portions of the ventilation system that serve areas required for safe shutdown of the reactor plant. Included herein are the plant control room and selected engineered safety feature equipment rooms.

h. Atmospheric tanks which contain radioactive materials are vented to the respective building ventilation system.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment

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

b. Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts, to the extent practicable.

c. Ventilating air is recirculated in the clean areas only.

d. Access and service of ventilation systems in potentially radioactive areas is provided by component location to minimize operator exposure during maintenance, inspection, and testing as follows:

    1. The outside air supply units and building exhaust system components are enclosed in ventilation equipment rooms. These equipment rooms are located in radiation Zone B and are accessible to the operators. Work space will be provided around each unit for anticipated maintenance, testing, and inspection.

    2. Local cooling equipment servicing the normal building requirements will be located in areas of low contamination potential (radiation Zones A or B) (refer to Figure 12.3-2).

12.3.3.4 <u>Design Description</u>

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

a. Containment building (see Section 9.4.6)

b. Auxiliary building (see Section 9.4.3)

c. Fuel building (see Section 9.4.2)

d. Radwaste building (see Section 9.4.5)

e. Turbine building (see Section 9.4.4)

f. Portions of the access control area (see Section 9.4.1.2).

Although the control room is considered to be a nonradioactive area, radiation protection is provided to assure habitability (see Section 6.4).

Other structures (e.g., pump intake structures, administrative building, etc.) contain no potential source of radioactivity and are not addressed in this chapter.

12.3.3.5 Air Cleaning System Design

The guidance and recommendations of Regulatory Guides 1.52 and 1.140 concerning maintenance and in-place testing provisions for atmospheric cleanup systems, air filtration, and adsorption units have been used as a reference in the design of the various ventilation systems. The extent to which Regulatory Guide 1.52 and 1.140 have been followed is discussed in Section 9.4.

Provisions specifically included to minimize personnel exposures and to facilitate maintenance or in-place testing operations are as follows:

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

b. Active elements of the atmospheric cleanup systems are designed to permit ready removal.

c. Access to active elements is direct from working platforms to simplify element handling. Ample space is provided on the platforms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling facilities and during any in-place testing operation.

d. No filter bank is more than three filter units high, where each filter unit is 2 feet by 2 feet. The access to the level or platform at which the filter is serviced is by stairs or elevators.

e. The clear space for doors is a minimum of 3 feet by 7 feet.

f. The filters are designed with replaceable 2 feet by 2 feet units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with bulkhead type doors that are closed against compression seals.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitoring

The area radiation monitoring system (ARMS) is provided to supplement the personnel and area radiation survey provisions of the plant radiation protection program described in Section 12.5 and to ensure compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CRF 50, 10 CFR 70, and Regulatory Guides 8.2, 8.8, and 8.12.

12.3.4.1.1 Design Bases

The principal objectives and criteria of the ARMS are provided below.

12.3.4.1.1.1 Safety Design Bases

The area radiation monitoring system has no function related to the safe shutdown of the plant or the capability to mitigate the consequences of accidents that could result in offsite exposures comparable to the guideline exposure of 10 CFR 100 and, therefore, has no safety design bases. See Appendix 7A for a discussion of Regulatory Guide 1.97.

12.3.4.1.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - The ARMS functions continuously to immediately alert plant personnel entering or working in nonradiation or low-radiation areas of increasing or abnormally high radiation levels which, if unnoticed, could possibly result in inadvertent overexposures.

POWER GENERATION DESIGN BASIS TWO - The ARMS serves to inform the control room operator of the occurrence and approximate location of an abnormal radiation increase in nonradiation or low-radiation areas.

POWER GENERATION DESIGN BASIS THREE - The ARMS complies with the requirements of 10 CFR 50, Appendix A, General Design Criterion 63 for monitoring fuel and waste storage and handling areas.

POWER GENERATION DESIGN BASIS FOUR - Certain monitors located near the spent fuel pool and new fuel storage vault act as criticality alarm monitors and conform to the requirements of 10 CFR 70, Regulatory Guides 8.5 and 8.12 and Standards ANSI/ANS-8.3-1979 and USAS N2.3-1967.

12.3.4.1.1.3 Codes and Standards

Codes and standards applicable to the area radiation monitors are indicated in Table 3.2-1.

12.3.4.1.2 System Description

12.3.4.1.2.1 General Description

The ARMS consists of five-decade range GM tube detectors located throughout the plant to warn personnel of abnormal gamma radiation levels. The detector signals are transmitted to the control room over individual cables. The displays, both local and in the control room, are five-decade logarithmic ratemeters. The alarms are both audible and visual, and are located in the control room and near the local detectors.

12.3.4.1.2.2 Criteria for Area Monitor Selection

The following design criteria are applicable to the area radiation monitoring system.

RANGE - The ARMS has a five-decade range from 10^{-1} to 10^{+4} mrem/hr, except for the reactor coolant sample room radiation monitor which has a range from 10 to 10^{5} mrem/hr. The ranges are made sufficiently wide to measure the radiation levels expected in the areas concerned. The system continues to read upscale if exposed to radiation levels above the maximum range.

SENSITIVITY - Gamma sensitive to photon energies of 100 keV and above.

RESPONSE - In any range, the readout indicates at least 90 percent of its end point reading within 5 seconds after a step change in radiation level at the detector.

ENERGY DEPENDENCE - The dose rate (mrem/hr) readout is within ±20 percent of the actual dose rate in each detected area from photon energies between 100 keV and 2.5 meV.

DIRECTIONAL DEPENDENCE - The dose rate readout does not vary more than 10 percent when exposed to a single point radiation source of approximately 1.0 MeV from any point in the frontal hemisphere of a horizontal plane.

ENVIRONMENTAL DEPENDENCE - The system meets the above requirements for all variations of temperature, pressure, and relative humidity within each area monitored, as listed below:

For instruments located outside the containment:

Temperature,°F	60 to 120
Humidity, %RH	5 to 95
Pressure	Atmospheric

For instruments located inside the containment:

Temperature,°F	50 to 150
Humidity, %RH	5 to 100
Pressure	± 2 psig

EXPOSURE LIFE - Each monitor located inside the containment maintains its characteristics up to an integrated dose of 10^7 rads. Each monitor located outside the containment maintains its characteristics up to an integrated dose of 10^6 rads.

12.3.4.1.2.3 Alarms

Each monitor channel is provided with a three-level alarm system. One alarm setpoint is below the background counting rate and serves as a circuit failure alarm. The other two-alarm setpoints provide sequential alarms on increasing radiation levels. Loss of power will cause an alarm on all three-alarm circuits. The alarms must be manually reset and can be reset only after the alarm condition is corrected.

12.3.4.1.2.4 Check Sources

Each monitor is provided with a check source, operated from the control room, which simulates a radiation level in the area for operational and gross calibration checks.

12.3.4.1.2.5 Power Supplies

The power supplies for all of the monitors are given in Table 12.3-4.

12.3.4.1.2.6 Calibration and Maintenance

The area radiation monitors are calibrated by the manufacturer, using a Cs-137 source. The manufacturer's calibration standards are traceable to The National Bureau of Standards primary calibration standard sources and are accurate to at least 5 percent.

The source-detector geometry during this primary calibration is identical to the source-detector geometry in field calibrations. A secondary standard counted in reproducible geometry during the primary calibration is supplied with each monitoring system.

Channel checks and source checks are performed at regular intervals to ensure proper monitor function. The monitors are recalibrated at regular intervals, and following repairs or modifications, using the secondary radionuclide standard.

12.3.4.1.2.7 Sensitivities

Each area radiation monitor is able to detect radiation levels as low as 0.1 mrem/hr, except for post-accident sample room radiation monitor which detects radiation levels as low as 10 mrem/hr.

12.3.4.1.2.8 Criteria for Location of Area Monitors

Generally, area radiation monitors are provided in areas to which personnel normally have access and for which there is a potential for personnel unknowingly to receive high radiation doses (e.g., in excess of 10 CFR 20 limits) in a short period of time because of system failure or improper personnel action.

Any plant area which meets one or more of the following criteria is monitored:

a. Zone A areas which, during normal plant operations, including refueling, could exceed the radiation limit of 0.5 mrem/hr upon system failure or personnel error or which would be continuously occupied following an accident requiring plant shutdown.

b. Zone B areas where personnel could otherwise unknowingly receive high levels of radiation exposure due to system failure or personnel error.

c. Areas in which quantities of special nuclear material in excess of 10 CFR 70.24 quantities are received, stored, or handled, e.g. spent fuel pool and new fuel storage vault. In the case of new fuel, two detectors are provided at the operating floor of the fuel building to serve as criticality monitors. In the case of spent fuel, two detectors are provided on opposite walls above the spent fuel pool to serve as criticality monitors. In both cases, the monitors will provide a distinct audible and visual personnel alarm at no more than 15 mrem/hr to signal the need for personnel evacuation. These monitors are provided in accordance with 10 CFR 70.24 requirements and will detect the criticality described in 10 CFR 70.24(a)(1).

d. For areas in which fuel is stored or radioactive waste systems and handling equipment are located, area radiation monitors are provided to detect conditions that might result in loss of residual heat removal capability and excessive radiation levels and to alert the operators to initiate appropriate safety action in accordance with GDC-63 of 10 CFR 50, Appendix A.

The location of each area radiation detector is indicated on the radiation zoning and access control drawings, Figure 12.3-2, and are listed in Table 12.3-2. Consistent with the above criteria, the following general areas are monitored:

a. Main control room

b. Radwaste building corridors

c. Auxiliary building corridors

d. Fuel storage and handling area

e. Radwaste pipe tunnel

f. Railway access (fuel building)

g. Hot machine and hot instrument shops

h. Containment

i. Radwaste solidification area

j. Sampling rooms and laboratory

12.3.4.1.2.9 Setpoints

The bases for the area radiation monitor setpoints are determined from the need to alert operators to abnormal radiation levels in the area. The setpoints are sufficiently above the normal radiation levels in the measured areas to avoid spurious alarms.

The normal setpoints for the individual area radiation monitors are provided in Table 12.3-2. To support temporary plant conditions or evolutions, the normal setpoint levels may be increased. At these increased levels, access to the area will be controlled in accordance with the Radiation Protection Program based on actual radiation levels. Area radiation monitor setpoints are administratively controlled when not at the normal setpoints.

12.3.4.1.2.10 Safety Evaluation

The ARMS is designed to operate unattended for extended periods of time, detecting and measuring ambient gamma radiation. Ambient radiation dose rate at the detector is indicated locally at the detector and remotely in the main control room. These monitors cause an audible and visual alarm at the detector and in the main control room if the radiation levels exceed preset limits. All components are solid state, and the system is designed for high reliability.

The system is not essential for safe shutdown of the plant, and it serves no active emergency function during operation. The system serves to warn plant personnel of high radiation levels in various plant areas. All monitors are independent, and failure of one unit has no effect on any other.

12.3.4.2 Airborne Radioactivity Monitoring

Monitoring for the presence of airborne radioactivity inside the plant is necessary for the protection of plant personnel, in compliance with Regulatory Guide 8.2 and within the limits established by 10 CFR 20.1-20.601.

The airborne radioactivity monitors provide information necessary to ensure that gaseous, particulate, and iodine radioactivity do not exceed 10 MPC hours in areas occupied by the station personnel.

The systems consist of permanently installed, continuous monitoring devices together with a program and provisions for specific sample collections and laboratory analyses.

12.3.4.2.1 Design Bases

The principal objectives and criteria of the airborne radiological monitoring systems (AiRMS) are provided below.

12.3.4.2.1.1 Safety Design Bases

SAFETY DESIGN BASES - There are no safety design bases for the monitoring of airborne radioactivity for inplant personnel protection. The control room ventilation monitors, the containment atmosphere monitors, the containment purge monitors, and the fuel building exhaust monitors are required to automatically initiate operation of engineered safety features systems in the event that airborne radioactivity in excess of the allowable limits exists. Additional design bases are given in the following sections:

a. Containment purge isolation system, Sections 6.2.4, 7.3.2, 9.4, and 11.5.

b. Fuel building ventilation isolation, Sections 7.3.3, 9.4.2, and 11.5.

c. Control room intake isolation, Sections 6.4.1, 7.3.4, 9.4.1, and 11.5.

These radioactivity monitors are protection system elements and are designed in accordance with IEEE Standard 279 due to safety design bases of 6.2.4, 6.4.1, 7.3, 9.4, and 11.5.

The safety evaluation of these systems is discussed in Section 7.3.

These monitors also serve for inplant worker protection, and this function is discussed at length in this section.

12.3.4.2.1.2 Power Generation Design Bases

POWER GENERATION DESIGN BASIS ONE - The airborne radioactivity monitors operate continuously to detect airborne particulates, iodine, and/or noble gases in the air upstream of all filters in the containment, auxiliary building, fuel building, radwaste building, waste gas decay tank rooms, access control area, and control room for the protection of the workers.

POWER GENERATION DESIGN BASIS TWO - The airborne radioactivity monitors are designed to detect 10 MPC-hours or better in any compartment or room served by the monitoring system.

POWER GENERATION DESIGN BASIS THREE - The containment atmosphere monitors are designed to detect leakage of radioactivity from the reactor coolant system into the containment atmosphere. This function is described in greater detail in Section 5.2.5. The containment purge monitors will serve as a backup to the containment atmosphere monitors while the purge is in operation.

12.3.4.2.1.3 Codes and Standards

Codes and standards applicable to the airborne radioactivity monitors are indicated in Table 3.2-1. The monitors listed in Section 12.3.4.2.1.1 have additional codes and standards applied due to their safety-related functions discussed in other sections, as noted above.

12.3.4.2.2 System Description

12.3.4.2.2.1 General Description

12.3.4.2.2.1.1 Data Collection

The AiRMS consist of particulate, iodine, and noble gas monitors with the attendant controls, alarms, pumps, valves, and indicators required to meet the design objectives in Section 12.3.4.2.1. Each monitor consists of the detector assembly and a local microprocessor. The local microprocessor processes the detector assembly signal in digital form, computes average radioactivity levels, stores data, performs alarm or control functions, and transmits the digital signal to the control room microprocessor. Signal transmission is accomplished via two two-wire daisy-chain loops. Each loop allows data transmission in either direction, ensuring that a single fault in the loop will not prevent the control room microprocessor from receiving the data.

