12.0 RADIATION PROTECTION

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

12.0 RADIATION PROTECTION

12.1 SHIELDING

12.1.1 DESIGN OBJECTIVE

The primary objective of the shielding design and access control is to protect operating personnel and the general public from potential radiation sources in the reactor, the radwaste system, and other auxiliary systems including associated equipment and piping.

Shielding is designed to perform the following functions:

A. Limit the dose to plant personnel, construction workers, vendors, and visitors during normal operation, including anticipated operational occurrences, to within a few percent of the guidelines of 10 CFR 20.1 - 20.601.

B. Limit the dose to plant personnel, in the unlikely event of an accident, to within the requirements of 10 CFR 50.67, to permit termination of accident conditions without undue risk to the general public.

C. Limit dose to certain components in high radiation areas and very high radiation areas within specified radiation tolerances.

D. Protect certain components to prevent excessive neutron activation and facilitate access.

E. Limit dose to persons at the boundary of the restricted area to a small fraction of the guidelines of 10 CFR 20.1 - 20.601 due to direct radiation during normal operation.

The following guides are used in shield design to achieve the above objectives. All plant areas are divided into zones according to the dose rates given below. These zones are for planning purposes; actual dose rates will be determined by surveys. As documented in NUREG-75/034, dated May 2, 1975, appropriate design features recommended by Regulatory Guide 8.8 (NRC acceptance criteria) have been included to maintain radiation exposures ALARA.

Zone Designation	Dose Rate (mrem/h)
I-A	≤ 0.2
I	≤ 0.5
II	≤ 2.5
III	≤ 15.0
IV-A	≤ 25.0
IV	≤ 100.0
V	>100.0

A. Access control and shielding design are considered according to the above guidelines in determining optimum plant layout that will allow personnel to perform their normal functions, based on required stay times to perform these functions, with the minimum of exposure.

B. All pipes and ducts penetrating the primary and secondary shields are located in positions so that a direct radiation shine from high radiation sources such as the reactor vessel and components of the reactor coolant loops is avoided.

Penetrations from all pipes and ducts through the shield walls are located to avoid a direct line of sight with the radiation source to prevent streaming into lower radiation zones. Grouting materials have been used to fill voids between the penetration and the wall where necessary.

C. Shield discontinuities include concrete hatch covers, shielding doors, and access labyrinths. To reduce radiation streaming through gaps between the main shield and a removable section, offsets have been used and the gaps are not in line of sight of the radiation source where this is feasible. Access labyrinths into rooms containing radiation sources such as gas decay tanks, coolant sampling equipment, evaporators, and filters have been designed to eliminate a direct shine through the offset passage to the accessible areas.

D. Radioactive piping is routed to minimize exposure to plant personnel. This is accomplished by:

1. Minimizing radioactive pipe routing through the corridors and low radiation zones.

2. Using shielded pipe trenches when the above method is not feasible.

3. Separating the locations of radioactive and nonradioactive pipes.

E. Motor-operated or diaphragm valves are used whenever feasible. Provision is made for drainage of associated equipment to minimize radioactive exposure during valve maintenance. In case of manual valves, provision is made to protect the operator from the radioactive valve by use of shield walls and valve stem extensions (reach rods).

F. Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions so that the personnel do not receive radiation exposures greater than 5 rem whole body or its equivalent to any part of the body for the duration of the accident, in accordance with 10 CFR 50, Appendix A, General Design Criterion 19.

The analysis of the control room doses, including ingress/egress, is covered in detail in chapter 15.

G. The principal shield material is concrete of a density of 145 lb/ft^3. Other miscellaneous material like steel, lead, high density concrete, or water are occasionally used as required.

H. The design shields major sources and activated components to allow access and provide adequate protection for inservice inspection.

I. Radioactive source data are based on full load plant operation with the equivalent of 1-percent fuel cladding defects.

12.1.2 DESIGN DESCRIPTION

12.1.2.1 General Description

Detailed drawings showing the layouts and cross-sections of buildings that contain process equipment for treatment of radioactive fluids are shown in figures 1.2-1 through 1.2-9. Drawing D-170084 is a detailed plot plan of the total plant layout within the site boundary, showing all outside storage areas and the location of the railroad siding.

Scaled isometric views and a layout drawing of the control room are illustrated in figures 1.2-1 and 12.1-1.

The radiation monitoring system functional block diagram is shown in drawing U-167647. The shield wall thickness, occupancy times, and maximum possible dose rates are shown in the radiation zones and controlled access diagrams (drawings D-176035, D-176036, D-176037, D-176038, D-176039, D-176040, D-176041, D-176042, D-176043, D-206035, D-206036, D-206037, D-206038, D-206039, D-206040, D-206041, D-206042, and D-206043).

Concrete radiation shields are designed to American National Standards Institute N101.6-1972, as modified by Regulatory Guide 1.69.

12.1.2.2 Justification for Shield Design - Physical and Mathematical Models

Shield design has been based on normal operations with 1-percent failed fuel or TID 14844 accident releases where applicable. Models for calculating source strengths of critical equipment and systems are discussed in subsection 12.1.3. The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, demineralizers, evaporators, and the containment is a finite cylindrical volume source shield and an infinite shielded cylinder in case of piping.

The mathematical models are based on formulations in "The Engineering Compendium of Radiation Shielding" and the "Reactor Shielding Design Manual." Nuclear data are derived from the "Table of Isotopes," "Reactor Physics Constants," ANL-5800, and XDC-59-8-179. Sources involving different isotopes are divided into different energy bins, corresponding to the gamma energies. The dose contribution from individual sources is calculated based on the above described model. The total dose to the receptor is taken as the sum of doses from each source During shutdown, in cases of corrosion products deposited on surfaces such as a pipe, the latter is treated as a cylindrical surface source.

The radiation shielding in the various plant buildings is described in the following paragraphs.

12.1.2.3 Containment

The containment shield is composed of a reinforced, prestressed, posttensioned, steel line concrete containment that completely surrounds the nuclear steam supply system (NSSS). This shield, together with the primary and secondary shields, reduces the radiation levels for accessibility outside the containment to 0.5 mrem/h. In case of an accident, the shielding will minimize the station doses to less than 5 rem whole body and the offsite doses to less than 0.5 mrem/h.

12.1.2.4 Primary Shield

The primary shield of 6-ft-thick reinforced concrete surrounds the reactor vessel. The cavity between the primary shield and the reactor vessel is air cooled to prevent overheating, dehydration, and degradation of the shielding properties of the concrete. The primary shield, in conjunction with the secondary shield, serves to attenuate the radiation from the reactor vessel and reactor coolant equipment. It permits limited access in the containment during normal power operation and allows limited access to reactor coolant equipment. The primary shield also reduces neutron activation of the components and structures over the life of the plant. Penetrations through the shield walls are described in subsection 12.1.1, item B.

12.1.2.5 Secondary Shield

The secondary shield consists of 2 to 3 1/2 ft of reinforced concrete and surrounds the reactor coolant equipment, steam generators, pressurizer, and associated piping. This shield supplements the primary shield by further attenuation of neutrons escaping the primary shield and permits limited access to the containment during full power operation by attenuating the nitrogen-16 gammas from the primary coolant system. Penetrations through the shield walls are described in subsection 12.1.1, item B.

12.1.2.6 Spent-Fuel Pool Shielding

Shielding is provided for protection during all phases of spent-fuel removal and storage. Operations requiring shielding of personnel are spent-fuel removal from reactor, spent-fuel transfer through refueling canal and transfer tube, spent-fuel storage, and spent-fuel cask loading operations.

Since all spent-fuel removal and transfer operations will be carried out under borated water, minimum water depths above the tops of the fuel assemblies have been established to provide radiation shielding protection. The dose rates at the water surface should normally be less than 2.5 mrem/h. The concrete walls of the fuel transfer canal and spent-fuel pool supplement the water shielding and limit the continuous radiation dose levels in working areas to normally less than 2.5 mrem/h. However, the radiation levels will be closely monitored during removal and transfer operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated dose specified in 10 CFR 20.1201-20.1208.

The refueling water and concrete walls also provide shielding from activated rod cluster control assemblies and reactor internals which will be removed at refueling times. Although dose rates will generally be less than 2.5 mrem/h in working areas, certain manipulation of fuel assemblies, rod cluster control assembly, or reactor internals may produce areas where dose rates exceed 2.5 mrem/h for short periods. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated dose specified in 10 CFR 20.1201 - 20.1208.

All spent-fuel pool penetrations are located higher than the minimum water depth above the fuel assemblies, so that a failure in any penetration will not drain the pool to less than the minimum water level.

12.1.2.7 Control Room

Control room shielding design is based on the requirements set forth in 10 CFR 50, Appendix A, General Design Criterion 19, which requires occupancy of and access to the control room under accident conditions. The dose to personnel will be limited to 5 rem total effective dose equivalent (TEDE) per 10 CFR 50.67 for the duration of the accident. The accident analysis in chapter 15 indicates that this dose to personnel will be less than 5 rem TEDE.

Protection of control room personnel from the fission product release in the containment is provided by the concrete walls between them. Emergency air conditioning and filtration systems are provided for accident conditions and are described in detail in subsection 9.4.1. Figure 12.1-1 contains the control room layout and isometrics of the control room and associated shielding.

12.1.2.8 Auxiliary Building

The auxiliary building shielding includes all concrete walls, covers, and removable blocks that protect personnel working near various system components of the waste processing system, chemical and volume control system, boron thermal regeneration system, and safety injection system. Typical radioactive sources are the waste evaporator, recycle evaporator, demineralizers, filters, waste gas decay tanks, waste holdup tanks, recycle holdup tanks, and the waste drumming area.

Equipment is shielded in compartments, and the shield walls in each compartment are evaluated on the basis of radiation levels within the compartment, the surrounding sources, and access and maintenance requirements.

All radioactive areas are accessible through service corridors that can be entered from the access control station. In the high radiation zones, manually operated valves necessary for system operation and normal maintenance of contaminated equipment have been provided with reach rods penetrating through the shield walls into the corridor or have remote manual operators. Gauges and instrumentation requiring visual checking periodically will be inspected from the corridors or on the local or central control boards.

12.1.2.9 Turbine Building

The turbine building is normally accessible during plant operation and shutdown. In the event of a maximum hypothetical accident, access to the turbine building is controlled for radiation protection. Access is normally controlled for security reasons. There is no direct radiation from the turbine building.

12.1.2.10 General Plant Yard Areas

The radiation field in the plant yard areas frequently occupied by plant personnel is limited to < 0.5 mrem/h. The exception being any radiation controlled area (RCA) being setup or established by Radiation Protection (RP).

12.1.2.11 Inspection of Steam Generator Tubing

Inspection of steam generator tubing when required is performed by use of eddy current techniques. Entrance is required to the primary side of the steam generator to install a probe positioner. This positioner is so designed that installation time is minimized. Depending on the number and location of tubes to be inspected, additional entrance to move the positioner may be required. The eddy current and probe inspection monitoring equipment is located and operated from a position remote from the steam generator.

The exposure to personnel is maintained as low as reasonably achievable by a combination of training, shielding, and location of the control, readout, and probe pushing equipment in a remote location.

12.1.2.12 Corridor Leading to the Personnel Access Hatch to the Containment at the 155-ft Level

The corridor has heating, ventilation, and air conditioning equipment rooms on either side which normally have no sources of radioactivity but might have a local hot spot on a contaminated filter. There are no sources in the containment near the access hatch. The steam generators and the pressurizer have concrete shielding around them. The reactor vessel is well below the 155-ft level. Above the corridor is the roof of the auxiliary building; there are no sources on the roof. Below the corridor are electrical penetration rooms, which have no sources. The corridor is in an area of the auxiliary building where radioactive contamination is possible; consequently, access to the area is controlled and personnel must wear an Optically Stimulated Luminescent dosimeter (OSLD) or other personnel dosimetry devices while in the area.

12.1.2.13 Operating Floor of the Containment During Cold Shutdown Condition

The only time that the reactor would normally be placed in a cold shutdown condition is for refueling. Degasification of the primary coolant system will be required in order to refuel the reactor. Therefore, the noble gases will be purged from the top of the pressurizer. The pressurizer and the steam generators have concrete shields around them on the operating floor. The reactor vessel is well below the operating floor and will be covered by water after the refueling canal is flooded. There may be miscellaneous hot spots around the sides of the steam generators and the pressurizer, but these pieces of equipment are shielded. The reactor head will be the hottest source on the operating floor during refueling; temporary shielding will be installed around it for personnel protection, if crud deposits cause hot spots that would otherwise contribute to excessive doses. There are no sources above the operating floor. Sources below the operating floor are shielded by the thick concrete floor.

12.1.2.14 Old Steam Generator Storage Facility

The six old steam generators removed from Unit 1 and Unit 2 primary containments are stored in the old steam generator storage facility (OSGSF), which is a reinforced concrete building that provides long-term storage of and shielding for the steam generators. Any reactor coolant system (RCS) elbows that may be replaced and concrete wall sections cut from the secondary shield walls during the steam generator replacement are also stored in the OSGSF. As shown on the site plot plan (drawing D-170084), this facility is located south of Unit 1, outside the protected area, but is within the owner controlled area and site boundary.

12.1.3 SOURCE TERMS

The shielding design source terms are based upon the three general plant conditions of normal full-power operation, shutdown, and design basis events.

Subsection 12.1.1 and paragraph 5.2.1.19 provide a complete description of design considerations and procedures used to ensure that field run process piping is designated and routed with appropriate regard for minimizing exposures to plant personnel.

12.1.3.1 Sources for Normal Full-Power Operation

The main sources of activity during normal full-power operation are N-16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products. The activity level of N-16 at various locations in the RCS is shown in figure 12.1-2. The isotopic inventory of fission, corrosion, and activation products in the reactor coolant is given in table 11.1-2. All shielding is based on the maximum case of clad defects in fuel rods producing 1.0 percent of core thermal power. Expected sources would be based on defects in fuel rods producing 0.25 percent of core thermal power as discussed in section 11.2. Each plant system was shielded according to the amount of activity present and adjacent zoning and access criteria. These systems include:

- RCS.
- Chemical and volume control system (CVCS).
- Waste processing system.
- Boron recycle system.
- Spent-fuel pool cooling and purification system.
- Steam generator blowdown processing system.