The local microprocessors for monitors which perform safety functions (control room ventilation, fuel building ventilation, containment atmosphere, and containment purge monitors, refer to Section 11.5) are wired directly to individual indicators located on the seismic Category I AiRMS cabinets in the control room. The input from the safety-related channels to the daisy-chain loop is an isolated signal to ensure that the safety-related signals will not be affected by signals or conditions existing in the nonsafety portion of the system.

The control room microprocessor provides controls and indication for the AiRMS. Indication is via a Visual Display located in the control room. The signals from each monitor may also be recorded on a system printer. The safety-related monitors are also recorded on analog strip chart recorders.

12.3.4.2.2.1.2 Selection Criteria for Airborne Monitors

12.3.4.2.2.1.2.1 Introduction

The type of fixed instrumentation used for monitoring airborne radioactivity is offline. The offline system extracts a sample from the process stream and transports that sample to the radiation monitoring system which contains the specified equipment to detect particulates, halogens, and/or noble gases.

12.3.4.2.2.1.2.2 Sampling Criteria

The sampling system for the particulate/halogen/noble gas monitors is designed and installed in accordance with ANSI N13.1-1969 guide to sampling of airborne radioactive materials. Systems whose sensitivity is dependent upon sample flow employ isokinetic nozzles and suitable control of the flow rate.

12.3.4.2.2.1.2.3 Detection Criteria

Since both radioactive particulates and radioactive noble gases are beta emitters, beta-sensitive scintillation detectors are used to sense radioactivity to minimize the effects due to background radiation and, consequently, obtain a lower minimum detectable concentration.

Where spectrometric analysis is required (such as in iodine monitoring) an NaI (Tl) gamma scintillation detector assembly is employed.

12.3.4.2.2.1.2.4 Instrumentation Criteria

Instrumentation necessary to indicate, alarm, and perform control functions is provided to complete the monitoring system. Since radioactive concentrations may vary substantially, wide-range instruments are utilized. All airborne radiation monitors include provisions for obtaining a gas sample for laboratory isotopic analysis. The particulate and charcoal filters can readily be removed for laboratory analyses.

The airborne particulate monitors each consist of a fixed filter upon which radioactive particulate matter is deposited by means of a positive displacement pump that draws a continuous sample, using an isokinetic nozzle from the ventilation exhaust duct for the particular area. The fixed filter is located in front of a beta scintillation detector coupled to a photomultiplier tube which responds to the scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal.

Each airborne iodine monitor consists of a charcoal cartridge upon which iodine is adsorbed. The air sample is prefiltered to remove particulates. The charcoal cartridge is located in front of a gamma scintillation detector coupled to a photomultiplier tube.

Each airborne noble gas monitor consists of a fixed volume sample chamber through which prefiltered sample air is passed. A beta scintillation detector is located within the sample chamber to detect the activity level of the air sample.

All of the detectors and sample chambers are enclosed in heavily shielded lead pigs. Two motor-operated valves, operated locally, are provided to permit air purging of the sample chamber to facilitate background activity checks.

The sensitivities and alarm setpoints are given in Table 12.3-3. The high-alarm points are based on the most restrictive isotopes which are expected to be present.

12.3.4.2.2.1.3 Criteria for Airborne Radioactivity Monitor Locations

The criteria for locating airborne radioactivity monitors are dependent upon the point of leakage, the ability to identify the source of radioactivity so that corrective action may be performed, and whether personnel may be exposed to the airborne radioactivity.

a. Airborne radioactivity monitors sample normally accessible personnel operating areas for which there is a potential for airborne radioactivity.

b. Areas not normally accessible are monitored, prior to personnel entry, with portable monitors or samplers, depending upon the potential for airborne radioactivity and the work to be performed in the area.

c. Exhaust ducts servicing an area containing processes which, in the event of major leakage, could result in concentrations within the plant approaching the limits established by 10 CFR 20.1-20.601 for plant workers are monitored.

d. Dilution from other exhaust ducts is considered when locating monitors in exhaust systems to ensure maximum coverage and still be able to detect 10 CFR 20.1-20.601 airborne radioactivity limits in the area with the lowest ventilation flow.

e. Outside air intake ducts for the control building are monitored to measure possible introduction of radioactive materials into the control room to ensure the habitability of those areas requiring personnel occupancy for safe shutdown.

f. Airborne radioactivity monitors are located so that the actual sample chamber and detector location are in an area where the background radiation is low. Detailed physical locations are provided on the radiation zone drawings, Figure 12.3-2.

12.3.4.2.2.1.4 Alarms

Each monitor channel is provided with a three-level alarm system. One alarm setpoint is below the background counting rate and serves as a circuit failure alarm. The other two alarm setpoints provide sequential alarms on increasing radioactivity levels. Loss of power will cause an alarm on all three-alarm circuits. The alarms must be manually reset and can be reset only after the alarm condition is corrected.

Alarms from the AiRMS are provided in the control room on the plant annunciator (audible and visual), the balance-of-plant computer alarm display (visual), and the AiRMS Visual Display. In addition, the safety-related channels have individual alarm lights on the safety-related indicators on the AiRMS control panel. The balance-of-plant and AiRMS computers also provide printouts of each alarm.

The pumping systems are controlled from the control room and are provided with a low-flow alarm to alert the operator of pump failure or any other condition which causes a loss of flow through the sample system, and a flow control valve and flow controller to automatically compensate for filter loading.

12.3.4.2.2.1.5 Check Sources

Each monitor is provided with a check source, operated from the control room, which simulates a radioactive sample in the detector assembly for operational and gross calibration checks.

12.3.4.2.2.1.6 Power Supplies

All Class 1E inplant AiRMS are powered from Class 1E motor control centers. The power supplies for all of the monitors are given in Table 12.3-4.

12.3.4.2.2.1.7 Calibration and Maintenance

The airborne radioactivity monitors are calibrated by the manufacturer for the principal radionuclides listed in Table 12.3-3. The manufacturer's calibration standards are traceable to National Bureau of Standards primary calibration standard sources and are accurate to at least 5 percent. The source detector geometry during this primary calibration is identical to the sample detector geometry in actual use. Secondary standards counted in reproducible geometry during the primary calibration are supplied with each continuous monitor.

Channel checks and source checks are performed at regular intervals to ensure proper monitor function. The monitors are recalibrated at regular intervals, and following repairs or modifications, using the secondary radionuclide standard.

12.3.4.2.2.1.8 Sensitivities

The AiRMS is capable of detecting 10 MPC-hours of airborne radioactivity.

The most restrictive isotope for each type of monitor is that isotope with the lowest worker maximum permissible concentration (WMPC), as defined in Table I, Column I, of Appendix B to 10 CFR 20.1-20.601.

For the containment atmosphere and containment purge system monitors, the High and Alert alarm set points are based on the Offsite Dose Calculation Manual.

The sensitivities and alarm setpoints are given in Table 12.3-3. The High alarm points are based on the most restrictive isotopes which are expected to be present. The concentration levels are as defined in Table I, Column I, of Appendix B to 10 CFR 20.1-20.601 or Technical Specification limits, considering dilution.

The sensitivity of the airborne radioactivity monitors is based on a 95 percent confidence level for 0.5 MeV beta or gamma radiation in a 1 mr/hr gamma radiation background at standard pressure and ambient temperature.

The fixed volume noble gas detector assemblies have a minimum detectable concentration of 2×10^{-7} μCi/cc, using Kr-85/Xe-133 as the limiting isotope.

The fixed filter particulate detector assemblies have a minimum detectable concentration of 1×10^{-11} μCi/cc, using Cs-137 as the limiting isotope. The filter assembly has a collection efficiency of 99 percent for particles of 0.3 micron or larger.

The charcoal filter halogen detector assemblies have a minimum detectable concentration of 1×10^{-11} μCi/cc, using I-131 as the limiting isotope. The charcoal filter assembly has a collection efficiency of at least 95 percent for iodine.

12.3.4.2.2.1.9 Ranges and Setpoints

The ranges of the various airborne radioactivity monitors were chosen based on the detection of radioactivity in concentrations ranging from 10 MPC-hours or lower in compartments served up to those from postulated accidents.

The fixed volume noble gas detector assemblies have a range of 10^{-7} to10^{-2} μCi/cc.

The fixed filter particulate detector assemblies have a range of 10^{-12} to 10^{-7} μCi/cc.

The charcoal filter halogen detector assemblies have a range of 10^{-11} to 10^{-6} μCi/cc.

The setpoints are chosen to alert the operators to airborne radioactivity that might be present so that 10 CFR 20.1-20.601 limits on worker exposure or Technical Specification limits will not be exceeded.

The setpoints for control of ventilation are discussed in Section 7.3.

The setpoints on the monitors used for reactor coolant pressure boundary leakage detection are discussed in Section 5.2.5.

The ranges and setpoints for the airborne radioactivity monitors are provided in Table 12.3-3.

12.3.4.2.2.1.10 Expected System Parameters

The expected ranges of system parameters, such as flow rate, composition, and concentrations, are summarized in Table 12.3-3. Detailed information on the individual HVAC systems can be found in Section 9.4.

12.3.4.2.2.2 Monitoring Systems

The systems discussed in the following sections are summarized in Table 12.3-3.

12.3.4.2.2.2.1 Access Control Area Ventilation Exhaust Radioactivity Monitor

The access control area ventilation exhaust radioactivity monitor, 0-GK-RE-41, continuously monitors for particulate radioactivity in the access control area ventilation exhaust upstream of the HVAC filters. The sample is extracted from the duct through an isokinetic nozzle, in accordance with ANSI Standard N13.1-1969, to ensure that a representative sample of the system is obtained. After passing through the fixed filter detector assembly and the pumping system, the sample is discharged back to the duct.

The high and high-high alarms function to alert the operator to airborne particulate radioactivity in the access control area.

Noble gases are monitored by the unit vent monitor described in Section 11.5. The airborne iodine radioactivity in the access control area can be inferred from the particulate and noble gas radioactivity levels. Since no radioactive equipment is in the area, there is no source for release of iodine or noble gas. Particulates may be present due to dust on the protective clothing becoming airborne.

Further determination of the source and the corrective action to be taken will be based on monitoring with portable detection and sampling equipment.

Indication for this system is provided on the AiRMS Visual Display in the control room.

12.3.4.2.2.2.2 Deleted

12.3.4.2.2.2.3 Waste Gas Decay Tank Area Ventilation Exhaust Radioactivity Monitor

The waste gas decay tank area ventilation radioactivity monitor, 0-GH-RE-23, continuously monitors for gaseous radioactivity in the discharge duct from the waste gas decay tank area upstream of the radwaste building exhaust filter adsorber. The sample point provides rapid detection of a leak in the waste gas processing system and, in conjunction with the radwaste building exhaust radioactivity monitor and the radwaste building effluent monitor, helps localize the affected area in the event of an alarm on either monitor.

The sample is extracted from the exhaust duct and passed through the fixed volume noble gas detector assembly and the pumping system. Then the sample is discharged back to the duct. The high alarm provides indication of a leak in the decay tanks, compressors, piping, or valves. The high-high alarm indicates that concentrations in the decay tank room are at or near 10 MPC for the most restrictive isotope expected to be present (Kr-85 or Xe-133).

Back up for this monitor is provided by the radwaste building exhaust and effluent monitors.

Indication of this monitor is provided on the AiRMS Visual Display in the control room.

12.3.4.2.2.2.4 Auxiliary Building Ventilation Exhaust Radioactivity Monitor

The auxiliary building ventilation exhaust radioactivity monitor, 0-GL-RE-60, continuously monitors for particulate radioactivity in the auxiliary building ventilation system upstream of the filter-adsorber units. The sample point is located to monitor between the last point of possible radioactivity entry to the ventilation system from the areas served and the filter adsorber unit. The sample is extracted through an isokinetic nozzle, in accordance with ANSI Standard N13.1-1969, to ensure that a representative sample is provided to the fixed filter particulate detector assembly. Then the sample is discharged through the pumping system back to the duct.

The cartridge filter will be removed for laboratory isotopic analysis upon activation of high activity alarm.

The high alarm alerts the operator to high airborne particulate radioactivity levels in the auxiliary building. Indication of this monitor is provided on the AiRMS Visual Display in the control room.

If required, the portable monitor described in Section 12.3.4.2.2.2.9 and/or grab samples will be utilized to determine airborne radioactivity levels and iodine concentrations in specific areas to aid in the determination of the source of the release.

12.3.4.2.2.2.5 Containment Atmosphere Radioactivity Monitors

The containment atmosphere radioactivity monitors, 0-GT-RE-31 and 0-GT-RE-32, continuously monitor the containment atmosphere for particulate, iodine, and gaseous radioactivity which could result in personnel exposure during periods of containment access. Other functions of these monitors are covered in Sections 5.2.5, 7.3, 9.4, and 11.5.

Samples are extracted from the operating deck level (El. 2047'-6") of the containment through the monitoring system sample lines.

The monitors are located as close as possible to the containment penetrations to minimize the length of the sample tubing and the effects of sample plateout. The sample points are located in areas which ensure that representative samples are obtained. Each sample passes through the penetration and then through the fixed filter (particulate), charcoal filter (iodine), and fixed volume (gaseous) detector assemblies. After passing through the pumping system, the sample is discharged back to the containment through a separate penetration.

Indication is provided for each monitor on individual indicators on the radioactivity monitoring system control panel and through isolated signals, on the AiRMS Visual Display in the control room.

The containment atmosphere radioactivity monitors are seismic Category I systems and completely redundant.

The high and high-high alarms alert the operators to high airborne radioactivity in the containment atmosphere.

12.3.4.2.2.2.6 Containment Purge System Radioactivity Monitors

The containment purge system radiation monitors, 0-GT-RE-22 and 0-GT-RE-33, continuously monitor the containment purge exhaust duct during normal purge operations for particulate, iodine, and gaseous radioactivity for worker protection as backup monitors for the containment atmosphere monitors. Other functions for the containment purge monitors are given in Sections 5.2.5, 7.3, 9.4, and 11.5.

The purge monitors are seismic Category I and completely redundant.

The sample points are located outside the containment between the containment isolation dampers and the containment purge filter adsorber unit.

Each monitor is provided with two isokinetic nozzles to ensure that representative samples are obtained from both normal purge and minipurge.

Isokinetic nozzle selection is accomplished by sample selector valves which automatically align the correct nozzle to the monitor, based on operation of the minipurge and normal purge exhaust fans. The sample is extracted through the selected nozzle and then passes through the selector valve, the fixed filter (particulate), charcoal filter (iodine), and fixed volume (gaseous) detectors. The sample then passes through the pumping system and is discharged back to the duct.

Indication is provided for each monitor on individual indicators on the radioactivity monitoring system control panel and, through isolated signals, on the AiRMS Visual Display in the control room.

The containment purge radiation monitors provide backup for the containment atmosphere radiation monitors. The high and high-high alarms alert the operators to high airborne radioactivity in the containment atmosphere.

For plant conditions during CORE ALTERATIONS and during movement of irradiated fuel within containment, the function of the monitors is to alarm only and the trip signals for automatic actuation of CPIS may be bypassed. One instrumentation channel at a minimum is required for the alarm only function during plant refueling activities.

12.3.4.2.2.2.7 Control Room Ventilation Radioactivity Monitors

The control room ventilation radioactivity monitors, 0-GK-RE-04 and 0-GK-RE-05, continuously monitor the supply air of the normal heating, ventilation, and air-conditioning system for particulate, iodine, and gaseous radioactivity to provide protection for the control room operators in the event of high airborne radioactivity in the control room HVAC supply duct.

This seismic Category I system is completely redundant.

Samples are extracted through individual isokinetic nozzles, in accordance with ANSI Standard N13.1-1969, and flow through the fixed filter (particulate), charcoal filter (iodine), and fixed volume (gaseous) detector assemblies prior to passing through the pumping system for discharge to the auxiliary building atmosphere.

The high and high-high alarms alert operators to high airborne radioactivity in the control room supply duct. The safety control functions are described in Sections 6.4, 7.3, and 11.5. Indication for these monitors is provided on individual indicators on the radioactivity monitoring system control panel and, through isolated signals, on the AiRMS Visual Display in the control room.

12.3.4.2.2.2.8 Fuel Building Ventilation Exhaust Radioactivity Monitors

The fuel building ventilation exhaust radioactivity monitors, 0-GG-RE-27 and 0-GG-RE-28, continuously monitor for particulate, iodine, and gaseous radioactivity in the fuel building ventilation exhaust system for the protection of the workers in the fuel building. The other functions for these monitors are described in Sections 9.4 and 11.5.

During normal operation, each of the monitors extracts a sample from the normal exhaust duct through individual isokinetic nozzles and sample selector valves. This normal sample point is upstream of the fuel building normal exhaust filter adsorber unit.

The high and high-high alarms alert operators to high airborne particulate, iodine, or gaseous radioactivity in the fuel building. The ventilation control functions are described in Sections 7.3 and 11.5.

Indication is provided by individual indicators on the AiRMS control panel and through isolated signals by the AiRMS Visual Display in the control room.

12.3.4.2.2.2.9 Portable Monitor

Portable air radioactivity samplers are provided to allow periodic localized monitoring of specific air volumes of interest independent of the fixed monitor systems. The samplers are used to verify that airborne radioactivity levels within the plant operating spaces are within allowable limits and also to verify the proper operation of fixed monitor systems.

A portable particulate, iodine, and gaseous continuous air monitor or a particulate only continuous air monitor will be used to monitor the local areas where there is a possibility of airborne radioactivity. Maintenance on the radioactive systems, refueling, anticipated operational occurrences, and accidents involving the spread of airborne radioactivity will be monitored locally, using the portable monitor.

12.3.4.2.2.3 Safety Evaluation

Due to their safety-related functions, discussed in other sections, the control room ventilation monitors, the containment atmosphere monitors, the containment purge monitors, and the fuel building exhaust monitors are redundant, independent, seismic Category I with Class 1E power supplies. These monitors all have safety-related control functions which are described in Section 7.3.

The following monitors are located upstream of filters and therefore, are effective for inplant personnel protection:

a. Containment atmosphere

b. Containment purge

c. Control room supply

d. Fuel building exhaust

e. Auxiliary building exhaust

f. Access control area exhaust

g. Waste gas decay tank room exhaust

h. Portable monitor

All the inplant areas where the potential for airborne radioactivity exists are, therefore, monitored. The process and effluent radioactivity monitors are discussed in Section 11.5.

When the dilution factors of the ventilation system are large enough to preclude direct detection of halogens and noble gases at MPC levels, the release of these isotopes will be accounted for because the particulate activity, present along with halogens and noble gases, will be detected at much lower concentrations.

Corresponding ratios among the various species of the airborne radioactivity are periodically determined by laboratory analysis so that the leakage of halogen and gaseous radioactivity can be accounted for. The waste gas decay tank room exhaust radioactivity monitor is designed to detect noble gas radioactivity, because this is the predominant type of activity present. Particulates and the iodine may be present in very small amounts, but will not be significant in this case since they will be removed by the waste gas processing system.

A portable particulate, iodine, and gaseous monitor or a particulate only continuous air monitor is used to monitor the local areas where there is a possibility of airborne radioactivity. Maintenance on the radioactive systems, refueling, anticipated operational occurrences, and accidents involving the spread of airborne radioactivity will be monitored locally, using the portable monitor.

On the basis of the above discussion, the AiRMS is adequate and sufficient to ensure personnel protection from airborne radioactivity. The system will provide indication to the operator that airborne radioactivity exists. The location of the airborne radioactivity can then be further identified by using portable air samplers that can be manually connected to various selected sample points in the building. Use of these sample points will direct radiation protection personnel to the particular area of concern. The exact location of the airborne radioactivity can then be found by sampling the particular subcompartments in that area.

The combination of the AiRMS, in conjunction with administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns throughout the plant, plant equipment layout, and restricted radiation Zone E areas, is sufficient to ensure that airborne radioactivity levels are safe in terms of the required duration of personnel access throughout all areas of the plant.

A general review of these concepts follows:

a. Equipment location is such that hot piping and equipment are located in Radiation Zone D and E areas, which are restricted, and entry is limited by administrative control. Radiation Zone B and C areas do not contain piping and components that would result in significant airborne radioactivity sources. This reduces the possibility of airborne radioactivity exposure to occupants of Radiation Zone B and C areas where general entry is permitted.

b. Air flow patterns are consistent with the basic ventilation design criteria of the plant. Clean filtered outside air is supplied to Zone B areas (corridors, clean areas); these areas are exhausted by drawing air into the rooms and areas of successively higher potential for airborne contamination. Air flow is such that air flow reversal or exfiltration from potentially contaminated areas is precluded. This ventilation arrangement eliminates the possibility of personnel exposure to airborne radioactivity in continuous occupancy areas, such as Radiation Zone B areas.

c. High radiation areas (Radiation Zone E areas) where dose levels exceed 100 mrem/hr are restricted and conspicuously posted in accordance with 10 CFR 20.100-20.2401. These areas are not normally entered.

d. Radiation protection programs are discussed in Section 12.5.

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TABLE 12.3-1 LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

GRACE I	Multigroup, multiregion, gamma-ray attenuation code used to compute gamma heating and gamma dose rates in slab geometry (Ref. 12).
GRACE II	Multigroup, multiregion, gamma-ray attenuation code used to compute the dose rate or heat generation rate for a spherical or a cylindrical source with slab or concentric shields (Ref. 13).
ANISN	Multigroup, multiregion code solving the Boltzman transport equation for neutrons or gamma-rays in one dimensional slab, cylindrical, or spherical geometry (Ref. 14).
SDC	Multigroup, multiregion, Kernal integration gamma-ray, shield design code which calculates dose rates for 13 geometry options (Ref. 15).
QAD	Multigroup, multiregion, three-dimensional, point Kernal code which calculates fast neutron and gamma-ray dose and heat generation rates (Ref. 16).
NAP	Determines activation emission source strengths as a function of neutron exposure and decay time (Ref. 17).
MORSE-CG	Three-dimensional Monte Carlo neutron and gamma ray general transport code (Ref. 18).
DOT III	Two-dimensional neutron, gamma ray, discrete ordinate, transport code (Ref. 19).
ORIGEN	Isotope generation and depletion code which solves equations of radioactive growth and decay for isotopes of arbitrary coupling (Ref. 21).
G^3	A general purpose gamma-ray scattering code (Ref. 20).

TABLE 12.3-2 AREA RADIATION MONITORS

Instrument Number	Location	Radiation Zone	Range (mrem/hr)	Hi Alarm (mrem/hr)	Hi-Hi Alarm (mrem/hr)
0-SD-RE-1	Radwaste Building Corridor, Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-2	Radwaste Building Corridor, Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-3	Radwaste Building Corridor, Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-4	Radwaste Building Corridor, Ground Floor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0 SD-RE-5	Radwaste Building Corridor, Ground Floor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-6	Solid Radwaste Area	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-7	Truck Space	B	$0.1 - 10^4$	15 (8)	100 (8)
0-SD-RE-8	Sample Laboratory	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-9	RW Bldg Valve Room Corridor	C	$0.1 - 10^4$	15 (1)(8)	100 (2)(8)
0-SD-RE-10	RW Bldg Valve Room Corridor	C	$0.1 - 10^4$	15 (1)(8)	100 (2)(8)
0-SD-RE-11	RW Bldg HVAC Filter Unit	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE 12	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-13	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-14	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-15	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-16	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-17	Pipe Tunnel & Personnel Access	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-18	Aux Bldg Ground Floor Corridor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-19	Aux Bldg Ground Floor Corridor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-20	Aux Bldg Valve Room Corridor Ground Floor	C	$0.1 - 10^4$	15 (1)(8)	100 (2)(8)
0-SD-RE-21	Aux Bldg Valve Room Corridor Ground Floor	C	$0.1 - 10^4$	15 (1)(8)	100 (2)(8)
0-SD-RE-22	Aux Bldg Corridor Ground Floor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-23	Aux Bldg Corridor Ground Floor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-24	RC Sample Room	C	$10 - 10^5$	15 (1)(8)	100 (1)(8)
0-SD-RE-25	Filter Unit Aux Bldg	A	$0.1 - 10^4$	0.5 (1)(8)	2.5 (2)(8)
0-SD-RE-26	RHR Heat Exchanger Outside	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-27	Ctmt Purge Filter Unit	C	$0.1 - 10^4$	15 (1)(8)	100 (2)(8)
0-SD-RE-28	Personnel Hatch	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-29	Laundry Decontamination Facility	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-30	Laundry Decontamination Facility	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)

TABLE 12.3-2 (Sheet 2)

Instrument Number	Location	Radiation Zone	Range (mrem/hr)	Hi Alarm (mrem/hr)	Hi-Hi Alarm (mrem/hr)
0-SD-RE-31	Hot Laboratory	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-32	Control Bldg Corridor	B	$0.1 - 10^4$	2.5 (1)(8)	15 (2)(8)
0-SD-RE-33	Control Room	A	$0.1 - 10^4$	0.5 (1)(8)	2.5 (2)(8)
0-SD-RE-34	Cask Handling Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
0-SD-RE-35	New Fuel Storage Area (3)	B	$0.1 - 10^4$	2.5 (1)	≤15 (2)
0-SD-RE-36	New Fuel Storage Area (3)	B	$0.1 - 10^4$	2.5 (1)	≤15 (2)
0-SD-RE-37	Fuel Storage Pool Area (3)	B	$0.1 - 10^4$	2.5 (1)	≤15 (2)
0-SD-RE-38	Fuel Storage Pool Area (3)	B	$0.1 - 10^4$	2.5 (1)	≤15 (2)
0-SD-RE-39	Seal Table Area	E	$0.1 - 10^4$	1000 (4)(8)	10000 (5)(8)
0-SD-RE-40	Personnel Access Hatch Area	E	$0.1 - 10^4$	1000 (4)(8)	10000 (5)(8)
0-SD-RE-41	Manipulator Bridge Crane (6)	E	$0.1 - 10^4$	30 (7)	100 (7)
0-SD-RE-42	Containment Building	E	$0.1 - 10^4$	1000 (4)(8)	10000 (5)(8)
0-SD-RE-47	Boron Concentration Monitoring System Room	E	$0.1 - 10^4$	15 (1)(8)	100 (2)(8)

(1) High alarm set for maximum radiation level for the radiation zone for that area.

(2) High-high alarm set for the maximum radiation level for the radiation zone above the one for that area.

(3) These monitors serve as criticality monitors.

(4) High alarm based on highest expected radiation level for the area during full power operation.

(5) High-high alarm set one order of magnitude above high alarm to indicate extremely high radiation level.

(6) This monitor will be operational on the bridge crane at these setpoints only during plant shutdown when fuel handling activities are occurring in mode 6. During modes 1, 2, 3, 4, and 5 the high alarm and high-high alarm setpoints will be reset to 5,000 mrem/hr and 10,000 mrem/hr respectively.

(7) High and high-high alarms are established for the protection of operators on the bridge crane during mode 6.

(8) High alarm and high-high alarm setpoints listed are the normal alarm setpoints. Alarm setpoints above these levels may be used to support plant conditions and will be administratively controlled.