The N-16 activity of the coolant is the controlling radiation source in the design of the RCS secondary shielding and is plotted in figure 12.1-2 as a function of transport time in a reactor coolant loop.

The radiation sources in the CVCS are given in table 12.1-1.

One of the purposes of the CVCS is to provide continuous purification of the reactor coolant water. The major equipment items include the regenerative and letdown heat exchangers, mixed-bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The boron thermal regeneration (BTR) subsystem contains the three BTR heat exchangers and the BTR demineralizers. The seal water subsystem for the reactor coolant pumps includes the injection and return filters and the seal water heat exchanger.

Table 12.1-1 gives a summation of the activity, by energy groups, of all isotopes listed in table 11.1-2. The delay time from the reactor coolant loop is sufficient for decay of the N-16 isotope.

The radiation sources in the ion exchangers, volume control tank, filters, and heat exchangers of the CVCS are also given in table 12.1-1.

The mixed-bed retains the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation bed can be used intermittently to remove lithium for pH control and to supplement the mixed bed in removing Y, Cs, Mo, and the crud metals.

The BTR beds are used to regulate the boron concentration in the reactor coolant water. They are utilized during load follow operations and in removing boron from the coolant as the nuclear fuel is depleted. These demineralizers also collect radioactive anions, such as iodine, which may have passed through the mixed bed.

The regenerative and excess letdown heat exchangers are located in the containment building. They provide the initial cooling for the reactor coolant letdown and their sources include N-16 activity. The balance of the CVCS heat exchangers is located in the auxiliary building where N-16 activity is not a significant factor.

The letdown heat exchanger provides second-stage cooling for the reactor coolant prior to entering the demineralizers. The activity at this point is identical to the letdown coolant source.

The thermal regeneration heat exchangers include the moderating, chiller, and letdown reheat units. The radiation sources in this equipment are modified to account for activity removed by the demineralizers upstream of the units.

The seal water heat exchanger cools the water from the reactor coolant pump seals. In the source tabulation, credit has been taken for activity removed by the demineralizers and the volume control tank.

The radiation sources in the waste processing system are tabulated in table 12.1-2. The major equipment items in the waste gas portion are the waste gas compressors, hydrogen recombiners, and gas decay tanks. The radiation sources in this equipment are based on cold shutdown procedures during which the radioactive gases are stripped from the RCS. The radiation sources in the waste gas equipment are conservatively assumed to be identical.

The liquid waste processing system is considered as several subsystems, based on its intended use during normal operation. The equipment items normally associated with processing reactor grade water are the waste holdup tank, waste evaporator feed filter, and waste evaporator. The evaporator distillate is directed to the waste condensate tank and may be further processed through the waste evaporator condensate demineralizer and filter, if required. The waste evaporator concentrates are sent to the drumming station or solidification and dewatering building for packaging.

Low activity, nonreactor grade water is directed to the floor drain or laundry and hot shower subsystems. Normally this water is analyzed, then discharged. If activity levels prevent this, the water can be processed by a demineralizer/filter or the waste evaporator. The equipment included in the subsystem is the floor drain tank and filter, laundry and hot shower tank and filter, waste monitor tank demineralizer and filter, and two waste monitor tanks. The floor drain and waste monitor tanks provide surge capacity for the waste holdup tank during periods when abnormal volumes of liquid waste are encountered. Hence, for shielding purposes the radiation sources in these tanks are assumed to be the same, i.e., degassed reactor coolant. Similarly, the sources on the floor drain tank filter are the same (100 R/h contact) as the waste evaporator feed filter since they can operate in similar service. The sources on the waste monitor tank demineralizer and filter are based on circulating reactor coolant through these components.

Radioactive spent resins discharged from the various demineralizers are retained in the spent-resin storage tank. The mixed-bed demineralizer contains the most radioactive resin discharged to the storage tank; these sources determine the tank shielding required. The short-lived activity is allowed to decay (~30 days), and the resin is then directed to the solidification and dewatering facility for packaging. The associated equipment includes the spent-resin storage tank and the resin sluice pump and filter. The resin sluice filter is shielded for radiation levels of 100 R/h contact.

Radiation sources in the various pumps in this system are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

Sources in the laundry, hot shower tank and filter, and waste condensate tank are negligible; these items do not require shielding.

The evaporator concentrates and the spent resin are packaged at the solidification and dewatering facility for shipment to an offsite burial facility. Prior to shipment, the packaged waste is stored as described in section 11.5. The shielding for the drum storage area is designed to accommodate the full storage capacity with each drum reading 1 R/h at 3 ft. Spent resin can be stored in a steel shipping shield, if necessary, to limit radiation levels.

The radiation sources in the boron recycle system are listed in table 12.1-3. The major equipment items included in this system are the recycle holdup tanks and the recycle evaporator with its associated equipment, i.e., feed demineralizers and filter, condensate demineralizer and filter, and concentrates filter. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

The evaporator feed demineralizers are located upstream of the holdup tanks and contain mixed-bed resins which remove nongaseous activity from the reactor coolant directed to the holdup tanks. A dilution factor of 10 across these beds is taken for all particulate activity.

The evaporator condensate demineralizer is charged with anion resin to remove any boron and iodine activity which may be carried over with the evaporator condensate.

The recycle holdup tanks are each equipped with a diaphragm. Gases which flash from the reactor coolant letdown to the holdup tanks are retained under the diaphragm until /500 ft^3 of gas has accumulated; the gases are then removed to the waste gas system. The radiation sources in the holdup tanks are based on 50 percent of the gaseous activity flashing into the vapor phase.

The recycle evaporator feed filter and condensate filter are located downstream of their respective demineralizers and serve to retain particulates and any resin fines which may escape from the demineralizers.

The Liquid Radwaste Processing System (LRWPS) utilizes a mixture of carbon filters, demineralizers, and reverse osmosis to process the incoming waste streams. Solids remaining from the processing are collected in the SHS concentrate tank and are later dried by LRWPS and transferred to drums for shipping offsite. Resins from the LRWPS demineralizers can be sent to the SHS spent resin storage tank. These resins are also dried by LRWPS and transferred to drums. The resins can also be transferred to the Solidification and Dewatering Facility (SDF) for processing. The LRWPS SHS spent resin tank, concentrate tank, and paddle dryer are surrounded by a shield wall. The drums filled from this equipment are transferred to an adjacent drum storage area which also has a shield wall.

The maximum radiation sources on these filters are listed below. The sources for the feed filter correspond to a radiation level of 100 R/h contact. The condensate filter sources result in levels of less than 1 R/h contact. The maximum activity of the liquid concentrates in the recycle evaporator is 40 µCi/g. The resultant radiation sources on the concentrates filter correspond to an exposure rate of approximately 3 R/h.
The radiation sources in the spent-fuel pool cooling system are given in table 12.1-4. The system demineralizer and filter are used to maintain water clarity and remove activity released during refueling operations and the subsequent fuel cooling period. The filter sources correspond to an exposure rate of 100 R/h contact.

The radiation sources for the steam generator blowdown processing system are given in table 12.1-5. The sources are based on removal of all radioactive contaminants in one 75-ft^3 bed assuming 144 gal/day primary to secondary steam generator leakage, 1 percent-fuel clad defects, and a 90-day service lifetime for the bed.

The exposure rate at site boundary per Ci of stored waste (including shipping casks) is dependent not only on the energy of emissions of the radioactive materials stored but also on the amount of shielding used. The radiation exposure at the site boundary from direct radiation from stored radioactive materials is expected to be a small fraction of the natural background radiation.

12.1.3.2 Sources for Shutdown Conditions

In the reactor shutdown condition the only additional sources of radiation requiring shielding is the residual heat removal system.

The maximum specific source strengths in the residual heat removal loops are given in table 12.1-6. The residual heat removal loop is placed in operation approximately 4 h after reactor shutdown and reduces the reactor coolant temperature to approximately 120°F within about 20 h after shutdown. The sources are maximum values with credit taken for 4 h of activity decay and purification.

12.1.3.3 Sources for Design Basis Events

The fission product sources released to the containment building following a core meltdown accident are based on the assumptions stated in TID-14844.[1] These are as follows:

<u>Core Meltdown Accident</u>

NSSS power level (MWt)	2774

Equivalent fraction of core melting	1.0
Fission product fractional releases	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Minimum full-power operating time (days)	650
Cleanup rate following accident	0.0

The assumptions used in the gap release accident are listed below:

<u>Gap Release Accident</u>

NSSS power level (MWt)	2774
Fraction of gap activity released to containment	1.0
Fraction of gap activity absorbed by the sump water	
Noble gases	0.0
All others	1.0
Reactor coolant volume (ft^3)	9,107
Refueling water volume (ft^3)	40,100
Total volume (ft^3)	49,207

The fission product sources released to the containment following an equivalent 100-percent core meltdown (TID-14844 release) are listed in table 12.1-7. These sources are used to calculate the postaccident radiation levels outside the containment building.

The radiation sources circulating in the residual heat removal loop and associated equipment are tabulated in table 12.1-8.

These sources are based on an accident in which the fission products in the gap region between the fuel pellets and cladding are released to the containment. The nongaseous activity is assumed to be transferred to the sump water which flows in the residual heat removal loop.

The design basis for the postaccident recirculation system is that residual heat removal pump compartments have sufficient shielding to permit limited access in the pump compartments following a gap release accident. The integrated exposure does not exceed 3 rem for any 8-h period after the accident.

12.1.4 AREA MONITORING

12.1.4.1 Design Bases

The area radiation monitoring system is provided to supplement the personnel and area radiation monitoring provisions of the plant radiation protection program described in section 12.3. Included in this system are nine permanently located radiation detectors for Unit 1 and ten permanently located radiation detectors for Unit 2, which provide continuous local and remote indication and alarm of direct radiation dose rate levels. The primary objectives of the system are:

A. To immediately alert plant personnel entering or working in normally unlimited occupancy areas of increasing or abnormally high radiation levels, which, if unnoticed, might possibly result in inadvertent overexposures.

B. To inform the control room operators of the occurrence and approximate location of abnormal events resulting in the release of radioactive materials or the degradation of shielding structures.

C. To provide, in the event of many types of hypothetical accidents leading to the contamination of the plant, a means of remotely determining external dose rates in those areas most likely to be contaminated, prior to entry by personnel.

D. To provide a continuous record of external dose rates at selected locations, thereby ensuring detection of transient increases in doses which are attributable to rapid changes in the radioactivity content of equipment and process streams.

E. To provide information on radiological conditions in the containment, in the event of a NUREG 0578 accident (Unit 1) or NUREG 0737 accident (Unit 2).

In addition, an exemption from 10 CFR 70.24, relative to the authorization to possess special nuclear material at Farley Nuclear Plant, has been granted by the Nuclear Regulatory Commission[4] that provides relief from the requirement to install criticality monitors. These monitors are not needed because inadvertent or accidental criticality will be precluded through compliance with the plant Technical Specifications, geometric spacing of fuel assemblies in the new fuel storage area and spent-fuel storage pool, administrative controls imposed on fuel handling procedures, and the use of nuclear instrumentation that monitors the behavior of nuclear fuel in the reactor vessel.

12.1.4.2 System Description

This system consists of multiple channels which monitor radiation levels in various areas of the plant, among which are the following:

Channel	Area Monitoring
R-1 (Unit 1 only)	Control room
R-1B (Unit 2 only)	Technical support center
R-2	Containment
R-3	Radiochemistry laboratory
R-4	Charging pump room
R-5	Spent-fuel building
R-6	Sampling room
R-7	Incore instrumentation area
R-8	Drumming station
R-9 (Unit 2 only)	Sampling panel room

These locations have been chosen as representative of plant locations where significant sources of radioactive material are stored and/or handled or where occupancy is highest.

Detecting medium for the channels is air with a corresponding temperature range of 40°F to 120°F. Each channel consists of a fixed position gross beta gamma Geiger-Mueller tube or ion chamber (R-2 & R-7 only) detector with range 1.0×10^{-4} to 1.0×10^{1} R/h (rad/h for R-2 & R-7). Drawing U-167647 contains a functional block diagram for the above area monitor channels.

The area radiation level is indicated locally at the detector, in the cable spreading room at the signal processing cabinet (R-2 & R-7 only), and at the radiation monitoring system cabinets. Radiation levels are recorded by a data acquisition system computer which can display data, on demand, to the operator. High radiation alarms are displayed at the radiation monitoring system cabinets and annunciated at the detector location and at the control board in the control room. The control board annunciator provides a single window which alarms for all area radiation monitor channels in addition to process radiation monitor channels R-10 through R-13, 2R-14, R-15, R-17 through R-20, 2R-21, 2R-22, R-23, Unit 1 R-29B (channels E & Composite Gas Channel)/R-29C (channels H & J), and Unit 2 R-29B (channels E & Composite Gas Channel)/R-29C (channels H & J). Channels R-24A and B and R-25A and B have individual annunciator windows on the control board. Verification of which area radiation monitor channel has alarmed is done at the radiation monitoring system cabinets in the control room.

To meet the requirements of NUREG 0578, Alabama Power Company (APC) has installed radiation detection systems R-27A and B to meet the requirements for a high-range containment radiation monitor. Each system consists of an ion chamber detector, signal processing cabinet, readout panel, and interconnecting cables. The detectors are located inside containment about 5 ft above the operating deck and approximately 90° apart. These locations ensure that detectors are not protected by massive shielding and they will provide a reasonable assessment of area radiation conditions inside the containment during and following an accident.