TABLE 12.3-3 INPLANT AIRBORNE RADIOACTIVITY MONITORS

Monitor	Type (continuous)	Range µCi/cc	MDC(1) µCi/cc	Controlling Isotope	Alert (15) Alarm µCi/cc	
OGTRE31	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1.0×10^{-9} (16)	
OGTRE32	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	1.0×10^{-8}	
Containment atmosphere monitors	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	1.0×10^{-4}	
OGTRE22	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	5×10^{-8}	
OGTRE33	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	5×10^{-8}	
Containment purge system monitors	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	(10)	
OGGRE27	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
OGGRE28	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	9×10^{-9}	
Fuel building exhaust monitors (2)	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	1.6×10^{-3}	
OGKRE04	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
OGKRE05	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	9×10^{-9}	
Control room air supply monitors	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Xe-133	1.1×10^{-3}	
OGLRE60	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-8}	
Auxiliary Building ventilation exhaust monitor						
OGKRE41	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	1×10^{-9}	(8)
Access control area ventilation exhaust monitor						
OGHRE23	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Kr85 Xe-133	1×10^{-5}	(8)
Waste gas decay tank area ventilation exhaust monitor						
Portable monitor	Particulate (3)	10^{-12} to 10^{-7}	1×10^{-11}	Cs137	NA	
	Iodine (4)	10^{-11} to 10^{-6}	1×10^{-10}	I131	NA	
	Gaseous (3)	10^{-7} to 10^{-2}	2×10^{-7}	Kr85	NA	

TABLE 12.3-3 (Sheet 2)

High (15) Alarm µCi/cc		Flow Ventilation Flow (cfm)	Subcompartment Flow Rate (cfm)	Dilution Factor		Minimum Required Sensitivity (µCi/cc)		Monitor Control Function
1×10^{-7}		420,000	NA	NA		1×10^{-7}(6)	(6)	NA
9×10^{-7}		420,000	NA	NA		9×10^{-8}(6)	(6)	NA
6.0×10^{-4}		420,000	NA	NA		1×10^{-4}(6)	(6)	NA
1×10^{-7}		20,000/4,000	NA	NA		1×10^{-7}(6)	(6)	See Table 11.5-3 for process and control functions.
9×10^{-8}		20,000/4,000	NA	NA		9×10^{-8}(6)	(6)	
(11) (14)		20,000/4,000	NA	NA		1×10^{-4}(6)	(6)	
1×10^{-7}		20,000	NA	NA		1×10^{-7}(6)	(6)	See Table 11.5-3 for process and control functions.
9×10^{-8}		20,000	NA	NA		9×10^{-8}(6)	(6)	
3.2×10^{-3}	(13)	20,000	NA	NA		1×10^{-4}(6)	(6)	
1×10^{-7}		2000	NA	NA		1×10^{-7}(6)	(6)	See Table 11.5-3 for process and control functions.
9×10^{-7}		2000	NA	NA		9×10^{-8}(6)	(6)	
2.2×10^{-3}	(12)	2000	NA	NA		1×10^{-4}(6)	(6)	
1×10^{-7}		12,000	100	8×10^{-3}	(5)	8×10^{-10}	(6),(9)	Alarms
1×10^{-8}	(7)	6,000	100	1.67×10^{-2}	(5)	1.67×10^{-9}	(6),(9)	Alarms
1×10^{-4}	(7)	500	250	0.5	(5)	5×10^{-5}	(6),(9)	Alarms
	NA	NA	NA	NA				Alarms
	NA	NA	NA	NA				
	NA	NA	NA	NA				

Sample Flow for each channel is 3 cfm.

(1) MDC = minimum detectable concentration.

(2) When fuel is in the building.

(3) Beta scintillation detector.

(4) Gamma scintillation detector.

(5) Dilution factor = $\frac{\text{Subcompartmental flow in cfm}}{\text{Total flow in cfm}}$

(6) Minimum required sensitivity of monitor in µCi/cc at 10 MPChrs for the controlling isotope = dilution factor x 10 MPC.

(7) 10 MPC x dilution factor.

TABLE 12.3-3 (Sheet 3)

(8) MPC x dilution factor.

(9) Grab samples will be analyzed in the laboratory, and iodine concentrations will be calculated, using previously established ratios.

(10) Alert alarm is administratively established at a point sufficiently below the High alarm so as to provide additional assurance that Offsite Dose Calculation Manual (ODCM) limits are not exceeded.

(11) High alarm is set to ensure that ODCM limits are not exceeded.

(12) Submersion dose rate does not exceed 2 mr/hr in the control room.

(13) Submersion dose rate does not exceed 4 mr/hr in the fuel building.

(14) High alarm setpoint is established to ensure that ODCM limits are not exceeded.

(15) Alert and High alarm setpoint values do not include instrument loop uncertainty estimates.

(16) Alert alarm value is set to meet the criteria of Note 10 and to meet RCS leakage detection requirements described in FSAR Section 5.2.5.2.3.

TABLE 12.3-4 POWER SUPPLIES FOR AREA AND IN-PLANT AIRBORNE MONITORS

Area Radiation Monitors

Monitor Number	Normal Power Supply	Restored After Loss of Offsite Power	Remarks
0-SD-RE-1 0-SD-RE-2 0-SD-RE-3 0-SD-RE-4 0-SD-RE-5 0-SD-RE-6 0-SD-RE-7 0-SD-RE-8 0-SD-RE-9 0-SD-RE-10 0-SD-RE-11 0-SD-RE-12 0-SD-RE-13 0-SD-RE-14 0-SD-RE-15 0-SD-RE-16 0-SD-RE-17 0-SD-RE-18 0-SD-RE-19 0-SD-RE-20 0-SD-RE-21 0-SD-RE-22 0-SD-RE-23 0-SD-RE-24 0-SD-RE-25 0-SD-RE-26 0-SD-RE-27 0-SD-RE-28 0-SD-RE-29 0-SD-RE-30 0-SD-RE-31 0-SD-RE-32 0-SD-RE-33 0-SD-RE-34 0-SD-RE-35 0-SD-RE-36	Supplied by regulated instrumentation ac power which is supplied from the diesel generators on loss of offsite power.	Yes	

TABLE 12.3-4 (Sheet 2)

Monitor Number	Normal Power Supply	Restored After Loss of Offsite Power	Remarks
0-SD-RE-37 0-SD-RE-38 0-SD-RE-39 0-SD-RE-40 0-SD-RE-41 0-SD-RE-42 0-SD-RE-47			
Containment atmosphere 0-GT-RE-31 0-GT-RE-32	Class 1E MCCs	Yes	
Containment purge system 0-GT-RE-22 0-GT-RE-33	Class 1E MCCs	Yes	
Fuel building exhaust 0-GG-RE-27 0-GG-RE-28	Class 1E MCCs	Yes	
Control room air supply 0-GK-RE-04 0-GK-RE-05	Class 1E MCCs	Yes	
In-Plant Airborne Radioactivity Monitors (Non-1E)			
Auxiliary building ventilation exhaust 0-GL-RE-60	Non-1E MCCs	No	Power is lost to system also, so monitor reading is not meaningful.
Access control area ventilation exhaust 0-GK-RE-41	Non-1E MCCS	No	Power is lost to system also, so monitor reading is not meaningful.
Waste gas decay tank area ventilation exhaust 0-GH-RE-23	Non-1E MCCs	No	Power is lost to system also, so monitor reading is not meaningful.

12.4 DOSE ASSESSMENT

Radiation exposures in the plant are primarily due to direct radiation from components and equipment containing radioactive fluids. In addition, in some plant radiation areas there can be radiation exposure to personnel due to the presence of airborne radionuclides. In-plant radiation exposures during normal operation and anticipated operational occurrences are discussed in Section 12.4.1. Radiation exposures due to direct radiation at locations outside the plant structures, such as the boundary of the restricted area, are a function of the plant layout, equipment selection, and detailed system and shielding designs and are expected to be negligible. Radiation exposures due to the airborne radioactive effluent plume at these locations are expected to be insignificant. The radiation exposures at these locations are discussed in Section 12.4.2.2.

Radiation exposures to operating personnel will be within 10 CFR 20 limits. Radiation protection design features described in Section 12.3 and the radiation protection program outlined in Section 12.5 will assure that the occupational radiation exposures (ORE) to operating personnel during operation and anticipated operational concurrences will be as low as is reasonably achievable (ALARA).

12.4.1 EXPOSURES WITHIN THE PLANT

12.4.1.1 Direct Radiation Dose Estimates

Annual man-rem doses from direct radiation during the performance of routine functions, such as operation and surveillance, normal maintenance, radwaste handling, refueling, and inservice inspection, have been estimated, using the following bases:

a. Radiation exposure data from operating PWRs are given in Tables 12.4-1 through 12.4-11 (Ref. 1, 2, 3) and Figure 12.4-1.

b. Expected average dose rates in plant radiation areas are discussed in this section.

c. Expected occupancy times for various work function personnel in the different plant radiation areas are listed in Table 12.4-12.

d. Anticipated manpower requirements for each unit are given in Table 12.4-12.

e. Table 12.4-12 provides an estimate of the distribution of the annual man-rem according to work function.

The maximum and expected average dose rates in the plant radiation areas are given below:

Zone	Maximum Dose Rate (mrem/hr)	Expected Average Dose Rate (mrem/hr)
A	0.5	0.1
B	2.5	0.5
C	15	2.5
D	100	15
E	>100	100

The maximum dose rates are determined by shielding calculations based on conservative assumptions regarding sources, self-shielding, locations, etc. The expected dose rates are estimated by assuming a failed fuel percentage of 0.12 and that stringent water chemistry control and improved design will minimize crud buildup and hence the expected dose rates in various radiation zones and recognition that real maximum doses in a given zone are localized effects. The expected average doses given above are used in computing the doses for personnel involved in all operations, except inservice inspection and special maintenance. For personnel involved in the performance of inservice inspection (ISI) and special maintenance tasks, an expected average dose rate of 200 mR/hr in the E Zone is used since these personnel will generally be working on reactor coolant system components.

Direct radiation exposures to plant personnel can result from the performance of special maintenance functions. In view of the radiation protection design features described in Section 12.3 and the radiation protection program outlined in Section 12.5, it is expected that exposures due to special maintenance will be minimized. However, an annual exposure of 150 man-rem/unit is realistically estimated, based on experience at operating PWRs.

Exposure to plant personnel from direct radiation during the performance of routine functions is estimated to be approximately 220 man-rem/yr/unit. Details of the man-rem estimates are given in Table 12.4-12. A breakdown of the exposures (including the special maintenance category) by work functions is provided in Table 12.4-12. Table 12.4-13 provides the percentage of the annual total man-rem associated with each work function.

Total occupancy in various radiation zones: Total occupancy time for personnel involved in different work functions in various radiation zones is as follows:

a. Routine operation and surveillance - The total occupancy time for personnel involved in this work function in various radiation zones is expected to be 2,080 hrs/yr. The major portion of the occupancy is expected to be in radiation Zones A and B. Combined occupancy in radiation Zones C, D, and E is expected to be approximately 4 percent of the total annual occupancy. The average annual dose for an individual involved in this work category is expected to be about 1 rem. The distribution of occupany times in various radiation zones is listed in Table 12.4-12.

b. Routine maintenance: The total occupancy time for personnel involved in this work function in various radiation zones is expected to be 2,080 hrs/yr. Personnel involved in this work category are expected to spend more time in high radiation areas. However, by following maintenance procedures, such as flushing equipment in high radiation areas before performing maintenance and also by removing smaller equipment from high radiation areas to lower radiation areas for maintenance, the dose rate for personnel can be minimized. Consequently, the annual average dose rate for an individual involved in this work function is expected to be about 3.7 rem. The distribution of occupancy time in various radiation zones is listed in Table 12.4-12.

c. Inservice inspection: The total occupancy time for personnel involved in this work function in various radiation zones is expected to be 240 hrs/yr (Table 12.4-12) at the rate of 40 hrs/wk for 8 wks/yr. The annual average dose for an individual involved in this work function is approximately 0.5 rem.

d. Refueling: The refueling work is performed by personnel involved in routine operation and surveillance, radiation protection and rad chemistry, and routine maintenance. The expected occupancy time for personnel involved in the refueling operation is about 160 hrs/yr. The average dose per individual in this category is estimated to be 0.5 rem over the occupancy period. The personnel involved in the refueling operation are expected to spend more time in the high radiation areas. The initial crud burst in the refueling pool and fuel storage pool regions will initially result in occupancies in radiation Zone C. However, the operation of the cleanup systems will result in significant reduction of the dose rates in the regions where occupancy is expected. In view of the above considerations, a combined occupancy of 70 percent in radiation Zones B, C, D, and E is considered reasonable. A realistic distribution of personnel occupancies in various radiation zones is provided in Table 12.4-12.

e. Special maintenance: The total occupancy time for personnel involved in special maintenance problems, generally unanticipated, in various radiation zones is expected to be 320 hrs/yr, and the dose rate per individual involved in this work function is estimated to be about 0.7 rem/yr. The distribution of occupancy time in various radiation zones is provided in Table 12.4-12.

f. Radwaste processing: The radwaste operations will be performed by personnel involved in work functions such as routine operation and surveillance, radiation protection, rad chemistry, and routine maintenance. The total times expended by personnel in various radiation levels for the category are given in Table 12.4-12. The expected occupancy time for personnel involved in radwaste processing is about 2080 hrs/yr. The annual average dose per individual in this category is estimated to be about 500 mrem.

12.4.1.2 Airborne Radioactivity Dose Estimates

12.4.1.2.1 Exposures Due to Airborne Radioactivity

As already discussed in Section 12.2.2, negligible airborne concentrations and consequently negligible airborne radio-activity exposures are expected in those areas of the auxiliary, radwaste, and turbine buildings which are accessible (Ref. 4). Exposures due to airborne radioactivity are possible in the containment and fuel building both during power operation and refueling. However, the design of the plant operating procedures described in Sections 12.3.3, 12.3.4, and 12.5.3, and expected limited occupancies in these buildings will minimize exposures to airborne activity and ensure that the doses to an individual from airborne radioactivity are small fractions of the 10 CFR 20 limits for occupational workers and that annual man-rem exposures comply with the ALARA criteria within the plant. The annual man-rem exposures from airborne radioactivity will be a small fraction of the annual man-rem exposures due to direct radiation. Annual occupancy (man-hours), dose rates, and man-rem due to airborne radioactivity in these areas are given in Tables 12.4-14 and 12.4-15, respectively.

12.4.1.2.2 Model for Calculating Exposures Due to Airborne Radioactive Sources

Thyroid and inhalation doses via inhalation pathway are calculated using the following equation:

$$D_o = \sum_i C_i \cdot (BR) \cdot t \cdot DF_{oi}$$

where

C_i = Airborne concentration of the i^{th} nuclide in pci/cm^3

and

BR	=	Breathing rate for occupational worker in cm^3/sec = 347
t	=	Time duration of inhalation of radioactivity contaminated air in seconds
DF_{oi}	=	Dose factor for adult for organ O (thyroid or lung) via inhalation in mrem/pCi inhaled for the i^{th} isotope (These dose factors are taken from Regulatory Guide 1.109, Rev. 1)
D_o	=	Dose in millirems to organ O due to inhalation

Total body submersion doses are calculated, using a finite cloud model.

Annual man-rem exposures due to airborne radioactivity are calculated using the following equation:

$$D_o = (DR)_o \cdot 10^{-3} \cdot h$$

where

$(DR)_o$	=	Dose rate for organ in mrem/hr

and

h	=	Annual occupancy in man-hours/yr
D_o	=	Annual exposure in man-rem/yr

12.4.1.3 Illustrative Examples of Dose Assessment

Dose assessments for various operations were based on actual operating plant data. A number of typical examples are provided in Table 12.4-1. Note that the maximum/ minimum values for number of personnel do not correspond to the maximum/minimum number of days required or dose rates so that the man-rem totals are not simple multiplications of the maximum/minimum factors. The dose assessments in Table 12.4-1 were derived from the average number of personnel, average length of time, and average dose rate to perform each particular operation.

a. Operation and Surveillance

The dose rates in the corridors and other normally occupied areas are expected to be much lower than the maximum radiation levels for each zone shown in the radiation zone drawings (Figure 12.3-2). The expected radiation levels are provided in Section 12.4.1.1. Based on these expected dose rates and typical time periods for operation and surveillance, exposures were calculated. Typical examples from operating plants are given in Table 12.4-1. The total exposures for this category range from 13-30 man-rem.

b. Routine Maintenance

A number of examples of normal man-rem associated with maintenance are provided in Table 12.4-1. The total number of annual man-rems of exposure associated with this category depend upon many variables, such as equipment run times and breakdowns, number of skilled personnel available, schedule, and crud trapping. Typically, routine maintenance can account for a large percentage of the annual man-rem.

c. Radwaste Processing

Annual exposures for radwaste processing were determined as a result of the system evaluations described in Section 12.1.2.4. Average expected dose rates and personnel stay times were used. Typical data from operating plants for various tasks are presented in Table 12.4-1.

d. Refueling

Based on operating plant data, refueling operations have required 15 to 30 days, using 8 to 61 personnel who receive from 32 to 347 mrem/day. Man-rem totals have ranged from 13 to 66. Typical man-rem exposures for individual tasks are provided in Table 12.4-1. Much of the exposure is associated with removal and replacement of the reactor vessel head and its appurtenances.

e. Inservice Inspection

Inservice inspections at operating PWRs have taken much of the refueling outage, requiring from 3 to 38 men depending on the inspections scheduled, at up to 2500 mrem/day. Total exposures have ranged from 10 to 24 man-rem. Steam generator eddy current testing, as required by the Technical Specifications, is perhaps the largest single cause of exposure for ISI. Table 12.4-1 provides a listing of various ISI functions and their associated man-rem from operating plant data.

f. Special Maintenance

Special maintenance is generally of a nonrecurring nature and not readily predictable. It includes implementation of design changes and unexpected repair or replacement of equipment and components.

Designs are continually being improved so that the newer plants should not experience all of the problems that have occurred on operating plants. Some special maintenance has created greater than 150 man-rems of exposure, but the frequency of occurrence is irregular. An estimated annual average exposure from special maintenance work is 150 man-rems.

Some examples of special maintenance are provided in Table 12.4-1.

Steam generator tube plugging is perhaps the operation causing the largest exposures in the category of special maintenance. The exposures associated with tube plugging are dependent on the number of tubes to be plugged. Remotely operated equipment and explosive plugging will be used to minimize personnel exposures for this task. Improvements in secondary system water chemistry and tube support plates have reduced the likelihood of the need to plug tubes.

The volatile chemical treatment to be employed for the secondary system will greatly alleviate the need for sludge lancing of the secondary side.

Special materials are being utilized for the radwaste evaporator tubes to maximize corrosion resistance and thereby minimize the maintenance that might be needed.

Other special maintenance, such as valve operator or pump impeller replacement, might occur several times in the life of the plant, but exposures would be much less than 150 man-rem.

12.4.2 EXPOSURES AT LOCATIONS OUTSIDE PLANT STRUCTURES

12.4.2.1 Direct Radiation Dose Estimates

Direct radiation to plant personnel outside of plant structures from the containment, auxiliary, radwaste and turbine buildings is negligible. Sources of radioactivity not stored in plant structures are the Old Steam Generator Storage Facility (OSGSF), the Independent Spent Fuel Storage Installation (ISFSI), the ISFSI Support Building, reactor makeup water storage tank, refueling water storage tank, condensate storage tank, and two discharge monitor tanks. A fence is provided around the ISFSI and refueling water storage tank to restrict access due to potential dose rates. Administrative controls will be used to limit access to the ISFSI Support Building if radioactive material is stored there.

Dose rate at the nearest site boundary from the original steam generators, the original reactor vessel closure head, the ISFSI with 48 casks loaded, the ISFSI support building, the reactor makeup water storage tank, refueling water storage tank, condensate storage tank, and two discharge monitor tanks has been calculated to be approximately 10^{-5} mrem/yr. Using the year 2020 projected populations, a population exposure of less than 10^{-3} man-rem will be received out to 50 miles in the exposed sectors.

12.4.2.2 Exposures Due to Airborne Radioactivity

Estimates of doses at the site boundary due to released activity are given in Section 11.3.3.

12.4.3 REFERENCES

1. NUREG-0109, Occupational Radiation Exposure at Light Water Cooled Power Reactors, 1969-1975.

2. NUREG-75/032, Occupational Radiation Exposure At Light Water Cooled Power Reactors, 1969-1974.

3. NUREG-0463, Occupational Radiation Exposure, Tenth Annual Report, 1977.

4. NUREG/CR-0140, In-Plant Source Term Measurements at Fort Calhoun Station Unit 1.

TABLE 12.4-1 ILLUSTRATIVE EXAMPLES OF DOSE ASSESSMENT

Operation and Surveillance:

Description of Task	No. of Days (1)	No. of Personnel (2)	Mrem/man-day	No. of man-rem
Fuel bldg.	Refueling	29	10-113	6
Aux. bldg.	Entire yr	5-9	9-186	2
Aux. bldg. equip. test	18	3	7-65	0.2
Ctmt. (initial survey after SD)	1-3	5-17		0.1-0.5
Total for oper. & surv.				13-30
Routine Maintenance				
Misc. instrument calibration for pressurizer	7	3	19-92	0.2
Aux. bldg. valves	19-22	4-16	13-70	2
Aux. bldg. general maintenance	8-45	8-13	3-242	0.3-2
Aux. bldg instrument calibration	41	11	5-48	0.5
Ctmt. decon.	Refueling	33	21-170	7
Fuel bldg. decon.	46	5	2-20	0.3
Radwaste bldg. general maintenance	11	9	28-137	0.8
Radwaste bldg. decon.	21-24	3	9-143	4
Aux. bldg. decon.	11	4	3-19	0.1
Ctmt. valves	8-74	10-27	33-185	0.9-6
Ctmt. instrument calibration	8-65	4-30	9-75	0.3-5
Replace oil in reactor coolant pumps	5	5	12-45	0.3
Remove charcoal filters for ctmt. cleanup	4-5	7-36		0.1-0.5
Replace charcoal filters for ctmt. cleanup	6-45	2-20		0.3-0.8
Repair dampers & duct	8	4-5		0.2-0.3
General decon. & relamping	29-68	33-71		7-11
Decon. Refueling canal	2-18	4-23		0.1-6

TABLE 12.4-1 (Sheet 2)

Description of Task	No. of Days (1)	No. of Personnel (2)	Mrem/man-day	No. of man-rem
Overhaul reactor coolant pump seals	21-30	18-34		
Reactor coolant pump motor work	35	18-30		6
Uncouple/couple reactor coolant pumps	22	15-19		0.8
Install/remove reactor coolant pump scaffolding	12	7-10		0.3
Repair ctmt. sump pumps & level Indicators	9-75	9-30		0.4-4
Repair pressurizer relief valves	5-31	4-9		0.2-0.3
Replace excore detectors	23-42	15-30		1-8
Polar crane maintenance	9-78	9-22		0.3-3
Ctmt. pressure test & valve repair	60	26-32		2
Remove & clean SI system strainers	3	14-25		0.6
Check & repair snubbers	3-43	6-21		0.1-2
Remove equip. hatch	5-10	2-9		0.002-0.1
Replace equip. hatch	2-3	3-7		0.03-0.04
<u>Radwaste Processing</u>				
Waste drum loading	Refueling	7	62-280	6
Loading spent resin casks	Refueling	10	78-760	4
Dry waste drum handling	Refueling	3-59		0.2-4
Spent fuel pool filter replacement	2	4		0.6
Transferring resin	1	4-5		0.1
<u>Refueling</u>				
Install and remove reactor cavity seal ring	7-16	7-30		0.4-0.9
Remove transfer tube flange	12-17	2-10		0.3-0.4
Install & remove upper guide structure lifting rig	3-31	6-25		0.2-3
Rig underwater lights	2-18	4-13		0.02-0.3

TABLE 12.4-1 (Sheet 3)

Description of Task	No. of Days [1]	No. of Personnel [2]	Mrem/man-day	No. of man-rem
Flood refueling canal	4	5-9		0.3
Fuel handling (ctmt.)	14-16	49-92		3-7
Fuel handling (fuel bldg.)	5-28	25-77		0.5-1
Incore instrumentation removal	3-36	13-24		2-6
Remove & replace cable to reactor head	11-13	9-27		0.9-3
Remove tool access flanges & uncouple control rods	10-13	9-39		0.8-9
Install bullet noses	1-9	6-15		0.7-2
Remove bullet noses	4-5	2-4		0.1-0.4
Remove & replace head insulation	7-10	4-5		0.3-0.4
RPV studs - remove, replace & clean	10-19	26-40		3-13
Install & remove alignment pins	2-11	3-12		0.1-1
Connect Polar Crane to Integrated Head Assembly (IHA) tripod & remove RPV head and IHA	2-10	11-37		0.6-3
Replace RPV head and IHA	1-3	6-18		0.4-7
Decon. PRV head & vessel flanges & replace "O" rings	7-14	10-24		3-5
Remove & replace holddown ring	4-6	14-31		3-8
Clean stud plugs & holes and install & remove plugs	4-11	12-26		1-6
Clean incore instrument flanges, studs, & nuts	9-28	12-32		2-6
RPV head gasket replacement	6-32	19-35		1-4
Drain, fill, and vent reactor coolant system	2-9	5-10		0.1-0.7
Install handrails around refueling canal	14	6-8		0.1

TABLE 12.4-1 (Sheet 4)

Description of Task	No. of Days (1)	No. of Personnel (2)	Mrem/man-day	No. of man-rem
Total for refueling	15-30	8-61	32-347	13-66
<u>Inservice Inspection</u>				
Steam generator tubes	7-51	13-38	29-2500	4-22
Spent fuel sipping	20	17	14-242	3
Spent fuel inspection	57	22-30		1-2
Remove & replace primary manway covers	4-14	12-37		3-7
Remove & replace secondary manway covers	7-8	4-16		0.4-0.8
Steam generator secondary side	2-4	11-17		0.2-2
Install tent around primary manway	6	6		0.9
Install primary loop dams	2	3		1
Clean steam generator manway studs	6	4-12		0.6
Total for ISI				10-24
<u>Special Maintenance</u>				
Replace upper guide structure on control rods	69	78	17-785	13
Sludge lancing of steam generator secondary side	5	16	19-105	2
Steam generator tube plugging	7-30	5-10		20-200

NOTES:

(1) The number of days indicates the period over which the particular task was performed. It does not mean that personnel were working full time for that period.

(2) The number of personnel indicates the total number of personnel who worked on a particular task. It does not mean that all of the personnel were working simultaneously or full time on that task.

TABLE 12.4-2 AVERAGE NUMBER OF PERSONNEL PER PWR UNIT FOR THE PERIOD 1969-1977 (REF. 1, 2, 3)

Year	Number of Units	Average Number of Personnel/Unit
1969	2	131
1970	2	493
1971	3	250
1972	4	401
1973	5	774
1974	10	602
1975	14	548
1976	25	593
1977	28	642
1969-1977	Overall average	580

NOTES:

1. Only PWRs at power levels ≥450 MWe have been considered, with the exception of San Onofre 1 (430 MWe)
2. Only PWRs that have been in commercial operation for at least 18 months at the end of the year have been considered. Multiple units have been considered for any year during which all units have been in commercial operation for at least 18 months prior to the end of the year.
3. Units considered for both tables are the same, with one exception. For the year 1972, Point Beach Unit 1 was used in Table 12.4-3, but not in Table 12.4-2 since no data on the number of personnel were available for 1972.

TABLE 12.4-3 AVERAGE OCCUPATIONAL RADIATION EXPOSURE (MAN-REM DOSE) PER PWR UNIT FOR THE PERIOD 1969-1977 (REF. 1, 2, 3)

Year	Number of Units	Average Man-Rem Dose/Unit
1969	2	74
1970	2	422
1971	3	274
1972	5	482
1973	5	610
1974	10	439
1975	14	460
1976	25	461
1977	28	422
1969-1977	Overall average	441

NOTES:

1. Only PWRs at power levels ≥450 MWe have been considered, with the exception of San Onofre 1 (430 MWe)

2. Only PWRs that have been in commercial operation for at least 18 months at the end of the year have been considered. Multiple units have been considered for any year during which all units have been in commercial operation for at least 18 months prior to the end of the year.

3. Units considered for both tables are the same, with one exception. For the year 1972, Point Beach Unit 1 was used in Table 12.4-3, but not in Table 12.4-2 since no data on the number of personnel were available for 1972.

TABLE 12.4-4 AVERAGE OCCUPATIONAL RADIATION EXPOSURE (MAN-REM DOSE) BASED ON PWR PLANT AGE (REF 1, 2, 3)

Initial Operation Date	Units Considered										
	Year of Operation (Range of months)	1 (9-18)	2 (21-30)	3 (33-42)	4 (45-54)	5 (57-66)	6 (69-78)	7 (81-90)	8 (93-102)	9 (105-114)	10 (117-126)
	No. of Units	14	15	13	9	8	5	4	3	2	2
	Man-Rem per Unit	123	330	664	571	775	481	341	465	665	745
1/68	Conn. Yankee	-	X	X	X	X	X	X	X	X	X
1/68	San Onofre 1	-	X	X	X	X	X	X	X	X	X
3/70	Ginna	X	X	X	X	X	X	X	X		
3/71	Robinson 2	X	X	X	X	X	X	X			
12/71	Palisades	X	X	X	X	X	X				
12/72	Maine Yankee	X	X	X	X	X					
12/72	Surry 1	X	X	X	X	X					
5/73	Surry 2	X	X	X	X	X					
9/73	Fort Calhoun 1	X	X	X	X						
6/74	Kewaunee	X	X	X							
9/74	Three Mile Isl. 1	X	X	X							
12/74	Arkansas 1	X	X	X							
4/75	Rancho Seco	-	X	X							
8/75	Cook	X	X								
12/75	Millstone Point 2	X	X								
5/76	Trojan	X									
12/76	St. Lucie	X									

NOTES:

1. Only PWRs operating at power levels > 450 MWe have been considered, with the exception of San Onofre 1 (430 MWe).
2. Multiunit plants have been excluded for the most part unless commercial operation dates are close together (example, Surry 1 & 2), or unless the additional units are not scheduled for commercial operation for approximately 2 years or more. (Example, Arkansas 1 & 2, Three Mile Island 1 & 2, St. Lucie, Davis-Besse 1, 2, & 3, Farley 1 & 2, and Salem 1 & 2).