A. Each detector is designed to measure gamma radiation.

B. The range of each detector is 1 R/h to 10^7 rad/h for photon radiation.

C. The energy response is ±14 percent to 81 keV to 3 MeV and ±12 percent from 100 keV to 3 MeV.

D. The calibration frequency will be once per refueling cycle as defined by the Technical Specifications. Capability exists for onsite calibration of the radiation detector to 10 rad/h.

12.1.4.3 Design Evaluation

Area monitors are located in areas of the plant which house equipment containing or processing radioactive fluid or where, because of personnel occupancy, it is deemed necessary to monitor continuously. These instruments continually detect and record operating radiation levels. If the radiation level should rise above the setpoint listed for each channel (see table 12.1-9), an alarm is initiated in the control room. Local annunciation is provided at the detector to indicate high radiation levels to personnel in the area. The radiation monitoring system operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning are thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed the limits of 10 CFR 20.1201 - 20.1208.

12.1.5 OPERATING PROCEDURES

12.1.5.1 General

The Radiation Protection manager is responsible for assisting the training manager in developing a radiation protection training program and for developing a radiation surveillance program to ensure that exposures of all personnel are kept within the limits of 10 CFR 20.1201 - 20.1208. See section 12.3 for a description of the Radiation Protection program as it relates to shielding operating procedures.

12.1.5.2 Procedures

The basic principles of time, distance, and shielding will be applied during operation and maintenance to ensure that personnel exposure will be within limits. Specifically, the following procedures and techniques will be employed:

A. During initial startup, neutron and gamma dose rate surveys will be performed to determine the adequacy of shielding.

B. During normal operations, dose rate surveys will be performed periodically throughout the plant and areas will be posted accordingly. This procedure will ensure that data are available for planning operation and maintenance activities.

C. Radiation areas will be conspicuously posted. High radiation areas will be conspicuously posted and barricaded. High radiation areas of 1 R/h measured at 30 cm but less than 500 rads/h measured at 1 m will be provided with locked or continuously guarded doors and the keys maintained under the administrative control of the shift supervisor or shift support supervisor and/or Radiation Protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. For individual high radiation areas with radiation levels, as measured at 30 cm from the radiation source or from any surface that the radiation penetrates, such that a major portion of the body could receive in 1 h a dose greater than 1000 mrem, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device. Entry into high radiation areas is controlled by the Technical Specifications. Very high radiation areas (levels greater than 500 rads/h at 1 m) will be conspicuously posted and access will be controlled in accordance with plant procedures. These control measures comply with the NRC acceptance criteria contained in 10 CFR 20.1601 and 20.1602, respectively.

D. A radiation work permit system will be employed to ensure proper administrative control over work in restricted areas. The permit is designed to ensure that the radiation conditions are known and that appropriate measures are taken to minimize the dose received by personnel.

E. Extension tools will be used when possible or practical to increase the distance from the radiation source to the worker.

F. Equipment will be moved to areas of lower radiation fields for maintenance when possible or practical.

G. Portable shielding in the form of lead bricks, lead sheets, lead shot, and/or high density concrete blocks will be considered for use when the requirements of items E and F are not possible or practical. Steel plates will be used in lieu of lead where high temperature may be a factor. (A shielding evaluation will be conducted prior to installing shielding on safety-related equipment.)

H. A personnel dosimetry program, as described in subsection 12.3.3, will be administered by the Radiation Protection group to ensure compliance with 10 CFR 20.1502.

Each permanent plant employee who is to be a radiation worker will attend radiation worker orientation prior to being allowed unescorted access.

Experience gained during the operation and maintenance of FNP and that of several nuclear plants with whom SNC has contact will be used to provide a basis for further evaluation and development of shielding procedures.

12.1.6 ESTIMATES OF EXPOSURE

12.1.6.1 Exposures in the Controlled area and in the Unrestricted Area

12.1.6.1.1 Normal Plant Operations

The total effective dose equivalent (TEDE) to individuals due to licensed operation will not exceed 100 mrem in a year in the controlled area, as defined in 10 CFR 20.1003.

Doses at the site boundary from radioactive liquid releases are given in subsection 11.2.9 and those due to gaseous effluents are dealt with in subsection 11.3.9.

At and beyond the site boundary, (unrestricted area, as defined in 10 CFR 20.1003), the interpreted man-rem values (for normal operations) as a function of distance are computed from the expected gaseous and liquid releases, and the atmospheric dilution factors are presented in subsection 11.3.8, liquid dilution and reconcentration factors in subsection 11.2.8, and the population density in subsection 2.1.3.

12.1.6.1.2 Operational Occurrences

Maximum radiation exposures resulting from operational occurrences are discussed in detail in chapter 15. Under the most severe conditions, the dose values at the site boundary, the low population zone distance, and at the visitors center will be well below 10 CFR 100 guidelines.

12.1.6.2 Exposures Within Restricted Areas

12.1.6.2.1 Normal Plant Operations

Administrative controls and controlled access will ensure that the plant personnel working in restricted areas, as defined in 10 CFR 20.1003, will not receive doses in excess of those established in 10 CFR 20.1201 - 20.1208.

The dose rates given in subsection 12.1.1 represent the upper limits, and the expected exposures will be significantly lower than these limits.

12.1.6.2.2 Operational Occurrences

Various operational occurrences are discussed in detail in chapter 15. Minor occurrences like spills or leakages will contribute no appreciable increase to normal exposures and will be of only local significance. However, if the radiation levels from an occurrence call for evacuation of the plant, this will be indicated on the radiation monitoring equipment and the emergency evacuation plan will remain in operation. Personnel essential to shut down the plant and maintain it in a safe condition under accident conditions will direct required operator actions from the control room. Maximum dose rates and protection of the control room are discussed in detail in paragraph 12.1.2.7.

12.1.6.2.3 In-Plant Radiation Monitoring

A program exists which will ensure the capability to accurately determine the airborne iodine concentration in certain plant areas where personnel may be present under accident conditions. This program shall include the following:

a. Training of personnel,
b. Procedures for monitoring, and
c. Provisions for maintenance of sampling and analyses of equipment.

12.1.6.3 Comparison with Other Operating Plants

Liquid effluent releases from operating pressurized water reactor plants are indicated in table 11.2-6 for years 1970 and 1977 and actual measured values for radioactive noble gas releases from FNP Unit 1 for years 1977 to 1983 are indicated in table 11.3-6.

The average anticipated releases from this plant during operation are included in tables 11.2-7 and 11.3-9, and doses from these releases are given in sections 11.3 and 11.2.

12.1.6.4 Estimated Annual Exposures

Table 12.1-10 gives typical doses received by personnel in operating plants based on data from 1977 to 1981. Those areas described in table 12.1-10 that have radiation levels greater than 100 mrem/h are classed as Radiation Zone 5 areas, as described in subsection 12.1.1.

The estimated annual exposure from the plant as designed is 350 man-rem per unit. This number is a typical value for relevant operating plants as shown in table 12.1-10 based on data from 1977 to 1981. The value of 350 man-rem per unit can be confirmed by analysis of other relevant plant data, as shown in table 12.1-11.

12.1.7 DESIGN REVIEW OF PLANT SHIELDING FOR POSTACCIDENT OPERATION

This subsection describes the design review of plant shielding of spaces for postaccident operations, as required by NUREG-0737, item II.B.2. Systems required to process primary reactor coolant outside the containment during postaccident conditions were selected for evaluation. Large radiation sources beyond the original plant design bases were postulated to be present in the selected systems. Areas and equipment which are vital for postaccident occupancy or operation were evaluated to determine whether access and performance of required operator activities might be unduly impaired due to the presence of the postulated radiation source in these systems.

12.1.7.1 Selection of Systems for Shielding Review

The criteria applied in selection of plant systems used in the shielding review resulted in several classifications of systems as discussed below.

A. Category A (Recirculation Systems)

   The first group of systems are those required by plant design to mitigate a design basis loss-of-coolant accident (LOCA) and which might contain highly radioactive sources in excess of the current design basis. A first-priority safety concern is to ensure that operation of those systems containing a significant source will not adversely impact operator functions required outside the containment. Therefore, the following systems have been selected to ensure this first-priority safety concern is adequately addressed by the existing plant shielding design:

   1. Those portions of the containment spray system used to recirculate water from the containment sump back into the containment.

   2. Those portions of the residual heat removal system used to recirculate water from the containment sump back into the containment.

   3. Those portions of the high-head safety injection system used to recirculate water from the containment sump via the residual heat removal system back into the containment.

B. Category B (Extensions of Containment Atmosphere)

In addition to systems listed above, there are other systems or portions of systems which would contain radioactivity by virtue of their connection to the containment following an accident. Proper operation of the emergency core cooling systems (ECCS) would prevent extensive core damage and mean that these systems would not be expected to contain the significant radioactive sources required by this special analysis. Nevertheless, such sources have been postulated in the following systems:

1. Those portions of the postaccident containment combustible gas control system external to the containment which would contain the atmosphere from the containment.

2. Those portions of the containment ventilation systems external to the containment up to the second isolation valve which could contain the atmosphere from the containment.

3. Those portions of the sampling system used to obtain a containment atmosphere sample.

C. Category C (Liquid Samples)

Item II.B.3 of NUREG 0737 required that certain postaccident liquid samples be obtained from the RCS for containment systems. Those portions of the sampling system which were identified for use to meet the intent of item II.B.3 were selected for this shielding review.

D. Category D (Letdown)

That portion of the letdown system from the RCS past the letdown heat exchanger up to the inlet valves to the letdown demineralizers has been selected for analysis.

12.1.7.2 <u>Quantification of Potential Radioactive Source Release Fractions</u>

The following release fractions were used as a basis for determining the concentrations for the shielding review:

A. Source A, containment atmosphere – 100% noble gases, 25% halogens.

B. Source B, reactor coolant – 100% noble gases, 50% halogens, 1- percent solids.

C. Source C, containment sump liquid – 50% halogens, 1% solids.

The above release fractions were applied to the total Ci available for the particular chemical species (i.e., noble gas, halogen, or solid) for an equilibrium fission product inventory for a light water reactor core.)

12.1.7.3 <u>Source Term Models</u>

The paragraph above outlines the assumptions used for release fractions for the shielding design review. These release fractions are, however, only the first step in modeling the source terms for the activity concentrations in the systems under review. The important modeling parameters of decay time and dilution volume also affect shielding analysis. The following sections outline the rationale for the selection of values for these key parameters.

12.1.7.3.1 Decay Time

For the first stage of the shielding design review process, no decay time credit was used with the above release. The primary reason for this was to develop a set of accident radiation zone maps normalized to no decay that could be used as a tool by the plant staff along with a set of decay curves to quantitatively assess the plant status quickly following any abnormal occurrence. However, the following decay times were used in assessing anticipated potential personnel radiation exposure due to those operator actions required post-LOCA.

For analyses of personnel exposures in vital areas outside the control room, radioactive decay equivalent to 10 min allowed for operator action was used as the <u>minimum</u> decay time.

A decay time of 24 min, which is consistent with the time for initiation of recirculation in accordance with chapter 6, was allowed for the review of those ECCS systems that are used to recirculate water from the containment sump back into the containment.

12.1.7.3.2 Dilution Volume

The volume used for dilution is important, affecting the calculations of dose rate in a linear fashion. The following dilution volumes were used with the release fractions and decay times listed above to arrive at the final source terms for the shielding reviews:

A. Source A, containment free volume - The volume occupied by the ECCS water was neglected.

B. Source B - RCS volume based on reactor coolant density at the operating temperature and pressure.

C. Source C - The volume of water present at the time of recirculation (RCS + refueling water storage tank + safety injection tanks).

12.1.7.3.3 Sources Used in Piping and Equipment for Each System Under Review

In defining the limits of the connected piping subject to contamination listed below, normally shut valves were assumed to remain shut.

A. Containment spray system - At the initiation of recirculation, source C was used.

B. High-head safety injection system - At the initiation of recirculation, source C was used.

C. Residual heat removal system - Source C was used for sump recirculation mode.

D. Sampling systems - The sources used in the shielding design review for sampling systems were as follows:

   1. Containment air sample - Source A.

   2. Reactor coolant sample - Source B.

E. Letdown system - The liquid source was source B.

12.1.7.4 <u>Shielding Design Review Methodology</u>

12.1.7.4.1 Analytical Shielding Techniques

The previous sections outlined the rationale and assumptions for the selection of the systems that would undergo a shielding design review as well as the formulation of the sources for those systems. The next step in the review process was to use these sources along with standard point kernel shielding analytical techniques to estimate dose rates from those selected systems. For compartments containing the systems under review, estimates were made for a general area dose rate rather than superimposing the maximum dose rate at contact with the surfaces of all individual components of that system in the compartment. For corridors outside compartments, reviews were done to check the dose rate transmitted into the corridor through the walls of adjacent compartments. Checks were also made for any piping or equipment that could directly contribute to corridor dose rate, i.e., piping that may be running directly into the corridor or equipment/piping in a compartment that could shine directly into corridors with no attenuation through compartment walls.

12.1.7.4.2 Accident Radiation Zone Maps

One of the two principal products of this review is the series of accident radiation zone maps. These zone maps represent the correlation of the dose rates as estimated above with the required operator actions and resultant necessary accessibility to vital areas.

The most conservative decay curve (source B) should be used if doubt exists about which source is causing the dose rate. The accident zone maps give total dose rates from all sources and do not distinguish between sources A, B, or C.

These zone maps are shown in drawings D-176075, D-176076, D-176077, D-176078, D-176079 (Unit 1), and drawings D-206075, D-206076, D-206077, D-206078, and D-206079 (Unit 2) for elevations 77 to 83 ft, 100 to 105 ft, 121 to 129 ft, 139 ft, and 155 ft.

The zone boundaries were formulated based on the following rationale:

Zone Designation	Rationale	Zone Dose Rate Limits ($\dot{D}$) (rem/h)
A-I	The first zone is consistent with personnel radiation exposure guidelines for vital areas requiring continuous occupancy.	$0 \leq \dot{D} \leq 0.015$
A-II	The second zone is consistent with the personnel radiation exposure guidelines for vital areas requiring occasional access or for corridors to these areas.	$0.015 \leq \dot{D} \leq 0.100$
A-III	The third zone is consistent with the personnel radiation exposure guidelines for vital areas requiring infrequent access or corridors to these areas.	$0.100 \leq \dot{D} \leq 5.0$

The subsequent zones were selected by grouping them by powers of 10 so that rapid assessment of additional shielding measures could be used via tenth-value layers of common shielding materials.

Zone Designation	Zone Dose Rate Limits ($\dot{D}$)(rem/h)
A-IV	$5 \leq \dot{D} \leq 50$
A-V	$50 \leq \dot{D} \leq 500$
A-VI	$500 \leq \dot{D} \leq 5000$
A-VII	$5000 \leq \dot{D} \leq 50,000$
A-VIII	$\dot{D} \geq 50,000$

These zone designations should not be confused with those used for the normal plant operation zone maps shown in drawings D-176035, D-176036, D-176037, D-176038, D-176039 (Unit 1), and drawings D-206035, D-206036, D-206037, D-206038, D-206039, (Unit 2).

12.1.7.4.3 Decay Curves

Figures 12.1-3 through 12.1-6, corresponding to sources A, B, and C outlined in paragraph 12.1.7.2, are a set of curves developed for the shielding design review to serve as generic tools to estimate transient decay credit. These generic curves were developed rather than developing a parametric set of curves for each source that would account explicitly for the effects of self-attenuation in the source material on actual dose rates. The primary assumption in the application of the curves was that the dose rate from a source was directly proportional to the total gamma ray energy release rate (in MeV/s) from the source in question, i.e., source A, B, or C. Therefore, the decay curves are more properly fission product energy release rate curves. These curves were developed in a similar manner as those from the work of Lurie, et al.[2] for the Sandia Laboratory research directed by Bonzon, et al.[3] All curves have been normalized to the initial energy release rate for the source in question.

12.1.7.5 Postaccident Access and Personnel Exposure

12.1.7.5.1 Access

Those operator actions required post-LOCA were reviewed to ensure that first-priority safety actions can be achieved in the postulated radiation fields. This review ensures that access is available and required operator actions can be achieved.

The following areas in the auxiliary building require postaccident access:

Area	Occupancy Period
Control room, technical support center	24 h/day
Radiation Protection area	24 h/day
Hallway 316	1 h/day
Hallway 409	1 h/day
Hallway 322 (outside sample room)	2 h/day
Cable spreading room	1/2 h
Filter rooms	2 h/day
Switchgear rooms (el 121 ft)	1/2 h
Hot shutdown panel	24 h/day

Component cooling water pump room	1/2 h
Corridor 161	1/2 h
Residual heat removal heat exchanger room	1/2 h
Stairway 1	Transit to elevations at west side of auxiliary building
Stairway 2	Transit to el 77 ft to el 83 ft
Stairway 8	Transit to elevations at north and east sides of auxiliary building

12.1.7.5.2 Personnel Radiation Exposure

The general basis for personnel radiation exposure guidelines was 10 CFR 50, Appendix A, General Design Criterion 19. The following additional radiation guidelines were used to evaluate occupancy and accessibility of plant vital areas. General area dose rates were used rather than maximum surface dose rates. Contributions from all sources were considered.

A. Vital areas requiring continuous occupancy – Vital areas such as control room and the onsite technical support center were verified to ensure the direct dose rate was less than 15 mR/h.

B. Vital areas requiring infrequent access or corridors to these vital areas - For these areas the dose rate was verified to be less than 5 R/h.

   For dose rates greater than 100 mR/h, a man-rem calculation including time and motion analysis was performed to ensure that the integrated exposure for an operator action would not exceed 5 rem as given in General Design Criterion 19.

REFERENCES

1. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962.

2. Lurie, N. A., Houston, D. H., and Naber, J. A., "Definition of Loss-of-Coolant Accident Radiation Source: Summary and Conclusions," SAND 78-0091, May 1978.

3. Bonzon, L. L., Gillen, K. T., and Salazar, E. A., "Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report: October-December 1978," NUREG/CR 0813 or SAND 79-0761, June 1979.

4. Letter from B. L. Siegel (Nuclear Regulatory Commission) to D. N. Morey (Southern Nuclear Operating Company), dated July 31, 1996, regarding exemption from the requirements of 10 CFR 70.24, “Criticality Accident Requirements,” for the Joseph M. Farley Nuclear Plant, Units 1 and 2.

TABLE 12.1-1 (SHEET 1 OF 4)

RADIATION SOURCES - CHEMICAL AND VOLUME CONTROL SYSTEM

Letdown Coolant Sources

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.5×10^5
0.8	2.6×10^5
1.3	1.4×10^5
1.7	1.2×10^5
2.2	1.9×10^5
2.5	1.7×10^5
3.5	1.9×10^4

Mixed-Bed Demineralizer Sources

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.3×10^8
0.8	2.5×10^8
1.3	3.0×10^7
1.7	1.2×10^7
2.2	3.5×10^6
2.5	2.4×10^5
3.5	1.6×10^5

Cation Bed Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	4.4×10^5
0.8	2.2×10^8
1.3	5.9×10^6

TABLE 12.1-1 (SHEET 2 OF 4)

Boron Thermal Regeneration Demineralizers

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	2.3×10^5
0.8	1.4×10^5
1.3	5.2×10^4
1.7	2.3×10^4
2.2	9.1×10^3

Volume Control Tank

Vapor Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	2.6×10^6
0.4	5.7×10^5
0.8	1.8×10^5
1.7	1.4×10^5
2.2	3.3×10^5
2.5	6.8×10^5

Liquid Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	7.3×10^4
0.8	4.3×10^4
1.3	1.4×10^4
1.7	2.6×10^4
2.2	3.9×10^4
2.5	8.7×10^4

TABLE 12.1-1 (SHEET 3 OF 4)

Reactor Coolant Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	5.7×10^7
1.3	1.5×10^7

Seal Water Injection Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	4.8×10^7
1.3	1.2×10^7

Seal Water Return Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.1×10^7
1.3	3.0×10^6

Regenerative Heat Exchanger and Excess Letdown Heat Exchanger

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.5×10^5
0.8	2.6×10^5
1.3	1.4×10^5
1.7	1.2×10^5
2.2	1.9×10^5
2.5	1.7×10^5
3.5	1.9×10^4
6.1	2.2×10^6
7.1	1.8×10^5

TABLE 12.1-1 (SHEET 4 OF 4)

Letdown Heat Exchanger

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.5×10^5
0.8	2.6×10^5
1.3	1.4×10^5
1.7	1.2×10^5
2.2	1.9×10^5
2.5	1.7×10^5
3.5	1.9×10^4

Boron Thermal Regeneration System (Moderating, Chiller, and Letdown Reheat Heat Exchangers)

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	1.1×10^5
0.8	6.7×10^4
1.3	1.4×10^4
1.7	3.9×10^4
2.2	1.2×10^5
2.5	1.6×10^5

Seal Water Heat Exchanger

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	3.2×10^5
0.4	7.3×10^4
0.8	4.3×10^4
1.3	1.4×10^4
1.7	2.6×10^4
2.2	3.9×10^4
2.5	8.7×10^4

TABLE 12.1-2 (SHEET 1 OF 4)

RADIATION SOURCES - WASTE PROCESSING SYSTEM

Waste Evaporator Condensate Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.4	2.1×10^4
0.8	4.0×10^4
1.3	2.6×10^3
1.7	1.5×10^3
2.2	4.5×10^2

Waste Monitor Tank Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.2×10^6
0.8	3.9×10^6
1.3	1.0×10^6
1.7	5.0×10^5
2.2	1.9×10^5

Evaporator Concentrates

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.8	1.2×10^6

TABLE 12.1-2 (SHEET 2 OF 4)

Drumming Station

Spent Resin

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.0×10^8
1.3	1.0×10^7

Evaporator Concentrates

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.8	1.2×10^6

Waste Holdup Tank, Floor Drain Tank, and Waste Monitor Tanks

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.4	3.7×10^4
0.8	2.1×10^5
1.3	1.0×10^5
1.7	4.8×10^4
2.2	1.7×10^4

Spent-Resin Storage Tank

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.9×10^8
0.8	2.5×10^8
1.3	3.0×10^7
1.7	1.2×10^7
2.2	3.5×10^6
2.5	2.4×10^5
3.5	1.6×10^5

TABLE 12.1-2 (SHEET 3 OF 4)

Hydrogen Recombiner, Waste Gas Compressor, and Gas Decay Tanks

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.1	2.0×10^6
0.4	3.5×10^5
0.8	9.4×10^4
1.7	7.5×10^4
2.2	1.4×10^5
2.5	3.3×10^5

Waste Evaporator Feed Filter and Floor Drain Tank Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.8	3.4×10^7
1.3	8.9×10^6

Spent-Resin Sluice Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.8	1.1×10^7
1.3	3.0×10^6

Waste Monitor Tank Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.4	6.7×10^6
0.8	2.1×10^7
1.3	5.7×10^6
1.7	2.8×10^6
2.2	1.1×10^6

TABLE 12.1-2 (SHEET 4 OF 4)

Waste Evaporator Condensate Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	1.2×10^5
0.8	2.2×10^5
1.3	2.5×10^4

Waste Evaporator Vent Condenser Vapor

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	1.1×10^7
0.4	3.6×10^6
0.8	1.5×10^6
1.7	1.5×10^6
2.2	3.8×10^6
2.5	5.3×10^6

TABLE 12.1-3 (SHEET 1 OF 3)

RADIATION SOURCES - BORON RECYCLE SYSTEM

Evaporator Feed Demineralizers

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	4.7×10^6
0.8	3.9×10^7
1.3	3.7×10^6
1.7	1.9×10^6
2.2	6.5×10^5

Recycle Evaporator Condensate Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	4.5×10^4
0.8	2.2×10^4
1.3	5.4×10^3
1.7	2.5×10^3
2.2	9.7×10^2

Recycle Holdup Tanks

Vapor Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	8.8×10^5
0.4	2.0×10^5
0.8	6.7×10^4
1.7	5.2×10^4
2.2	1.1×10^5
2.5	2.5×10^5

TABLE 12.1-3 (SHEET 2 OF 3)

Liquid Phase

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	1.6×10^5
0.4	5.5×10^4
0.8	2.9×10^4
1.3	1.0×10^3
1.7	1.6×10^4
2.2	5.9×10^4
2.5	8.1×10^4

Recycle Evaporator Feed Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.8	1.1×10^7
1.3	3.0×10^6

Recycle Evaporator Condensate Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.4	1.6×10^5
0.8	8.0×10^4
1.3	3.3×10^4

Recycle Evaporator Concentrates Filter

Gamma Energy (MeV/γ)	Specific Source Strength ($MeV/cm^3/s$)
0.8	1.2×10^6

TABLE 12.1-3 (SHEET 3 OF 3)

Recycle Evaporator

Vent Condenser Vapor

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.1	1.1×10^7
0.4	3.6×10^6
0.8	1.5×10^6
1.7	1.0×10^6
2.2	3.8×10^6
2.5	5.3×10^6

Evaporator Concentrates

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.8	1.2×10^6

TABLE 12.1-4

RADIATION SOURCES - SPENT-FUEL POOL COOLING AND PURIFICATION SYSTEM

Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	2.1×10^6
0.8	7.2×10^5
1.3	2.2×10^3
1.7	4.4×10^3

Filter

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.8	1.1×10^7
1.3	3.0×10^6

TABLE 12.1-5

RADIATION SOURCES - STEAM GENERATOR BLOWDOWN PROCESSING SYSTEM

Demineralizer

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/cm^3/s)
0.4	2.2×10^5
0.8	9.2×10^5
1.3	8.7×10^3
1.7	1.0×10^4

TABLE 12.1-6

RADIATION SOURCES - RESIDUAL HEAT REMOVAL SYSTEM

Gamma Energy (MeV/γ)	Specific Source Strength (MeV/g/s)
0.1	1.9×10^5
0.4	5.0×10^4
0.8	8.5×10^4
1.3	3.0×10^4
1.7	1.9×10^4
2.2	1.7×10^4
2.5	2.9×10^4

TABLE 12.1-7

INSTANTANEOUS DIRECT GAMMA SOURCE STRENGTH (MeV/s)

	Time after Release				
Gamma Energy (MeV)	0 hour	1 hour	2 hours	1 day	1 month
Gases					
0.4	1.9×10^{18}	1.8×10^{18}	1.7×10^{18}	5.3×10^{17}	1.0×10^{15}
0.8	3.4×10^{18}	1.8×10^{18}	1.3×10^{18}	9.5×10^{16}	9.0×10^{14}
1.3	1.1×10^{17}	8.8×10^{16}	6.9×10^{16}	3.0×10^{14}	0.0
1.7	1.1×10^{19}	5.3×10^{17}	4.2×10^{17}	1.8×10^{15}	0.0
2.2	6.3×10^{18}	1.9×10^{18}	1.2×10^{18}	4.9×10^{15}	0.0
2.5	4.7×10^{18}	3.2×10^{18}	2.3×10^{18}	6.7×10^{15}	0.0
3.5	4.7×10^{18}	0.0	0.0	0.0	0.0
Particulates					
0.4	5.0×10^{17}	4.4×10^{17}	4.3×10^{17}	4.1×10^{17}	2.2×10^{17}
0.8	8.2×10^{18}	6.4×10^{18}	4.8×10^{18}	9.5×10^{17}	1.1×10^{17}
1.3	2.6×10^{18}	1.7×10^{18}	1.3×10^{18}	8.2×10^{16}	1.8×10^{15}
1.7	1.1×10^{18}	5.5×10^{17}	4.1×10^{17}	1.0×10^{17}	2.2×10^{16}
2.2	3.9×10^{18}	3.4×10^{18}	3.0×10^{18}	2.9×10^{17}	1.3×10^{15}
2.5	4.3×10^{17}	2.3×10^{17}	1.7×10^{17}	1.8×10^{16}	2.0×10^{15}
3.5	4.6×10^{17}	9.9×10^{16}	2.9×10^{16}	3.5×10^{14}	9.2×10^{13}