TABLE 12.4-4 (Sheet 2)

3. Exposures reported to be in excess of 1,000 man-rem/year-unit are:

Reactor	Years of Operation at Year of Record	Man-Rem Dose
Palisades	2	1,109
Ginna	3	1,032
Surry 1 & 2	4	3,165
Ginna	5	1,224
Robinson 2	5	1,142
Surry 1 & 2	5	2,307

TABLE 12.4-5 DISTRIBUTION OF THE NUMBER OF PERSONNEL (>100 MILIREM/YR) ACCORDING TO WORK FUNCTION

Work Function	Percentage as Per NUREG-75/032 (for the year 1974) (Ref. 2)	Percentage as Per NUREG-0109 (for the year 1975) (Ref. 1)
1. Reactor operations	19.2	9.1
2. Routine maintenance	34.5	59.5
3. Inservice inspection	1.4	4.1
4. Special maintenance	28.7	16.4
5. Waste processing	2.1	7.6
6. Refueling	14.1	3.3

See notes at end of Table 12.4-7.

TABLE 12.4-6 DISTRIBUTION OF PERSONNEL (>100 MILLIREM/YR) ACCORDING TO EMPLOYEE CATEGORY

Category	Percentage as Per NUREG-75/032 (for the year 1974) (Ref 2)	Percentage as Per NUREG-0109 (for the year 1975) (Ref 1)
1. Station employees	47.4	34.7
2. Utility employees	18.1	6.1
3. Contract workers	34.5	59.2

See notes at end of Table 12.4-7.

TABLE 12.4-7 PERCENTAGES OF PERSONNEL DOSE BY WORK FUNCTION (REF 3)

Work Function	Percent of Dose			
	1974	1975	1976	1977
Reactor operations and surveillance	14.0%	10.8%	10.2%	10.6%
Routine maintenance	45.4%	52.6%	31.0%	28.9%
In-service inspection	2.7%	3.0%	6.0%	6.6%
Special maintenance	20.4%	19.0%	40.0%	41.4%
Waste processing	3.5%	6.9%	5.0%	5.9%
Refueling	14.0%	7.7%	7.9%	6.6%

Notes on Tables 12.4-5, 6 and 7:

1. PWRs and BWRs operating at all power levels were considered in compiling these tables.
2. Percentages for 1974 and 1975 are based on approximately 39 percent and 50 percent of the total exposures reported in the appropriate year for light water reactors which had been in commercial operation for at least one full year, as of 12/31/74 and 12/31/75, respectively.
3. Percentages for 1976 and 1977 are on only those facilities which have been in commercial operation for at least one full year, as of 12/31/76 and 12/31/77, respectively.
4. Distributions of personnel receiving an annual exposure of greater than 100 millirems, according to either work function or employee category, are not available for 1976 or 1977.

TABLE 12.4-8 ANNUAL OCCUPATIONAL EXPOSURES FOR VARIOUS PWR VENDOR'S UNITS (REF. 1,2,3)

	Westinghouse		C-E		B&W		
Approx Yr. of Operation	No. of Reactors	Man-Rem Yr-Unit	No. of Reactors	Man-Rem Yr-Unit	No. of Reactors	Man-Rem Yr-Unit	Comments
1	7	149	5	117	2	47	---------------
2	8	243	4	522	3	211	CE: Palisades - 1,131 $\frac{man-rem}{yr}$
3	7	608	3	419	3	335	W: Ginna - 1,032 $\frac{man-rem}{yr}$ CE: Palisades produces very little power.
4	6	741	3	229	---	---	W: Surry 1 & 2 - 3,165 $\frac{man-rem}{yr}$
5	6	876	2	471	---	---	W: Ginna 1 - 1,224 $\frac{man-rem}{yr}$ Robinson 2 - 1,142 $\frac{man-rem}{yr}$ Surry 1 & 2 - 2,307 $\frac{man-rem}{yr}$
6	4	553	1	100	---	---	---------------
7	4	341	---	---	---	---	---------------
8	3	443	---	---	---	---	---------------
9	2	665	---	---	---	---	W: San Onofre - 880 $\frac{man-rem}{yr}$
10	2	745					

Notes - On Table 12.4-8:

1. Only PWRs operating at power levels >450 MWe have been considered with the exception of San Onofre 1 (430 MWe).

2. The power plants for this table are the same as those appearing on Table 12.4-4.

TABLE 12.4-9 AVERAGE ANNUAL OCCUPATIONAL EXPOSURE AND POWER FOR INDIVIDUAL PWRS (REF 1, 2, 3)

Plant Name	Total Yrs of Operation (approx.)	Full Power MWe (net)	Annual Average Percentage of Full Power	Annual Ave. Man-Rem	Comments
Conn. Yankee	10	575	79	462	First year data not avail. 1973 - only 50 percent of full power was produced
San Onofre 1	10	430	77	325	First year data not avail. Highest annual exposure: 1976 - 880 man-rem 1977 - 847 man-rem
Ginna	8	490	64	587	Percentage of full power (about 52%, 51%) was low in 1974 and 1976. Highest annual exposures: 1972 - 1,032 man-rem 1974 - 1,225 man-rem
Robinson 2	7	665	75	607	1971 - Only 53 percent of full power was produced. Highest annual exposure 1975 - 1,142 man-rem
Palisades	6	798	43.4*	490	*Average per 5 yrs 1974 - Only produced 1 percent full power. 1972 - operated at restricted levels - 27 percent full power produced. Highest annual exposure 1973 - 1,133 man-rem.
Maine Yankee	5	790	68	237	Percentages of full power produced were low in 1973 (52 percent) and 1974 (55 percent)
Fort Calhoun 1	4	457	63	244	1975- Only 55 percent of full power was produced.

TABLE 12.4-9 (Sheet 2)

Notes on Table 12.4-9:

1. Only single unit PWRs have been considered.

2. Full power data based on information given in "Commercial Nuclear Power Plants" by NUS Corp., 1978, Ed. 10.

3. Robinson full power is based on a summertime value.

4. Only plants which have been in commercial operation for equal to or greater than 4 years were considered.

TABLE 12.4-10 AVERAGE INDIVIDUAL EXPOSURE BASED ON PWR PLANT AGE (REF. 1,2,3)

Year of Operation (approx)	No. of Units	Ave. No. of Personnel/Unit	Ave. Exposure man-rem yr-unit	Ave. Exposure per Individual (rem-yr)
1	13	362	123	0.34
2	15	532	330	0.62
3	13	661	664	1.00
4	9	623	571	0.92
5	8	691	775	1.12
6	5	627	481	0.77
7	4	540	341	0.63
8	3	583	465	0.80
9	2	987	665	0.67
10	2	940	745	0.79

Notes on Table 12.4-10:

1. Units considered are the same as those shown on Table 6, except that Palisades has been omitted from the first year average since no data is available for the number of personnel for that year.

2. The high exposure average for years 3-5 include the following:

Plant Year	Ginna Average Exposure	Surry 1 & 2 Average Exposure	Robinson Average Exposure
3	1.52	---	---
4	---	1.15	---
5	1.39	1.24	1.35
	Rem/yr	Rem/yr	Rem/yr

TABLE 12.4-11 CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION - PWRS (REF. 1, 2, 3)

Years of Operation Plant	1	2	3	4	5	6	7	8	9	10
Conn. Yankee	-	106	398	379	366	432	394	438	439	462
San Onofre 1	-	42	99	82	126	171	155	174	226	327
Ginna	207	319	556	478	627	613	616	589		
Robinson 2	364	290	425	484	618	634	608			
Palisades	78	606	613	536	568	490				
Maine Yankee	117	269	285	235	237					
Surry 1 & 2	152	518	895	463	1631					
Fort Calhoun 1	71	183	226	244						
Kewaunee	28	149	146							
Three Mile Island 1	73	180	240							
Arkansas 1	21	155	189							
Rancho Seco	-	58	225							
Cook	116	208								
Millstone Pt 2	168	205								
Trojan	174									
St. Lucie	152									
Total	1721	3288	4297	3904	4173	2340	1773	1201	701	789
No. of Units	14	15	13	9	8	5	4	3	2	2
Average	123	219	331	434	522	468	433	400	351	395

Note:

1. Plants considered are the same as those shown on Table 12.4-4.

TABLE 12.4-12 ESTIMATES OF OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA DOSES TO PLANT PERSONNEL

Operation	Zone	Percentage of Occupancy	Hrs/yr	Hourly Dose Rate (Rems/Hr)	Yearly Dose Rate (Rems/Yr)	No. of men	Annual Exposures (Man-Rem/yr-Unit)
Operation & Surveillance	A	75	1,560	1×10^{-4}	0.15	38	5.7
	B	21	437	5×10^{-4}	0.21	38	8.3
	C	3	62	2.5×10^{-3}	0.15	38	5.9
	D	0.9	19	1.5×10^{-2}	0.28	38	10.8
	E	0.1	2	1×10^{-1}	0.20	38	7.6
Total		100	2,080		0.99		38.3
Routine Maintenance	A	75	1,560	1×10^{-4}	0.15	28	4.4
	B	11	229	5×10^{-4}	0.11	28	3.2
	C	10.5	218	2.5×10^{-3}	0.54	28	15.3
	D	2.5	52	1.5×10^{-2}	0.78	28	21.8
	E	1	21	1×10^{-1}	2.1	28	58.8
Total		100	2,080		3.68		104
Radwaste Processing	A	70	1,456	1×10^{-4}	0.14	8	1.16
	B	20	416	5×10^{-4}	0.2	8	0.68
	C	8	166	2.5×10^{-3}	0.41	8	3.28
	D	1	21	1.5×10^{-2}	0.31	8	2.52
	E	1	21	1×10^{-1}	2.1	8	16.8
Total		100	2,080		3.16		24.4
Refueling	A	30	48	1×10^{-4}	0.004	6	0.22
	B	40	64	5×10^{-4}	0.03	6	1.47
	C	24	39	2.5×10^{-3}	0.09	6	4.48

TABLE 12.4-12 (Sheet 2)

Operation	Zone	Percentage of Occupancy	Hrs/yr	Hourly Dose Rate (Rems/Hr)	Yearly Dose Rate (Rems/Yr)	No. of men	Annual Exposures (Man-Rem/yr-Unit)
	D	4	6	1.5×10^{-2}	0.09	6	4.14
	E	2	3	1×10^{-1}	0.30	6	13.8
Total		100	160		0.51		24.1
Inservice Inspection (ISI)	A	55	132	1×10^{-4}	0.01	46	0.6
	B	36	86	5×10^{-4}	0.04	46	1.96
	C	7.0	17	2.5×10^{-3}	0.04	46	1.96
	D	1.0	2.5	1.5×10^{-2}	0.03	46	1.72
	E	1	2.5	2×10^{-1}	0.5	46	23
Total		100	240		0.62		29.2
Special Maintenance	A	75	240	1×10^{-4}	0.024	200	4.8
	B	20	64	5×10^{-4}	0.03	200	6.4
	C	3	10	2.5×10^{-3}	0.025	200	5
	D	1	3	1.5×10^{-2}	0.04	200	9
	E	1	3	2×10^{-1}	0.6	200	120
Total		100	320		0.71		145

TABLE 12.4-13 DISTRIBUTION OF DIRECT RADIATION MAN-REM DOSES ACCORDING TO WORK FUNCTIONS

Operation	Annual Exposures (Man-rem/year Unit)	Percentage
Operation and surveillance	38.3	10.5
Routine maintenance	104	28.5
Radwaste processing	24.4	6.7
Refueling	24.1	6.6
Inservice inspection	29.2	8
Special maintenance	145	39.7
Total	365	100

TABLE 12.4-14 ANNUAL OCCUPANCY IN PLANT AREAS CONTAINING AIRBORNE RADIOACTIVITY

Building	hr/yr per man	No. men	man-hours yr
Containment			
-power	250	3	750
Containment			
-refueling	62.5	32	2,000
Fuel building			
-power	250	2	500
Fuel building			
-refueling	125	14	1,750

TABLE 12.4-15 DOSES TO PLANT PERSONNEL CAUSED BY AIRBORNE RADIOACTIVITY

Location	Dose Rate mrem/hr			Annual Occupancy (man-hr/yr)				Annual Expose (man-rem/yr)		
	Thyroid	Lung	Whole Body	Operation & Surveillance	Maintenance	Refueling	Total	Thyroid	Lung	Whole Body
Containment - power	8.64	0.05	0.48	100	650	---	750	6.48	0.04	0.36
Containment - refueling	1.34	0.93	N	---	1000	1000	2000	2.68	1.86	N
Fuel building - power	0.62	0.62	N	400	100	---	500	0.31	0.31	N
Fuel building - refueling	4.71	3.28	N	---	---	1750	1750	8.25	5.74	N
Total								18	8	0.4

N = Negligible

12.5 RADIATION PROTECTION PROGRAM

For descriptions of the radiation protection program, refer to Section 12.5 of the Site Addendum.

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RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA)

12.1.1 POLICY CONSIDERATIONS

12.1.1.1 Management Policy

Union Electric Company is committed at the corporate level to maintaining occupational radiation as low as reasonably achievable. This policy is promulgated in general terms by the Senior Vice President and Chief Nuclear Officer and is implemented by specific procedures and directives at the plant level by the Senior Director, Nuclear Operations. Implementation of the ALARA program will be in accordance with the requirements of 10CFR19 and 10CFR20 and will follow the intent of the guidance provided in applicable sections of Regulatory Guides 1.8, 8.8 and 8.10.

12.1.1.2 Organizational Structure

The Senior Director, Nuclear Operations, through the Radiation Protection Department staff, is ultimately responsible for development, implementation, and continuing conduct of the overall plant Radiation Protection program of which the ALARA program is a part. The Manager, Radiation Protection directs the Radiation Protection Department and is responsible to the Plant Director for the ALARA program. The Manager, Radiation Protection has the responsibility and authority for developing, implementing and managing the plant ALARA program.

The organization of the Radiaton Protection Department is described in Section 12.5.1.