TABLE 12.1-8

ACCIDENT SOURCE STRENGTH IN RESIDUAL HEAT REMOVAL LOOP (MeV/cm^3/s)

Gap Release Accident

Gamma Energy (MeV/ϒ	Time After Release 0 hour	1 hour	2 hours	8 hours	1 day	1 week
0.4	1.7×10^7	1.5×10^7	1.4×10^7	1.2×10^7	1.1×10^7	1.0×10^7
0.8	1.4×10^8	1.2×10^8	1.1×10^8	8.3×10^7	7.4×10^7	5.2×10^7
1.3	9.1×10^6	6.6×10^6	4.9×10^6	7.7×10^5	5.2×10^4	1.2×10^2
1.7	5.2×10^6	3.7×10^6	2.7×10^6	4.4×10^5	1.9×10^4	3.1×10^2
2.2	4.9×10^6	3.9×10^6	3.2×10^6	6.0×10^5	1.9×10^5	----
2.5	1.8×10^6	1.3×10^6	8.8×10^5	1.4×10^5	7.6×10^3	----
3.5	4.8×10^5	3.0×10^5	2.1×10^5	2.7×10^4	1.1×10^2	----

TABLE 12.1-9

AREA MONITOR ALARM SETPOINTS

Channel	Area Monitor	Alarm Level (R/h)[a]
R-1A (Unit 1 only)	Control room	0.75×10^{-3}
R-1B (Unit 2 only)	Technical support center	0.75×10^{-3}
R-2	Containment	At power 90×10^{-3}; after shutdown 20×10^{-3}
R-3	Radiochemistry laboratory	2.0×10^{-3}
R-4	Charging pump room	Inside room 50×10^{-3}
R-5	Spent-fuel building	2.0×10^{-3}
R-6	Sampling room	15.0×10^{-3}
R-7	Incore instrumentation area	50.0×10^{-3}
R-8	Drumming station	15.0×10^{-3}
R-9 (Unit 2 only)	Sample panel room	15.0×10^{-3}
R-27A and B	Containment high radiation monitor	50.0

a. These setpoints are typical of those anticipated during initial plant operation and are subject to change during the life of the plant. Actual setpoints are incorporated in plant procedures.

TABLE 12.1-10 (SHEET 1 OF 3)

TYPICAL DATA FOR OPERATING PLANTS

	MWe	man-rem (1979)[(a)]
Reactor 1	800	132
Reactors 2 and 3	810@	805
Reactors 4 and 5	1044/1100	718
Reactor 6	197	495
Reactor 7	906	30
Reactors 8 and 9	0/859	1279
Reactor 10	911	636
Reactor 11	772	154
Reactor 12	802	472
Reactor 13	898	449
Reactors 14, 15, and 16	860@	1001
Reactor 17	873	126
Reactors 18 and 19	775@	3584
Reactor 20	788	1170
Average		553

Component

Inservice Inspection Doses

	man-rem
Reactor head	1.1
Reactor vessel	7.1
Pressurizer	1.66
Steam generator	2.275
Reactor coolant piping	3.235
Total	15.37

TABLE 12.1-10 (SHEET 2 OF 3)

Component Radiation Dose Rates (mrem/h)

	Reactor A	Reactor B	FNP[b]
Primary loop piping (outside)	200-350	50	200-300
Primary loop piping (inside)	1000-13,000		800-10,000
Steam generator plenum	15,000 (inside shutdown)	10,000 (outside, operating)	12,000-15,000 (inside, shutdown)
Reactor vessel head (outside)	400-500		250-300
Reactor vessel head (inside)	15,000		25,000
Reactor vessel nozzles (outside)	15-20		50-150
Reactor vessel nozzles (inside)	5000		-

Component Radiation Dose Rates (mrem/h)

	Reactor J	Reactor K	Reactor L	FNP[b]
Seal water filter	165,000	140	950	1000-2000
Letdown ion exchanger	>1,000,000	800	100,000	1,000,000 (before depletion)
Letdown filter	600,000	1200	50,000	25,000-50,000
Primary drain tank	100-500	35		
Liquid waste monitor tank		2	20	10
Compactor		3		3-5
Waste gas flash tank		19		20-50
Radwaste process filter		85		60-500
Liquid waste holdup tank		120		100-300
Liquid waste holdup pump (area)		8		10

TABLE 12.1-10 (SHEET 3 OF 3)

	Reactor J	Reactor K	Reactor L	FNP[b]
Charging pump (area)		5		5-10
Seal water heat exchanger		50		50-100
Radiochemistry lab drain tank (area)		140		5
Top of steam generator		1		1
Side of steam generator		5		20
Side of pressurizer		90		30
Top of pressurizer		30		100
Reactor coolant pump		33		30-50
Residual heat removal heat exchanger	500	120	30-50	20-250
Residual heat removal pump (area)		40		50-100
Containment (outside secondary shield)			50-200	10-200 (shutdown)
Boric acid drums	200	200		
Shutdown cooling (area)	100			50-100
Spent fuel pool heat exchanger (area)	30			10
Waste gas storage tank			100	10-50

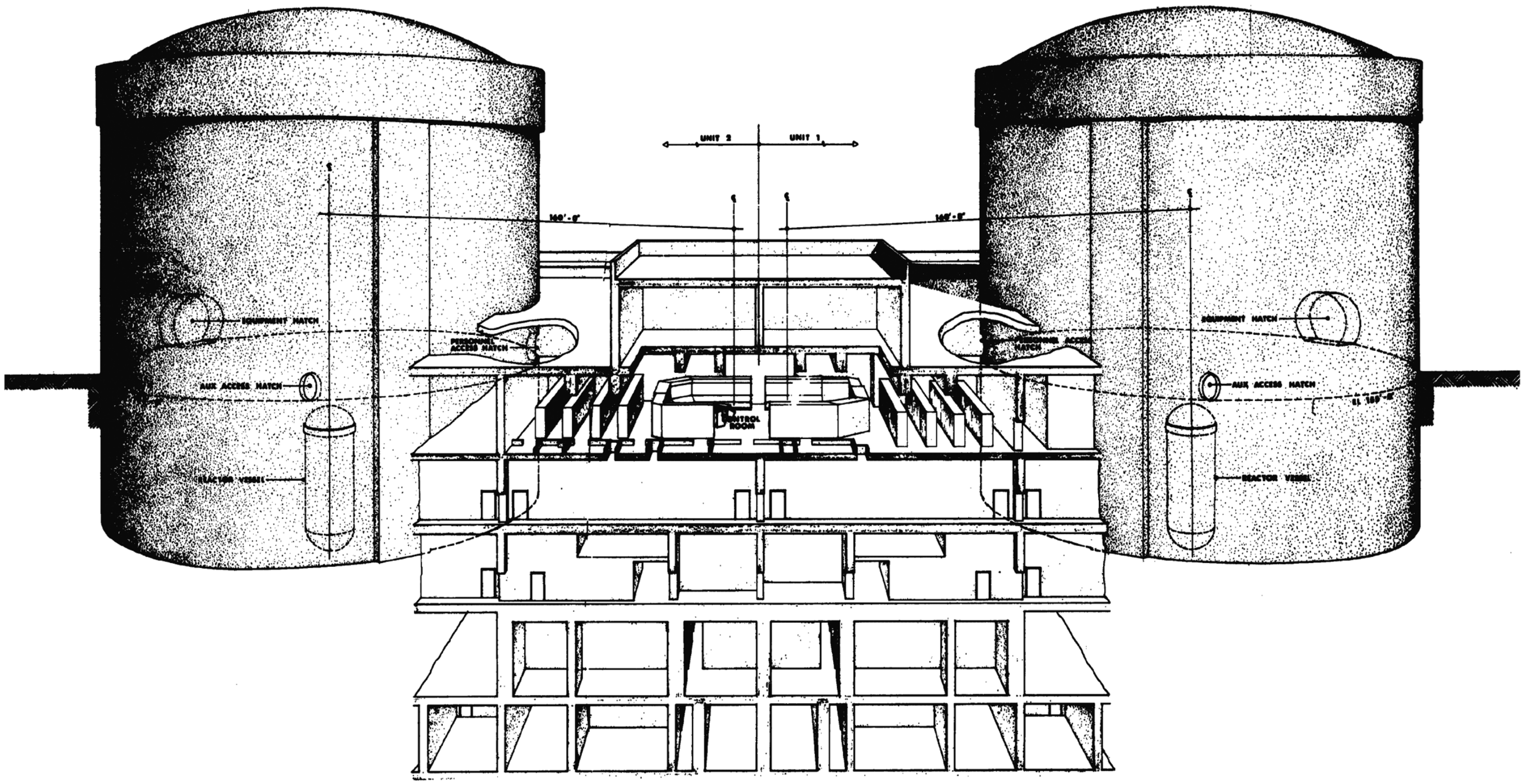
a. Based on NUREG 0713, Volume 1, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1979," Appendix A.

b. FNP Unit 1 (August 1977 through October 1981).

TABLE 12.1-11

EXPECTED ANNUAL MAN-REM PER UNIT BASED ON UNIT 1 OPERATING PLANT DATA

Group	rem/month	rem/year
Operators	5.09	61.1
Maintenance	8.23	98.7
Radiation Protection	4.72	56.6
Engineers	1.0	12.0
Administrators	0.6	7.2
Security	0.1	1.2
Contractors	9.0	108.0
Visitors	Negligible	Negligible
Total	28.74	344.88



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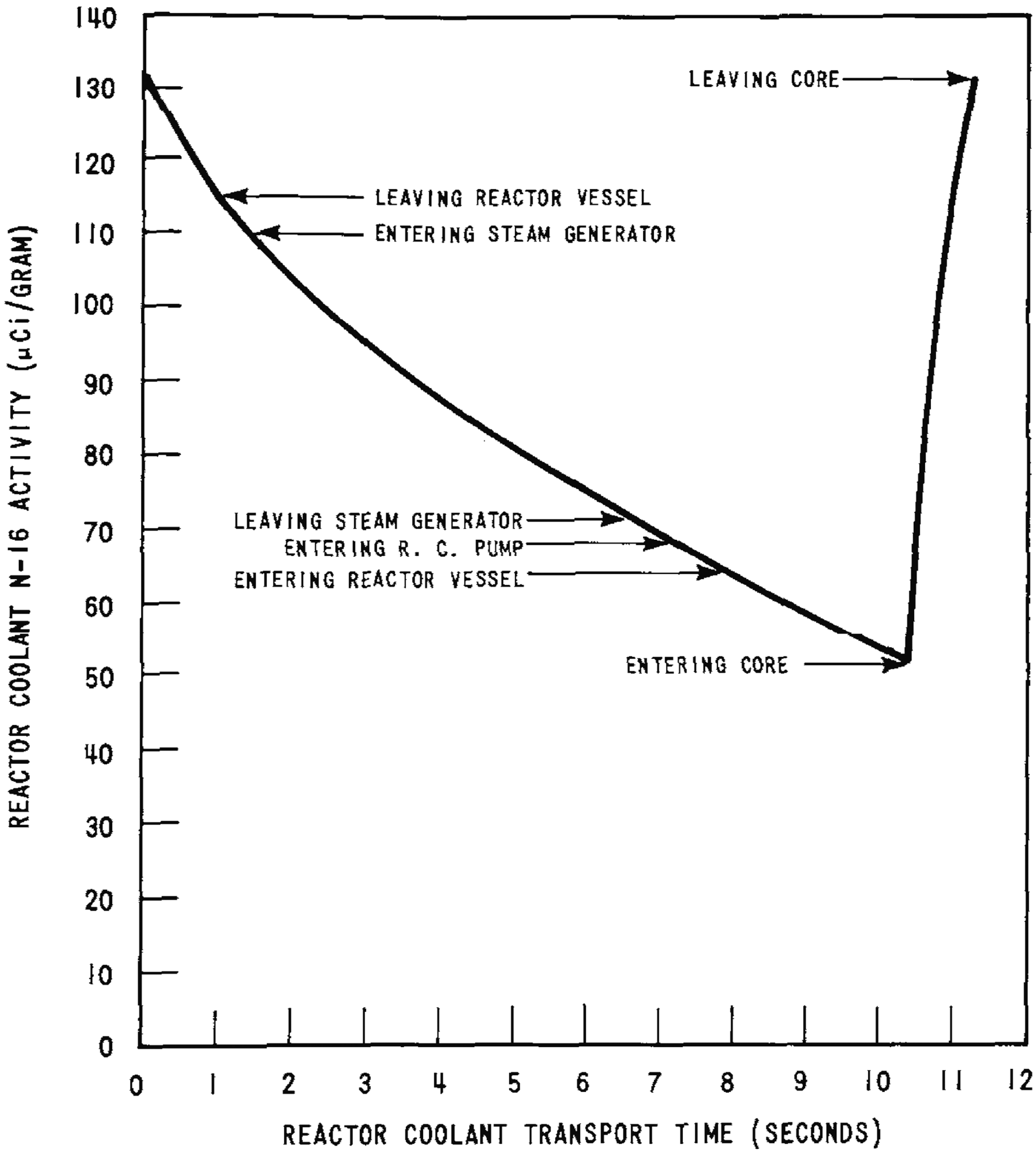
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PERSPECTIVE OF CONTROL ROOM

FIGURE 12.1-1



13

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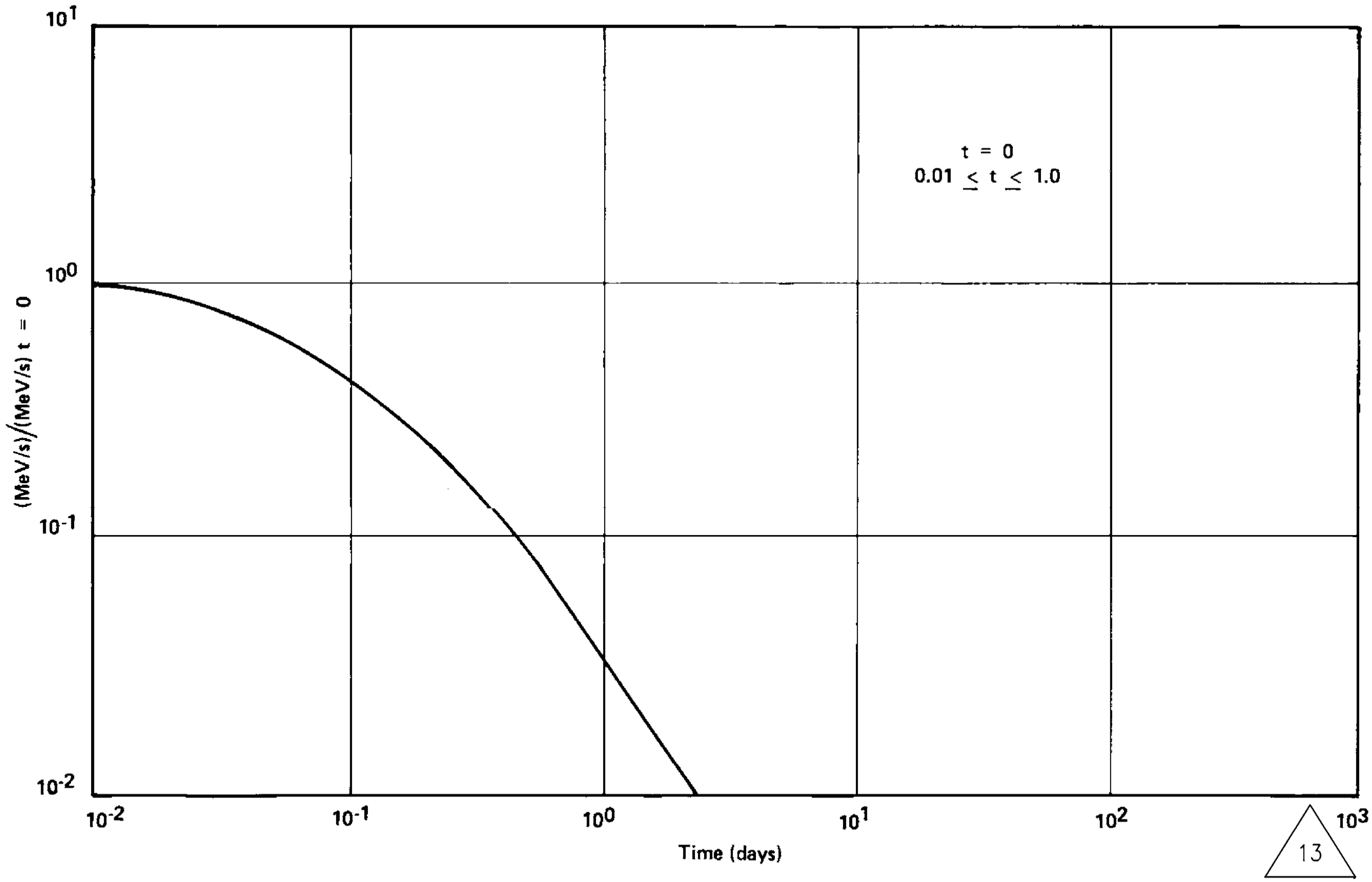
REACTOR COOLANT N-16 ACTIVITY

FIGURE 12.1-2



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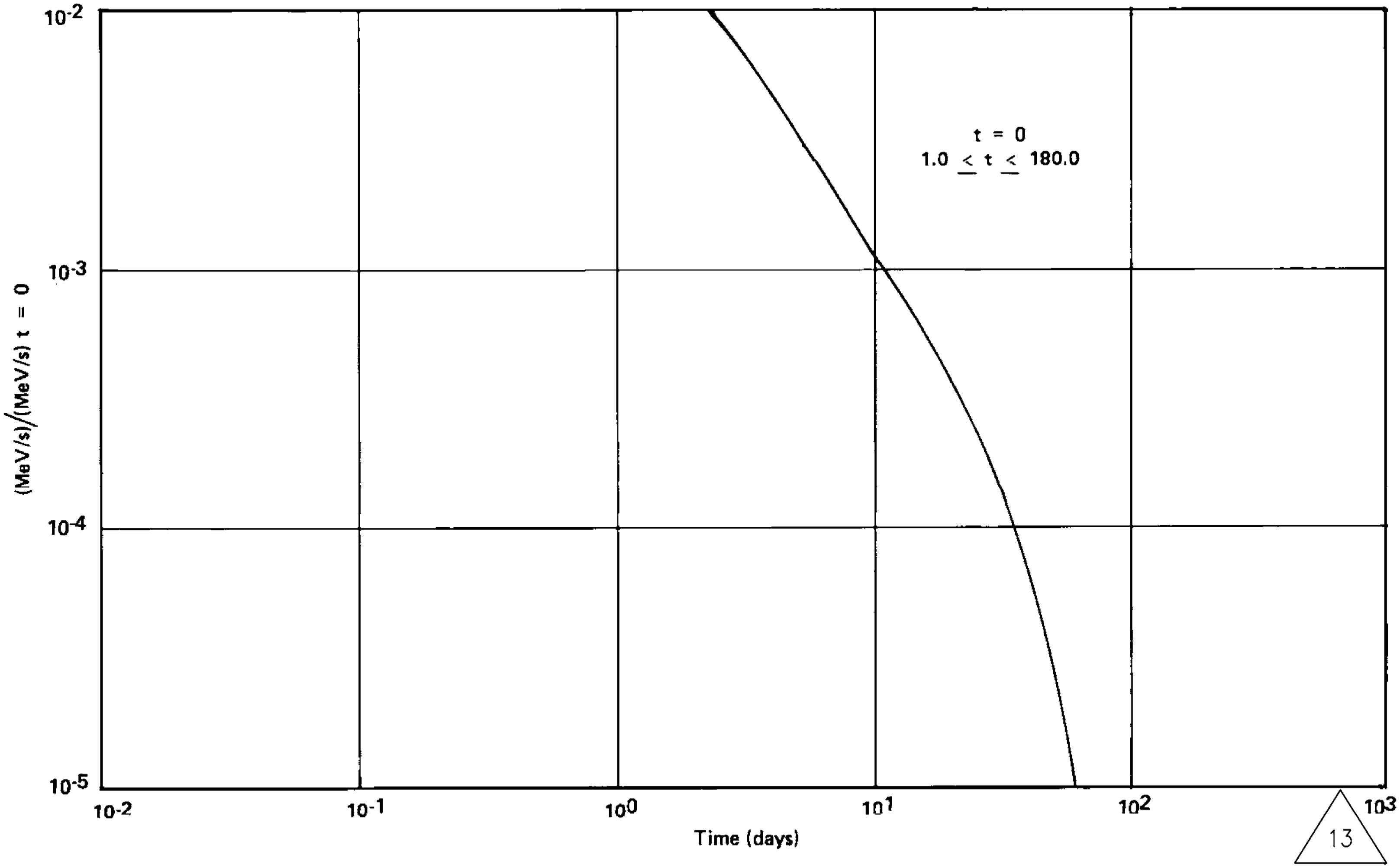
DECAY CURVE FOR SOURCE A, PART 1

FIGURE 12.1-3



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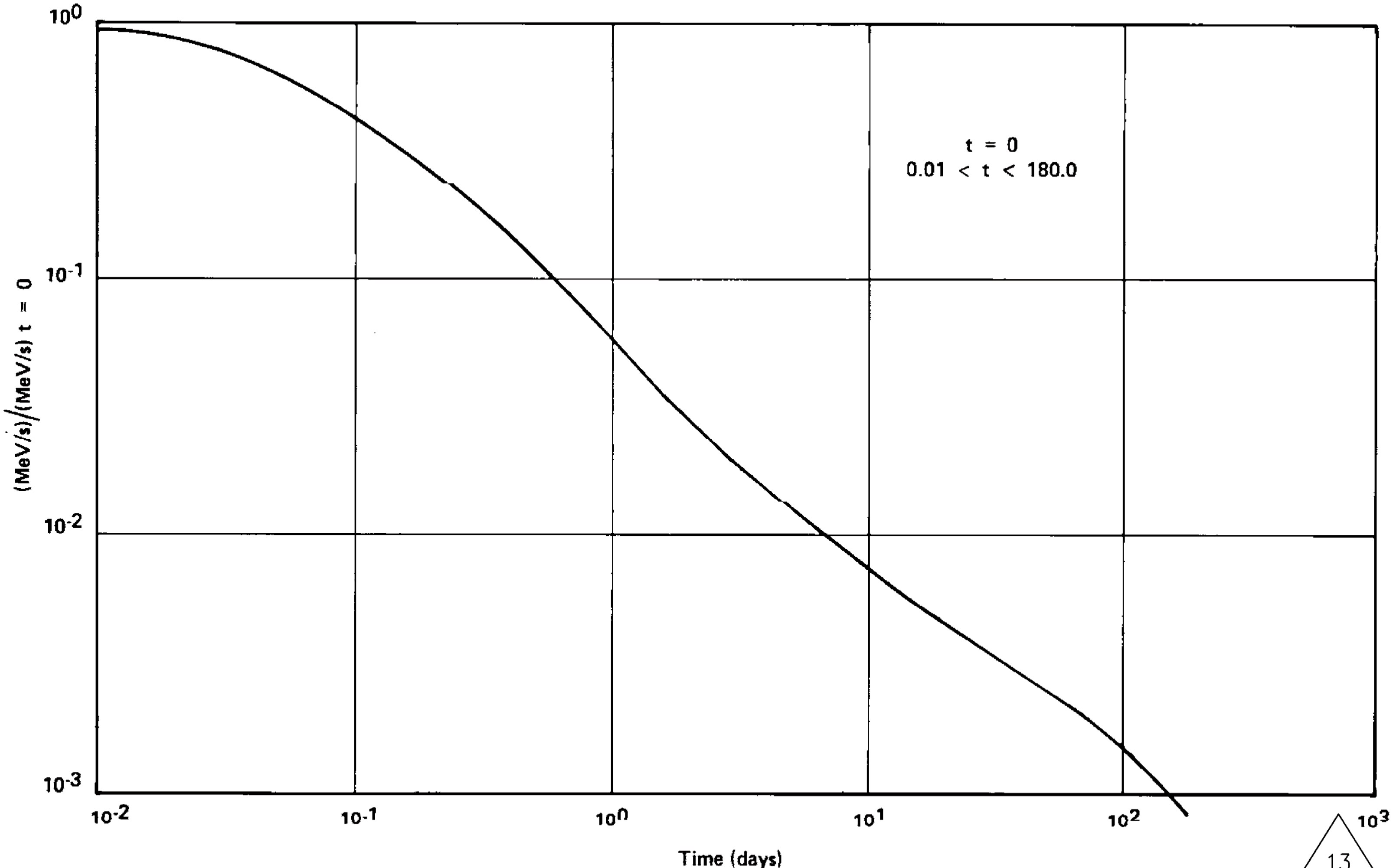
DECAY CURVE FOR SOURCE A, PART 2

FIGURE 12.1-4



13

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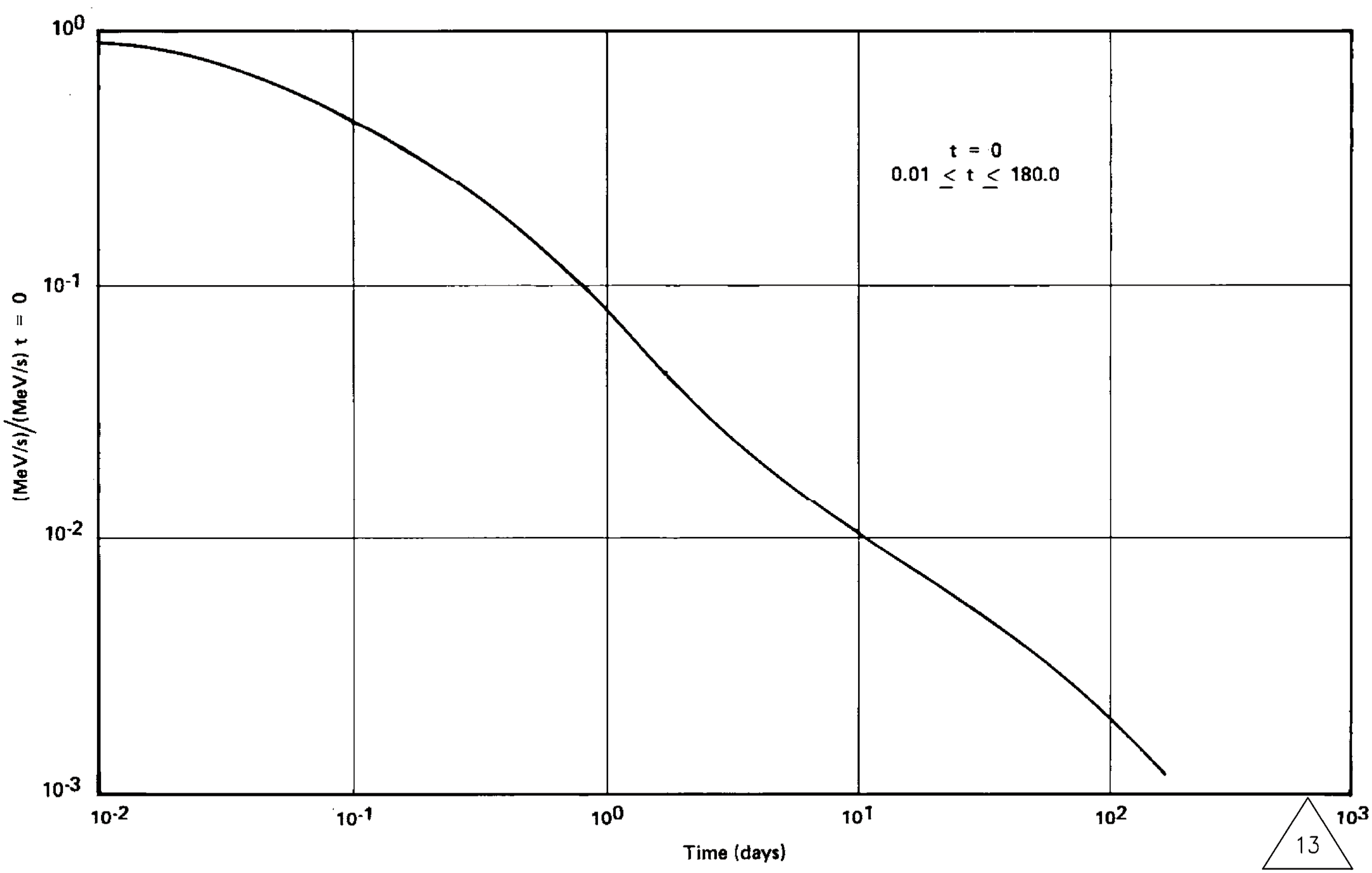
DECAY CURVE FOR SOURCE B

FIGURE 12.1-5



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NUCLEAR PLANT
UNIT 1 AND UNIT 2

DECAY CURVE FOR SOURCE C

FIGURE 12.1-6

12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The plant ventilation systems, in addition to their primary function of preventing extreme thermal environmental conditions for operating personnel and equipment, will provide effective protection for operating personnel against possible airborne radioactive contamination in areas where this may occur.