ALARA reviews can be performed by qualified nuclear engineering personnel (i.e., Design Engineering, Technical Support Engineering, Systems Engineering, or Radiation Protection) for plant modifications or maintenance activities.

12.1.1.3 Application of the Union Electric ALARA Policy to Plant Operations

The Callaway Plant ALARA program is implemented through the adherence to Administrative and Radiation Protection procedures which cover, but are not limited to, the following subjects:

a. Qualifications of and the conduct of operations of the plant radiation protection department.

b. ALARA program administration - reviewing job planning, job exposure data, radiation work permit comparisons with personnel exposure, procedure reviews, review of design change packages for installation and long term operations and maintenance concerns, and trends of dose data for potential programmatic changes.

c. Training of personnel in the fundamentals of radiation protection and in health physics exposure control procedures.

d. Radiation Work Permit Program

e. Internal Personnel Monitoring

f. External Personnel Monitoring

g. Area Posting

h. Radioactive Material Control

i. Radiological Surveys

j. Instrumentation

k. Radiological Incidents

l. Radiation Work Practices

m. Environmental Monitoring

n. Respiratory Protection

12.4 DOSE ASSESSMENT

This section contained the dose assessment to the construction workers during the construction of Callaway Plant Unit 2. With the cancellation of Callaway Plant Unit 2 in October 1981, this section is no longer applicable.

12.5 RADIATION PROTECTION PROGRAM

12.5.1 ORGANIZATION

12.5.1.1 Radiation Protection Program Objectives

The objective of the radiation protection program for the Callaway Plant is to maintain the occupational radiation exposure of personnel working at the plant as low as is reasonably achievable (ALARA). This objective is accomplished by adherence to the requirements of Title 10 of the Code of Federal Regulations, Part 20, by following the intent of guidance given in USNRC Regulatory Guides 8.2, 8.8, 8.10, 8.34, 8.35, 8.36 and 1.8, and by using industry accepted radiation protection practices.

The radiation protection program is administered by the organization described in subsections 12.5.1.2 and 13.1.2.2.2 to accomplish the following:

a. Identify and review radiation protection training requirements for personnel assigned to work in radiological controlled areas commensurate with their duties, responsibilities, and the degree of radiation hazards anticipated. Inform these personnel of methods for maintaining occupational radiation exposure ALARA and assist them in carrying out their radiation safety responsibilities.

b. Evaluate and review the radiological status of the plant by monitoring radiation, contamination and airborne radioactivity levels to control or eliminate radiological hazards.

c. Control external and internal radiation exposure of personnel through the implementation of radiological controls during operations and maintenance.

d. Review and evaluate radiation protection records and appropriate plant operating and maintenance procedures for methods to reduce radiation exposure to personnel.

e. Maintain and evaluate personnel exposure records to ensure that occupational radiation exposures are maintained ALARA.

f. Maintain control of radioactive materials on-site and maintain releases of radioactive materials in effluents to unrestricted areas ALARA.

g. Evaluate and review the swipe samples from instruments containing radioactive sources that other Union Electric fossil power plants have provided.

12.5.1.2 Organization of the Radiation Protection Department

The experience and qualifications of the personnel responsible for administering the radiation protection program are presented in Section 13.1.3.1. The responsibilities of the personnel are discussed in Section 13.1.2.2.2.

The Radiation Protection Department is comprised of personnel in the following classifications:

a. Manager, Radiation Protection

b. Supervising Health Physicist

c. Health Physicists/Nuclear Scientists

d. Radiation Protection Supervisors

e. Radiation Protection Technicians

f. Radiation/Chemical Helpers

12.5.1.3 Personnel Training

The training of plant personnel in radiation protection is described in Section 12.5.3.4.

12.5.2 EQUIPMENT, INSTRUMENTATION AND FACILITIES

12.5.2.1 Equipment and Instrumentation

Radiation Protection instrumentation and equipment will be available for the assessment of plant radiological conditions and to support the operation of the Callaway Plant Radiation Protection Program. The three classifications of equipment that will be utilized in radiological monitoring and surveillance activities include:

a. Installed Equipment

b. Laboratory Equipment

c. Portable Equipment

Installed equipment includes both the in-plant area radiation monitoring system and the process and effluent radioactivity monitoring system. Laboratory equipment consists of analytical instrumentation used to analyze and quantify radioactivity. Portable equipment includes portable radiation detection instrumentation, air samplers and personnel dosimetry used for performing radiation and contamination surveys, airborne radioactivity monitoring and sampling and personnel monitoring.

Portable and laboratory instrumentation used in the radiation protection program will be selected in accordance with the following criteria:

a. Ability to measure the quantity of interest to an acceptable degree of precision and accuracy.

b. Ease of operation, maintenance, and calibration.

c. Appropriate sensitivity and range for various operational situations, including normal operations, anticipated operational occurrences and accident conditions, as determined by the requirements of applicable regulations.

d. Operational reliability.

12.5.2.1.1 Installed Radiation Monitoring Equipment

Installed area radiation monitors and continuous air monitors are described in Section 12.3. Process and effluent monitors are described in Section 11.5.

12.5.2.1.2 Radiation Protection Laboratory Instrumentation

Radiation protection laboratory instrumentation will include a gamma spectroscopy system, a liquid scintillation counter, a gas flow proportional counter and a geiger-mueller type counting system. Additional information on radiation protection laboratory equipment is provided in Table 12.5-1. The primary location of radiation protection laboratory equipment will be the count room, 1984' elevation of the Control Building. However, additional counting equipment will be available at other plant locations including the Emergency Operations Facility.

12.5.2.1.3 Portable Radiation Detection Instrumentation

Portable radiation detection instrumentation is described in Table 12.5-2 which includes information on the type, range, accuracy and typical quantities of instruments. Survey instruments will be available in-plant to facilitate access. Instruments requiring calibration or repair will be identified as out-of-service and segregated from in-use equipment.

12.5.2.1.4 Portable Air Sampling Equipment

Portable air sampling equipment includes high and low-volume air samplers capable of accepting particulate filters and charcoal cartridges for grab samples of radioactive particulates and iodine. In addition, portable continuous air monitors, typical of that described in Table 12.5-2, will be used for continuous surveillance of airborne radioactivity levels. Air sampling instruments are calibrated periodically in accordance with an established calibration program.

12.5.2.1.5 Instrument Calibration

Calibration of portable and laboratory radiation protection instrumentation will be performed in accordance with an established calibration program. Requirements of this program include as a minimum:

a. written procedures for operation and calibration of each instrument.

b. a mechanism for tracking and recall of instruments for calibration.

c. established calibration frequencies for each instrument and recalibration following maintenance and repair that could change the instrument performance characteristics established during the previous calibration.

d. use of calibration standards which are traceable to the National Institute of Standards and Technology (NIST).

e. required calibration accuracies and tolerances.

f. periodic response checking of instruments to verify continuing proper operation.

g. maintenance of records documenting calibration activities.

h. tagging and labeling of instruments.

12.5.2.1.6 Protective Clothing

Protective clothing is prescribed and issued to plant personnel by the radiation protection staff based upon the actual or potential radiological conditions expected for the job assignment. Protective clothing stations are established at strategic locations within the plant as required to ensure efficient operations and to preclude the spreading of contamination. Protective clothing available at the plant includes the following:

a. Coveralls.

b. Caps and hoods.

c. Shoe covers.

d. Plastic, rubber, and cotton gloves.

e. Plastic suits.

f. Face shields/masks

12.5.2.1.7 Respiratory Protection Equipment

Respiratory protection equipment is available to plant personnel and is also prescribed and issued to individuals as required by actual or potential radiological conditions of the work assignment. The Callaway Plant Respiratory Protection Program follows the guidance of Regulatory Guide 8.15 and complies with the requirements of 10 CFR Part 20. Respiratory devices available at the Callaway Plant include the following:

a. Full-face respirators with high-efficiency particulate and/or charcoal filters.

b. Full-face respirators with supplied air.

c. Hoods with supplied air.

d. Full-face respirators in self-contained breathing apparatuses.

e. Half-face respirators with HEPA and/or charcoal filters.

Purchases of respiratory protection equipment such as air purifying respirators, supplied air respirators, self-contained breathing apparatuses and accessory equipment are made following the guidance given by 30 CFR Part 11 and the NIOSH Certified Equipment Manual.

12.5.2.1.8 External Dosimetry

All personnel entering the plant's Radiological Controlled Area (RCA), are monitored for occupational radiation dose in accordance with the requirements of 10CFR20. Optical Stimulated Luminescent Dosimetery (OSL) is the primary method of monitoring occupational radiation dose from exposure to beta and photon radiation. CR-39 is the primary method of monitoring occupational radiation dose from exposure to neutron radiation. Primary dosimetry is processed by a vendor that is accredited by the National Voluntary Laboratory Accrediation Program (NVLAP) in accordance with the requirements of 10CFR20. An appropriate secondary monitoring device is issued to each individual who enters the RCA. Portable neutron survey instruments and stay time calculations are used as a secondary method for determining neutron dose. Exposure records for each individual will be maintained in accordance with Regulatory Guide 8.7.

12.5.2.1.9 Internal Dosimetry

Internal radiation exposure is assessed using either in vivo counting, specimen analysis or calculational techniques based on surveillance program data. The methodology for internal radiation exposure assessment will follow the guidance of the International Commission on Radiological Protection (ICRP). The concepts, models, equations and assumptions used for internal radiation exposure assessment will reflect the appropriate recommendations of the ICRP.

An in vivo counter is located at the plant for measurement of plant personnel, visitors, or support personnel. The in vivo counter will provide preliminary background information, periodic evaluation, and emergency capability for detecting internal exposure conditions. Assessment of internal radiation exposure of those individuals who regularly enter areas where the potential exists for inhalation, ingestion, or absorption of radioactive material will be performed annually.

12.5.2.2 Facilities

12.5.2.2.1 Radiation Protection Facilities

The radiation protection facilities consist of a radiation protection office, counting room, hot laboratory, and personnel decontamination shower located at elevation 1984' of the Control Building.

12.5.2.2.2 Access Control Facilities

Radiation protection facilities at the 1984' level of the Control Building are designed to function as the primary access point for entry to the radiological controlled areas of the plant. The plant arrangement is such that traffic between radiological controlled areas and uncontrolled areas is routed through this access control area. The purpose of the access control point is to provide positive control over access to radiological controlled areas for exposure control purposes and to prevent the spread of contamination to uncontrolled areas of the plant. Contamination control features of access control include controlled entrance and exit locations with contamination monitoring provisions at the exit of the radiologically controlled area.

In addition to the primary access control point described above, auxiliary radiological control points can be established at necessary locations within the plant for the purpose of personnel and contamination control. These radiological control points are equipped with protective clothing, portable survey equipment and other radiation protection materials and are established at locations of strategic importance for contamination and exposure control.

12.5.2.2.3 Equipment Decontamination Facilities

Equipment used for the cleaning and maintenance of contaminated parts, instruments and equipment is located at elevation 2000' adjacent to the Auxiliary Building. When hand decon is not practical (such as for high level of contamination or ALARA constraints), special cleaning equipment will be used for removing radioactive contamination from items requiring repair or maintenance.

12.5.2.2.4 Personnel Decontamination Facilities

A personnel decontamination area separated from normal showering facilities is provided on the 1984' level of the Control Building in the access control area. Facilities provided include decontamination sinks, shower, and a drying and monitoring area.

12.5.2.2.5 Radiation Protection Count Room

A counting room is located near access control and is used for radioactivity analyses. A separate hot laboratory is provided for sample preparation. The count room is equipped to perform routine analyses required for personnel protection, surveys, and related radiation protection functions. The counting room is equipped with the necessary instrumentation to perform routine counting on plant radioactivity samples (water, air, swipe survey, etc.).

12.5.2.2.6 Laundry Decontamination Facility

The laundry decontamination facility is located on the 2000 elevation adjacent to the Auxiliary Building. It is equipped with washer-extractors, dryers, a clean laundry monitor and clothes sorting tables.

12.5.2.2.7 Radiation Protection Office

The radiation protection office is located adjacent to the entrance to the access control area and will allow radiation protection personnel to observe personnel entering and leaving the radiological controlled area. The radiation protection office provides work space for radiation protection personnel to perform routine administrative activities such as completing survey records and generating RWP's. It also provides a location from which plant radiation protection activities can be coordinated and plant personnel can obtain information concerning work activities or plant radiological conditions.

12.5.2.2.8 Locker and Change Facilities

Change areas consisting of lockers and benches are provided within the plant. A storage area for protective clothing is provided adjacent to the change areas, thus allowing personnel to dress-out for activities near the contaminated areas.

12.5.2.2.9 Calibration Facilities

Instrument calibration facilities are provided for the calibration, routine maintenance and storage of portable radiation protection instrumentation. The radiation protection calibration facilities will be equipped with fixed and portable calibration sources of sufficient range and strength to allow calibration of most portable radiation detection instruments and personnel dosimetry devices used at the Callaway Plant.

12.5.2.2.10 Dosimetry Processing Facilities

Primary dosimetry is processed by a vendor.

12.5.2.2.11 Respiratory Protection Facilities

On-site facilities set aside for respiratory protection functions include areas for conducting quantitative fit testing and an area for administering pulmonary function testing as part of the medical qualification program for respirator users.

Other on-site areas are set aside for washing, drying, repair and storage of respirators which are used for supporting day to day work. The plant warehouse serves as the bulk storage point for receiving and storing respiratory protection equipment held in reserve for future use.

12.5.2.2.12 Respirator Issue and Storage Area

A Respirator Issue Area is provided within the Access Control Area for issuing of respirators to individuals as required by actual or potential radiological conditions of the work assignment.

12.5.3 PROCEDURES

The radiation protection administrative, departmental and technical procedures used at the Callaway Plant are an integral part of the plant ALARA program. The provisions and guidance of Regulatory Guides 8.2, 8.7, 8.8, 8.10, 8.13, 1.8, and 1.39, are used in the development of the plant radiation protection procedures. Exception to Reg. Guide 1.33 has been submitted under separate correspondence to the NRC.

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

12.5.3.1 <u>Radiation Surveys</u>

Radiological surveys are performed periodically during operation and shutdown at the Callaway Plant for the assessment of radiation-fields, radioactive contamination, and airborne radioactivity levels. Portable instruments, equipment, and techniques for surveys are addressed in department and technical procedures.