The systems will operate to ensure that the maximum airborne radioactivity levels for normal operation, including anticipated operational occurrences, are within the limits of column 3, Table 1, Appendix B to 10 CFR 20 for areas within plant structures and on the plant site where construction workers and visitors are permitted. The average airborne radioactivity levels meet the requirements of column 3, Table 1, Appendix B to 10 CFR 20 and 10 CFR 50 and, in fact, will be considerably smaller since average coolant inventories and actual equipment leakages will be small.

The systems will operate to ensure compliance with normal operation offsite release limits as discussed in Section 11.3.

The control room ventilation system will also operate to provide a suitable environment for equipment and continuous personnel occupancy in the control room under postaccident conditions in accordance with 10 CFR 50, Appendix A, General Design Criterion 19 and 10 CFR 50.67.

The expected airborne radioactivity levels for normal operations and anticipated operational occurrences, in the containment and auxiliary buildings are presented in Table 12.2-1. The methods used are discussed in subsection 12.2.6. Assumptions used to calculate these airborne radioactivity levels are presented in Table 12.2-2. A discussion of the estimated doses to personnel at the site is also presented in subsection 12.2.6.

12.2.2 DESIGN DESCRIPTION

In order to accomplish the design objectives, certain general design guidelines are followed when possible and applicable:

A. Air movement patterns are provided from areas of lesser contamination potential to areas of progressively greater contamination potential prior to final exhaust.

B. Slightly negative pressures are maintained, where applicable, to prevent uncontrolled exfiltration of contamination. Slightly positive pressure is maintained in the control room to prevent infiltration of potential contaminants.

C. Valves and equipment are maintained as leaktight as possible in order to prevent leakage of radioactive water and subsequent airborne contamination.

D. Individual air supplies are provided for each building in order to keep potentially contaminated airflows separate from noncontaminated air.

E. High efficiency particulate air (HEPA) and charcoal filters are provided on the exhaust side of the radioactive and fuel handling ventilation systems to remove airborne activity and to reduce onsite and offsite radiation levels.

F. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered to prevent contamination of the control room.

These guides are incorporated in the heating and ventilation design described in Section 9.4. The following is a brief summary of those systems.

12.2.2.1 <u>Control Room Ventilation</u>

During normal plant operation, control room air is recirculated through air conditioning units to maintain control room design conditions of temperature and relative humidity. Fresh air makeup is provided by a supply duct from the computer room air conditioning unit. Redundant radiation monitors are provided on the makeup air supply duct. When a high radiation level is sensed by the monitors, a high radiation alarm is actuated in the control room, the air path is isolated to prevent entry of radioactive contaminants, and the control room ventilation system is aligned to the recirculation mode. After isolation of the control room and when conditions permit, fresh air can be brought in manually through redundant control room pressurization charcoal filter systems.

In the event of a loss-of-coolant accident, the control room is automatically sealed, the ventilation system automatically shifts to recirculation, and the pressurization systems automatically actuate to build up a positive internal pressure.

There are two 100-percent capacity control room filtration systems which are designed to recirculate air through charcoal filters following an accident. A complete description of control room ventilation is found in subsection 9.4.1. The control room area volume is 114,000 ft^3 which includes the space above the suspended ceiling. Control room ventilation system components are described in Table 9.4-1.

12.2.2.2 <u>Containment</u>

The containment cooling system consists of recirculating air cooling units to maintain the design containment temperature and relative humidity. A complete description of the containment ventilation system is found in subsection 6.2.3, and containment cooling system components are described in Table 6.2-28. Containment area volume is listed in Table 6.2-1.

12.2.2.3 Auxiliary Building

The auxiliary building for each unit is served by separate ventilation systems for the fuel handling area, the radioactive waste area, and the nonradioactive area. The shared control room is served by two separate and redundant air conditioning systems. A complete description of the auxiliary building ventilation is found in subsection 9.4.2. The area volume of the auxiliary building is 1.02 x 10^6 ft^3. Components in the system are discussed in detail in subsection 9.4.2.

12.2.2.4 Radwaste Area

Outside air will be filtered, tempered, and delivered to the clean areas such as the lower level corridors. A pressure gradient will be maintained to create airflow from the corridors into the equipment cells, where it will be exhausted after removing airborne contaminants. The exhaust air functions to maintain the area under a negative pressure with respect to the outside.

The volume of the radwaste area is 666,400 ft^3. A complete description of the radwaste area ventilation system is found in subsection 9.4.3, and principal components are described in Tables 9.4-8 and 9.4-9.

12.2.2.5 Turbine Building

The turbine building is provided a recirculating ventilation system which conditions the air for maximum safety and convenience for operating personnel. Passive smoke/heat vents in the turbine building roof allow smoke and heat to exit the turbine building.

The area volume of the turbine building is 4.25 x 10^6 ft^3. A complete description of the turbine building ventilation system is found in subsection 9.4.4. Heating, cooling, and filtration system component design parameters in the turbine building ventilation system are listed in Table 9.4-11.

12.2.2.6 Maintenance Considerations

12.2.2.6.1 Filter Housings

All nonsafety-related charcoal absorbers are of a vertical, fixed-bed design. Contaminated charcoal is conveyed pneumatically from the filter unit to disposal drums in a closed pipeline, with no necessity for personnel to enter the filter housing. Following removal of charcoal, personnel enter the filter housing and place each HEPA filter and prefilter in individual plastic bags, which are sealed before the filters are removed from the filter housing.

12.2.2.6.2 Temporary Ducting

Temporary ducting is used for pneumatic removal of contaminated charcoal from all filter units having fixed charcoal filters. The ducting is used in a closed and leaktight system, and no airborne radioactivity will be released.

12.2.3 SOURCE TERMS

During reactor operation, airborne activities can originate due to system leakage from the following sources:

A. Reactor coolant leakage to the containment building.

B. Reactor coolant leakage to the auxiliary building.

C. Secondary side system leakage.

D. Waste gas processing system leakage.

A complete identification of all radioactive sources and an estimate of resulting radioactive effluents are further described in Section 11.1.

12.2.3.1 Reactor Coolant Leakage to the Containment Building

Leakage into the containment atmosphere is based on leakages from equipment such as pumps and valves. The leakage is estimated to be 40 lb/day.

12.2.3.2 Reactor Coolant Leakage to the Auxiliary Building

This effluent represents nonrecyclable reactor coolant from system leaks in the auxiliary building. It is assumed that the total amount of leakage is 20 gal/day.

12.2.3.3 Secondary Side Leakage to the Turbine Building

The rate of steam leakage from the secondary system is estimated to be 5 gal/min when condensed.

In addition, liquid leakage from systems operating below 212°F is estimated to be 12.5 gal/min.

12.2.3.4 Waste Gas Processing System Leakage

The gaseous waste processing system is designed to contain the gaseous waste for the lifetime of the plant. However, although all precautions are taken to avoid any leakage from the system, an estimated leakage of 100 sf^3/year is assumed.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

An analysis of the auxiliary building was conducted in order to identify the potential points of releases of airborne radioactive material in the form of contaminated steam or liquid discharges from valves, pumps, tanks, sumps, and other release mechanisms. For plant design, an NRC acceptance criterion, discussed in subsection 12.1.2 of the FNP FSAR Safety Evaluation Report, required concentrations of airborne radioactive material to be controlled such that limits stated in 10 CFR 20 would not be exceeded. In-plant airborne radioactive materials concentration limits that were in effect at the time of plant design are specifically stated in 10 CFR 20.103, which references column 1, Table I of Appendix B to 10 CFR 20.1 - 20.601.

During plant operations, access to the rooms, enclosures, or operating areas containing release points, and having the potential of causing operating personnel to be exposed to airborne radioactive material to an average concentration in excess of the limits specified in Appendix B, Table 1, of 10 CFR 20.1001 - 20.2401, will be controlled by a program of:

A. Surveys or continuous online type of sampling.

B. Clear identification of spaces with appropriate caution signs.

C. Locked doors as appropriate.

D. Administrative controls through the use of radiation work permits and procedures.

Other areas of potential airborne contamination, such as the containment, penetration room, and spent fuel area, are monitored by the fixed airborne radiation monitoring instruments described in Section 11.4. The continuous radiation monitors in these areas will be augmented by the use of periodic portable air activity samplers.

The samplers will be used as a check on the fixed monitoring system during normal and maintenance operations and to determine airborne activity levels should an accident occur or after receipt of an alarm from the fixed monitoring system. Systems such as the plant vent air particulate monitor system and the plant vent gas monitor described in subsection 11.4.2 will be checked using grab samples.

The results of these checks will be logged and filed as part of the plant records.

12.2.4.1 Containment Airborne Radioactivity Monitoring

Two Kr-85 radiation monitors are provided in the containment purge exhaust ductwork. The monitors are capable of measuring and alarming 1 MPC-h of Kr-85 when operating in a background of 2 mR/h of 1 MeV gamma rays.

Radiation measurements from these monitors are included in the administrative controls utilized to evaluate the containment radioactivity levels prior to permitting personnel to enter the containment.

The location of the radiation monitors are shown in drawings D-175010, sheet 2, and D-205010, sheet 2.

12.2.4.2 Spent-Fuel Area Airborne Radioactivity Monitoring

The spent-fuel area is continuously exhausted to the plant vent during plant operation. The exhaust flow is continuously monitored for high radiation by two Kr-85 gas monitors capable of alarming when a level of 1 MPC-h is reached.

Dilution factors were not considered in the analysis since representative gaseous samples will exist in the exhaust ductwork over the period of 1 h, due to dispersion within the spent-fuel area.

The radiation monitors have a sensitivity of 5×10^{-7} $\mu Ci/cm^3$ in a background of 2 mR/h of 1 MeV gamma rays.

The location of the radiation monitors is shown in drawings D-175045 and D-205045.

12.2.4.3 Penetration Room Airborne Radioactivity Monitoring

Access to the penetration room will be on an as-required basis for maintenance or repair of equipment. Manual samples will be taken routinely with portable sampling equipment to permit personnel to enter the penetration room compartments as required.

12.2.5 OPERATING PROCEDURES

12.2.5.1 General

The radiation protection group is responsible for developing a radiation protection program which will ensure that inhalation exposure is kept as low as is reasonably achievable, consistent with 10 CFR 20.1101 and 20.1701 - 20.1704.

12.2.5.2 <u>Procedures</u>

Inhalation exposure will be minimized during operations and maintenance by using the following procedures and techniques to determine and cope with the hazards present:

A. Monitoring

Air samplers of various flowrates will be used to collect particulates on high efficiency filter media for subsequent counting. For tritium analysis, freeze-out methods may be used to obtain samples for counting.

Routine smear surveys will be performed to establish the levels of removable contamination throughout the plant so that personnel protection measures or decontamination may be effected.

Assay of noble gases will be performed by drawing an air sample into a sample container and analyzing it on a multichannel analyzer system.

B. Respiratory Protection

In areas where airborne radioactivity can cause exposures in excess of that allowed by 10 CFR 20.1201 - 20.1207 respiratory devices and/or portable HEPA filtration systems may be required. It is the responsibility of the radiation protection group to monitor such areas, to establish the requirement for respiratory equipment, and to control access to such areas through the radiation work permit program.

Each individual who enters a radiation controlled area will be trained or briefed in accordance with 10 CFR 19.12. A notice describing where radiation control procedures may be examined is posted in accordance with 10 CFR 19.11.

12.2.6 ESTIMATES OF INHALATION DOSES

Peak airborne radioisotopic concentrations in the different buildings of operating pressurized water reactor (PWR) plants have shown that these concentrations are insignificant for PWR plants. The inhalation doses to plant personnel at these plants have been found to be negligible.

The doses to plant personnel and construction workers from airborne radioactivity will depend upon the extent of their occupancy and the time when this occupancy occurs. These doses will be controlled by limiting personnel occupancy in the contaminated areas and by provision of respiratory protection equipment if required. The highest dose to plant personnel will therefore be limited to the maximum permissible dose for occupationally exposed individuals, as specified by 10 CFR 20.1201 - 20.1207.