12.5.3.1.1 Radiation - Surveys

Routine measurements of radiation field intensities are performed in accessable areas of the plant, using portable instrumentation appropriate for the type of radiation present. These surveys are performed on a schedule which is determined by:

a. Actual or potential radiation levels.

b. The variability of radiation level.

c. The occupancy factor of the location.

Radiation surveys are conducted to monitor and detect any significant changes in radiation levels and to evaluate the effectiveness of radiological controls. High radiation areas are usually surveyed upon entry and periodically thereafter while work in the area is in progress. Additional radiation surveys are performed as necessary to evaluate and minimize personnel radiation exposure during operational and maintenance functions.

Records are maintained of the results of these surveys, by location, so that trends in the radiation level are readily identified. Results of these radiation surveys are correlated with the readings of the Area Radiation Monitoring System, where appropriate.

Prior to the initiation of any operation for which a radiation work permit is required, a survey will be made of the radiation field in the vicinity where the operation is to be performed. The results of this survey are recorded on the radiation work permit.

12.5.3.1.2 Surface Contamination Surveys

Contamination surveys are performed on a regularly scheduled basis in all accessible areas during operation and shutdown to evaluate the hazard due to removable and non-removable radioactive contamination. Locations of importance for controlling the potential spread of contamination are surveyed at a frequency commensurate with the contamination hazard present using the "smear" technique or an appropriate portable instrument. These survey frequencies could increase or decrease depending on factors such as the actual or potential radioactive concentration, occupancy factor, location, and plant status.

Contamination surveys are also made on personnel, equipment, and materials from time to time as necessary to ensure complete control over the levels and spread of removable contamination. Appropriate techniques, instruments, and frequency of surveys will be delineated in the radiation protection procedures.

The results of these surveys are recorded and tabulated by location so that trends in the data may be readily observed.

12.5.3.1.3 Airborne Radioactivity Surveys

Surveys to assess airborne radioactivity levels are performed to ensure 10 CFR Part 20 limits are not exceeded, engineering controls are functioning, and respiratory protection techniques are adequate.

The results of routine airborne radioactivity surveys are recorded by location so that trends in the radiation levels can be readily identified. Air sampling surveys are performed on a routine basis in areas with both a high occupancy factor and a high potential for the existence of airborne radioactivity. Other locations are surveyed periodically depending on the significance of the location as a source of occupational radiation exposure from airborne radioactivity. These survey frequencies could increase or decrease depending on factors such as the actual or potential radioactive concentration, occupancy factor, location, and plant status.

Airborne samples are collected and analyzed for operations and maintenance activities which are expected to produce airborne radioactivity.

12.5.3.2 Radiation Exposure Control

In order to control occupational radiation exposures and the spread of radioactive contamination, varying degrees of access to plant areas will be established. Area access and exposure time for personnel will be determined by the necessity for such access and by radiation and contamination levels.

Areas of the plant are subject to access control restrictions proprortionate to the potential for radiation exposure in each area. The particular access control requirements for each area will be specified by Radiation Protection Supervision. A radiological controlled area is any area where actual or potential radiological hazards exist in the form of radiation, contamination, airborne radioactivity, or stored radioactive materials. Radiation areas, contamination areas, high radiation areas, very high radiation area, airborne radioactivity areas and radioactive material areas will be considered radiological controlled areas. Radiological controlled areas will be posted and controlled in accordance with the requirements of 10CFR20. Access to such areas is controlled as warranted by the degree of radiological hazard involved via use of access control, auxiliary radiological control points and the radiation work permit program.

The plant arrangement is such that traffic between radiological controlled areas and uncontrolled areas is routed through access control. The access control facilities for normal traffic are located in the control building where people entering and leaving all controlled areas will check for contamination. Within the controlled area, access is controlled by utilizing locked and/or annunciated doors and gates, fences, alarms, rope barriers, and the posting of signs.

High radiation areas in which the radiation field is greater than 100 mrem/hr but less than 1000 mrem/hr will be posted as a high radiation area and entrance into the area will be controlled by the radiation work permit program. High radiation areas in which the radiation field is greater than or equal to 1000 mrem/hr will be provided with locked doors to prevent unauthorized entry. For individual areas accessible to personnel that are located within large areas, such as the Containment, where no enclosure exists for purpose of locking, and where no enclosure can be reasonably constructed, the individual area shall be barricaded, conspicuously posted and a flashing light activated as a warning device. The flashing light may be omitted if positive control of access to the area is provided.

Very high radiation areas with radiation levels in excess of 500 rads in one hour at one meter from the radiation source will be conspicuously posted as a very high radiation area. Entrances to these areas will be barricaded and locked.

Administrative measures for access control in high radiation areas include the requirement for issuance of a radiation work permit for any operation to take place in such an area. Depending upon the operation, a radiation protection technician may be assigned to supervise stay-time and make appropriate surveys while the operation is in progress.

The radiation work permit (RWP) program will be established as an integral part of the ALARA policy implementation. This program will be the responsibility of the Manager, Radiation Protection. In addition to entries into and work performed in high radiation areas, RWP's may be required for the following type activities:

a. Entries into designated airborne radioactivity areas, radiation areas, contamination areas and radioactive material storage areas.

b. Work involving maintenance or other adjustments to any system or component which contains, stores, transports or collects radioactive materials.

c. Other activities which in the judgement of Radiation Protection Supervision warrant the issuance of a radiation work permit prior to initiating the operations.

The radiation work permit stipulates the purpose of entry, work location, radiation conditions, surveillance and dosimetry requirements, stay-time, protective clothing and equipment, and other procedural requirements and precautions.

The primary objectives of the radiation work permit program are to insure that:

a. Radiological conditions are known as accurately as possible.

b. Proper protective measures are taken to safely perform the required duties.

c. Each person involved in these operations acknowledges his understanding of the radiation conditions, the protective and safety measures required, and his willingness to follow the radiation work permit requirements.

d. Appropriate supervisors are aware of the task being performed, the radiation conditions, and the prescribed protective measures.

e. A means for maintaining the accountability of personnel is provided.

12.5.3.3 Contamination Control

Areas that may be contaminated with radioactive material will be decontaminated to a level that is reasonably achievable using available methods and techniques. Since the complete removal of surface contamination from parts of the plant is a practical impossibility, certain plant areas may be designated as "contamination areas". These areas will be posted with the proper warning placards and barricaded. Entry to these areas will be controlled by radiation protection personnel and allowed only through the issuance of an appropriate radiation work permit. Personnel, equipment, and material exiting from contaminated areas will be monitored to prevent the spread of contamination to clean areas. All contaminated equipment will be properly packaged and identified before removal from a contaminated area. At access control facilities near the radiation protection office, a final survey will ensure that all personnel, material, and equipment are free of significant contamination, and thus will provide assurance that no radioactive material will spread to the uncontrolled areas of the plant.

12.5.3.4 Radiation Protection Training

The Manager, Technical Training is responsible for the radiation protection training program at the Callaway Plant. Plant personnel, both permanently assigned and temporary, receive training in the principles of radiation protection commensurate with the individual's job function and the anticipated radiation hazards.

The radiation protection training program will include as a minimum the following topics:

a. Fundamentals of radiation and radioactivity.

b. Biological effects of radiation on humans.

c. Measurement of radiation and radioactivity.

d. Principles and techniques of radiation protection.

e. Use of protective clothing and equipment.

f. General regulatory and specific facility license radiation protection requirements.

g. Emergency planning.

h. ALARA program - concepts and methods.

The radiation protection training program will maintain the proficiency of these employees through periodic retraining lectures and exercises.

12.5.3.5 Personnel Dosimetry

12.5.3.5.1 External Radiation Dosimetry

See Section 12.5.2.1.8.

12.5.3.5.2 Internal Radiation Dosimetry

See Section 12.5.2.1.9.

12.5.3.6 Airborne Radiation Evaluation and Control

Airborne radioactivity will be routinely assessed using local sampling, portable continuous air monitors, and the fixed radiation monitoring system. Airborne radioactive materials (particulates, noble gases, halogens, tritium) will be sampled and analyzed using appropriate techniques. Since local sampling will provide better estimates of airborne contamination levels existing in a work area than will a monitor reading, such special air sampling will be used in the radiation work permit program to keep radiation exposure due to airborne radioactivity ALARA. Portable continuous air monitors and fixed airborne radioactivity monitors will be used to provide alarm indications and additional information which will be used with local sampling for the assessment of airborne radioactivity.

Control of airborne radioactivity levels will be assured through the use of the plant's heating, ventilation, and air conditioning (HVAC) systems, portable air movers and filters. The HVAC systems provide controlled air movement and filtration for those areas with a high potential for airborne radioactivity problems. Special control techniques can be used, such as plastic enclosures which isolate and vent airborne radioactivity arising from special work projects. Respiratory protection equipment will be available for use in those situations where airborne radioactivity hazards exist and other control measures are inadequate at the location and time.

12.5.3.7 Respiratory Protection Program

The respiratory protection program will be developed through the guidance of Regulatory Guide 8.15 and will satisfy the requirements of 10CFR20. As a minimum, the respiratory protection program will provide the following:

a. Procedures to implement the selection, use, maintenance and storage of respiratory protection devices.

b. Adequate facilities to support the storage, issue, cleaning and maintenance of respiratory protection devices.

c. Use of only NIOSH certified or NRC approved equipment.

d. Medical certification program for respirator users.

e. Periodic review of the overall effectiveness of the respiratory protection program via the internal dosimetry program.

12.5.3.8 <u>Radioactive Materials Handling</u>

Methods and procedures will be developed to control, handle, and store by-product, source, and special nuclear material in accordance with regulatory requirements.

Subjects to be covered by these procedures will include the following:

a. Storage of radioactive material in appropriately shielded, labeled and secured containers.

b. Minimizing the distance that radioactive samples are transported by personnel and the use of special extension and remote handling tools when applicable.

c. Use of shielded sample transporters as appropriate

d. Periodic testing to verify the integrity of the sealed material.

e. Emergency procedures which detail the proper actions to be taken in the event of leakage and spills.

f. Accountability of any by-product or special nuclear material.

TABLE 12.5-1 RADIATION PROTECTION LABORATORY EQUIPMENT

INSTRUMENT	RADIATION DETECTED	DETECTOR	TYPICAL QUANTITY	LOCATION	REMARKS
Gamma Spectroscopy System	Gamma	HPGe w/graded shield	4	Counting Room and/or Maintenance Training Annex/ Operations Support Facility	Provides Gamma Isotopic analysis capabilities
Gas Proportional Counter	Alpha, Beta, Gamma	Gas Flow Proportional	1	Counting Room	Used for counting smears, effluent and radio chemistry samples.
Liquid Scintillation Counter	Beta	---	1	Counting Lab	Tritium determination
In Vivo Counter	Gamma	NaI (TI) and/or HPGe	1	In Vivo Count Room Central Processing Facility	Personnel In Vivo Counting; Multidetectors

Note: Sensitivities of this analytical instrumentation are dependent upon counting parameters such as sample geometry, count time and background, however, equipment from various manufacturers will be evaluated to ensure that the equipment when purchased will be sufficiently sensitive to perform the tasks for which it was intended. Instruments will be selected to satisfy measurement and reporting requirements in effect at the time the equipment is ordered.

TABLE 12.5-2 PORTABLE RADIATION PROTECTION EQUIPMENT

TYPE	TYPICAL DETECTOR TYPE[1]	APPROXIMATE RANGE	ACCURACY	TYPICAL QUANTITY	LOCATION
Portable Count Rate Meter/ Frisker	Geiger-Mueller/ Scintillation	0-500,000 CPM	± 10% FS	10	See Note 3
Low-Range Survey Meter	Geiger-Mueller/ IIonization Chamber	0-2 R/hr/ 0-5 R/hr	± 10% FS	25	
High-Range Extendable Probe Survey Meter	Geiger-Mueller	0-1000 R/hr	± 10% FS	4	
Mid-Range Survey Meter	IGeiger-Mueller/ IIonization Chamber	0-50 R/hr 0-100 R/hr	± 10% FS	5	
High-Range Survey Meter	IGeiger-Mueller/ IIonization Chamber	0-1000 R/hr IIndication	± 20% FS	3	
Neutron Survey Meter	BF Tube Inside Polyethylene Sphere	0.2 mrem/hr 10 rem/hr	± 10% FS	4	
Ultra-High Range Survey Meter	Geiger-Mueller/ IIonization Chamber	0-10,000 R/hr	± 10% FS	2	
Alpha Count Rate Meter/ Frisker	Scintillation	0-500,000 CPM	± 10% FS	5	
Portal Monitor/Personnel Contamination Monitor	Gamma Scintillation/ Gas Flow Proportional	Variable	-----	6	
Beta-Gamma Count Ratemeter/Frisker	Geiger-Mueller	0-500,000 CPM	± 10% FS	20	
Electronic Dosimeter	Solid State	0-100 R/hr/ 0-1000 R	± 10% FS	250	

TABLE 12.5-2 (Sheet 2)

TYPE	TYPICAL DETECTOR TYPE[1]	APPROXIMATE RANGE	ACCURACY	TYPICAL QUANTITY	LOCATION
Electronic Dosimeter Reader	Not Applicable	Not Applicable	Not Applicable	8	See Note 3
Calibration Transfer Instrument	IIonization Chamber	Various	± 5%	1	
Portable Area Monitor	IGeiger-Mueller/ Ionization Chamber/ Solid State	0-2 R/hr 0-100 R/hr	± 25% Max.	5	
Gamma Survey Meter	NaI	0-500k CPM	± 10%	2	
High Volume Air Sampler	Not Applicable	> 15 CFM	± 20%	5	
Low Volume Air Sampler	Not Applicable	0-3 CFM	± 20%	15	
Portable Continuous Air Monitor (3 channel type)	NaI (I-131); Beta Scintillation (Particulate and Gaseous)	1E-11 to 1E-5 (I-131 and particulate) 1E-11 to 1E-1 (gaseous)	± 20% of Indication	4	

Note 1: The detector types listed are typical of the instrumentation used. As instrumentation technology improves, an equivalent or better type of instrumentation may be substituted.

Note 2: The quantities listed for each type of instrumentation are the typical numbers calibrated and ready for use.

Note 3: Instrumentation is located in the RP Facilities on 1984' elevation of the Control Building, the Calibration Facilities, Dosimetry Processing Facilities or at specific work locations in the plant while in use.

Note 4: The above information is referenced in the Radiological Emergency Response Plan, Section 6.7.3, and complies with Regulatory Guide 1.97.