The assumptions used to estimate concentrations and inhalation doses in the containment, turbine building, and certain regions within the auxiliary building are listed in Table 12.2-2. The airborne peak concentrations in each of the regions mentioned above are given in Table 12.2-1 In addition, the table gives the Derived Air Concentrations for airborne activity in these areas as defined in column 3, Table 1, of Appendix B to 10 CFR 20.

The annual inhalation doses to plant personnel due to the airborne radioisotopes in each of the above mentioned regions are presented in Table 12.2-3.

TABLE 12.2-1

PEAK AIRBORNE RADIOISOTOPIC CONCENTRATIONS IN THE DIFFERENT REGIONS OF THE PLANT

	Turbine Building		Containment @ Power		Containment @ Refueling		Waste Gas Processing Area		Waste Monitor Tank Rooms		Radwaste Area	
Isotope	Concentration (µCi/cc)	DAC for 40 hr/wk (µCi/cc)[1]	Concentration (µCi/cc)	DAC for 4 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 40 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 2 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 2 hr/wk (µCi/cc)	Concentration (µCi/cc)	DAC for 40 hr/wk (µCi/cc)
Kr-83m	8.03E-10	1.00E-02	0	1.00E-01	0	1.00E-02	0	2.00E-01	0	2.00E-01	0	1.00E-02
Kr-85	5.35E-10	1.00E-04	2.49E-08	1.00E-03	2.56E-09	1.00E-04	4.08E-06	2.00E-03	2.68E-07	2.00E-03	1.32E-09	1.00E-04
Kr-85m	6.85E-09	2.00E-05	1.20E-07	2.00E-04	5.01E-12	2.00E-05	6.24E-08	4.00E-04	3.86E-06	4.00E-04	1.89E-08	2.00E-05
Kr-87	1.77E-09	5.00E-06	2.73E-08	5.00E-05	3.79E-16	5.00E-06	5.76E-09	1.00E-04	2.47E-06	1.00E-04	1.22E-08	5.00E-06
Kr-88	1.01E-08	2.00E-06	1.48E-07	2.00E-05	8.95E-13	2.00E-06	6.24E-08	4.00E-05	7.29E-06	4.00E-05	3.60E-08	2.00E-06
Xe-131m	1.34E-09	4.00E-04	3.62E-08	4.00E-03	0	4.00E-04	0	8.00E-03	0	8.00E-03	0	4.00E-04
Xe-133	3.86E-07	1.00E-04	1.32E-05	1.00E-03	1.21E-07	1.00E-04	1.82E-05	2.00E-03	1.54E-04	2.00E-03	7.54E-07	1.00E-04
Xe-133m	8.43E-09	1.00E-04	2.25E-07	1.00E-03	8.50E-10	1.00E-04	2.11E-07	2.00E-03	2.90E-06	2.00E-03	1.44E-08	1.00E-04
Xe-135	2.49E-08	1.00E-05	4.97E-07	1.00E-04	1.49E-10	1.00E-05	5.28E-07	2.00E-04	1.07E-05	2.00E-04	5.31E-08	1.00E-05
Xe-135m	9.55E-11	9.00E-06	1.15E-09	9.00E-05	4.69E-36	9.00E-06	0	1.80E-04	3.86E-07	1.80E-04	1.89E-09	9.00E-06
Xe-138	2.93E-10	4.00E-06	2.94E-09	4.00E-05	5.86E-33	4.00E-06	0	8.00E-05	1.29E-06	8.00E-05	6.51E-09	4.00E-06
I-131	4.69E-11	2.00E-08	4.23E-09	2.00E-07	1.97E-14	2.00E-08	1.10E-10	4.00E-07	2.44E-10	4.00E-07	1.97E-11	2.00E-08
I-132	4.34E-12	3.00E-06	3.11E-10	3.00E-05	6.72E-19	3.00E-06	4.56E-13	6.00E-05	8.79E-11	6.00E-05	4.78E-12	3.00E-06
I-133	3.96E-11	1.00E-07	4.90E-09	1.00E-06	2.41E-15	1.00E-07	1.73E-11	2.00E-06	3.92E-10	2.00E-06	2.09E-11	1.00E-07
I-134	2.14E-13	2.00E-05	9.03E-11	2.00E-04	1.23E-23	2.00E-05	9.60E-14	4.00E-04	5.86E-11	4.00E-04	2.98E-12	2.00E-05
I-135	9.20E-12	7.00E-07	1.60E-09	7.00E-06	1.50E-16	7.00E-07	3.36E-12	1.40E-05	2.22E-10	1.40E-05	1.07E-11	7.00E-07

1. Derived Air Concentrations for 40 hours/week occupancy are from 10 CFR Part 20, Appendix B, Table 1, Column 3; other DACs are multiples of these values.

TABLE 12.2-2 (SHEET 1 OF 2)

ASSUMPTIONS USED TO ESTIMATE PEAK AIRBORNE CONCENTRATIONS AND INHALATION DOSES

Leak Rates (lb/day)	
Steam generator tube leak (primary coolant)	166.9
Leak into containment (primary coolant)	40
Leak into auxiliary building (primary coolant)	166.9
Steam leak into turbine building	6×10^4
Liquid leak into turbine building	1.5×10^5
Leak from waste gas processing system	100 sf^3/year
Partition Factors or Ratio of Liquid Activity to Airborne Activity (iodines)	
Steam generator	100
Air ejector	10,000
Liquid leakage to turbine building	100
Liquid leakage to auxiliary building	100
Containment building, primary coolant leakage	100
Leakage from waste gas processing system (partition in the volume control tank)	100
Ventilation (ft^3/min)	
Exhaust rate from turbine building	5000
Flowrate for recirculation in the turbine building	11,500
Containment purge rate (Main/Mini)	25,000/2500
Preaccess filter system in containment, flowrate for recirculation	20,000
Exhaust rate from waste gas processing region of radwaste area	3500
Exhaust rate from waste monitor tank rooms (region of radwaste area containing waste holdup, floor drain, and waste monitor tanks)	660
Exhaust rate from radwaste area (excluding the waste gas processing region and waste monitor tank rooms)	4.90×10^4
Filter Efficiency (percentage)	
Halogen recirculation filter efficiency in containment	90

TABLE 12.2-2 (SHEET 2 OF 2)

Occupancy in the Regions (h/week; weeks/year)

Turbine building	40; 50[a]
Waste gas processing area	2; 50[a]
Waste monitor tank rooms	2; 50[a]
Radwaste area (excluding waste gas processing region and waste monitor tank rooms)	40; 50[a]
Containment during refueling or shutdown purge	40; 4

Volumes of the Regions (ft^3)

Turbine building	4.25×10^6
Containment	2.05×10^6
Waste gas processing region	38,000
Waste monitor tank rooms	8400
Radwaste area (excluding waste gas processing region and waste monitor tank rooms)	6.2×10^5

Other Factors

Fuel defects (percent)	0.25
Plant load factor (percent)	100
Duration of the containment purge, hot shutdown (h)	24
Duration of the preaccess filter operation (h)	40[b]
Duration of the containment refueling shutdown purge (h)	8

a. Full occupancy a year means 50 weeks/year for plant load factor 1.0; 40 weeks/year for plant load factor 0.8.

b. For refueling or shutdown purge, recirculation through preaccess filters will be for a total of 16 + 8 = 24 h. Purging at power is done by a constant 2500-ft^3/min flow through the minipurge system.

TABLE 12.2-3

INHALATION DOSES DUE TO AIRBORNE RADIOISOTOPES

Region	Occupancy	Lung Dose (rem/year)	Thyroid Dose (rem/year)
Turbine building	40 h/week for 50 weeks/year[a]	6.13×10^{-4}	0.141
Containment during hot shutdown purges	4 h/purge for 3 purges/year	3×10^{-3}	0.144
Containment during refueling or shutdown purge	40 h/week for 40 weeks/year	3.56×10^{-9}	1.56×10^{-6}
Containment total		3×10^{-3}	0.145
Auxiliary building waste gas processing region	2 h/week for 50 weeks/year[a]	3.95×10^{-5}	1.48×10^{-2}
Auxiliary building waste monitor tank rooms	2 h/week for 50 weeks/year[a]	2.74×10^{-4}	4.14×10^{-2}
Auxiliary building radwaste area excluding waste gas processing region and waste monitor tank rooms	40 h/week for 50 weeks/year[a]	3.26×10^{-4}	6.14×10^{-2}

a. For plant factor 1.0, occupancy is 50 weeks/year; for plant factor 0.8, occupancy is 40 weeks/year equivalent.

12.3 RADIATION PROTECTION PROGRAM

12.3.1 PROGRAM OBJECTIVES

12.3.1.1 Objectives

It is the objective of the Farley Nuclear Plant Radiation Protection program to provide effective radiation protection for plant personnel and visitors during operations, maintenance, refueling, and emergencies, and further to keep exposures as low as reasonably achievable (ALARA). The Radiation Protection group is responsible for developing and administering such a program consistent with 10 CFR 20, Standards for Protection Against Radiation, paragraph 20.1101.

12.3.1.2 Organization

The Radiation Protection group consists of a radiation protection manager, radiation protection support superintendent, radiation protection operations superintendent, plant health physicist, radiation protection supervisors, and radiation protection technicians. Overall responsibility for plant operation lies with the plant manager, but the responsibility for radiation protection operations is delegated to the radiation protection manager. (See figure 13.1-6.)

The radiation protection manager reports to the plant manager and is responsible for keeping him informed at all times of radiation hazards and conditions related to potential exposure, contamination of plant equipment, or contamination of site and environs. As the administrator of the radiation protection program, the responsibilities of the radiation protection manager include:

A. Training and supervising the radiation protection technicians, supervisors, health physicist, and radiation protection superintendents.

B. Planning and scheduling radiation protection coverage and surveillance activities.

C. Establishing and maintaining data on plant radiation and contamination levels, personnel exposures, and work restrictions.

D. Writing and maintaining current radiation protection procedures which incorporate the provisions of Regulatory Guides 8.8, Revision 3, and 8.10, Revision 1 (this should include radiation protection reviews of appropriate design changes).

E. Ensuring that plant operations comply with 10 CFR 20.1001 - 20.2401.

F. Advising the emergency director during emergencies involving radiological hazards.

G. Advising other group supervisors with regard to dose equalization among their personnel. Personnel will be rotated insofar as practical for uniformity of occupational radiation exposure within each group.

H. Managing the shipment and disposal of all solid radwaste.

I. Supervising the ALARA program.

To carry out the responsibilities of the radiation protection manager, the Radiation Protection group is organized to:

A. Perform radiation monitoring for plant operations and maintenance activities as required and maintain records of all surveys performed.

B. Establish and maintain a radiological surveillance program to collect and document data concerning radiation and contamination levels throughout the plant and on the plant site.

C. Make plant personnel aware of radiological conditions by posting areas throughout the plant based on radiation and contamination levels.

D. Provide and maintain protective clothing and respiratory equipment for plant operation and maintenance and instruct plant personnel in their use.

E. Recommend procedures for dealing with radiation hazards in performing day-to-day operation and maintenance and verify effectiveness of such procedures.

F. Specify dosimetry requirements for radiation work.

G. Assist in the plant training program by providing specialized training in radiation protection when necessary to support the training group.

H. Make recommendations and assist in performing equipment, area, and personnel decontamination.

I. Assist with the receipt and shipment of radioactive materials to ensure compliance with Federal and State regulations.

J. Ensure that radiation protection equipment designated for service is operational and calibrated.

K. Establish and implement an active ALARA program.

12.3.1.3 Personnel Qualification and Training

The radiation protection manager will have qualifications equivalent to those in Regulatory Guide 1.8, Revision 1, September 1975 (Personnel Selection and Training).

Each permanent plant employee who is classified as a radiation worker is required to attend radiation protection training, the depth of which will depend on the work assignments, individual responsibilities, and the degree of radiation hazard anticipated. Personnel whose duties entail entering restricted areas are required to attend training and to demonstrate a minimum level of knowledge prior to being permitted unescorted access to restricted areas.

Requiring individuals to demonstrate a minimum level of knowledge in radiation protection ensures that each individual is qualified to perform his duties safely. Further, plant supervisory personnel are advised to screen their employees with respect to conscientiousness and responsibility in performing their duties with regard for approved radiation protection procedures.

12.3.1.4 Plans and Procedures

To ensure that the internal and external occupational exposure is kept ALARA during the activities of maintenance, inspection, refueling, and nonroutine as well as routine operations, radiation protection procedures are utilized. When practical, these procedures covering the areas listed below are developed in accordance with Regulatory Guide 8.8, Revision 3, Information Relevant to Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable, and Regulatory Guide 8.2, Revision 0, Guide for Administrative Practices in Radiation Monitoring.

A. Personnel monitoring.

B. Personnel, equipment, and area decontamination.

C. Access to controlled areas.

D. Use and cleaning of protective clothing.

E. Use of respiratory protection equipment.

F. Air and liquid sampling.

G. Radiation work permit applicability and use.

H. Portable and fixed radiation protection equipment calibration.

I. Area and process radiation monitoring calibration.

J. Receipt and shipment of radioactive materials.

K. Leak testing of radioactive sources in accordance with the Technical Requirements Manual.

12.3.2 FACILITIES AND EQUIPMENT

12.3.2.1 Facilities

The radiation protection facilities consist of a radiation protection office, briefing room, survey preparation room, and material and personnel frisking room located in the auxiliary building access control area at el 155 ft.

Clothing issue rooms, a calibration lab, a respirator issue room, a decontamination room, a drumming room, and a nuclear laundry are located in the auxiliary building radiation controlled area at el 155 ft. The radiation protection office is located near the boundary between the clean area and the radiation controlled area so that radiation protection services and decontamination may be conveniently provided to those who enter or leave this area. Personnel decontamination can be performed in the hot toilet rooms which are conveniently located adjacent to the radiation protection office. Radiation controlled area entry is through an administratively controlled one-way door. Prior to leaving the radiation controlled area, one passes through a portal monitor or uses friskers near the radiation protection office.

12.3.2.2 Shielding and Handling Methods

Shielding (e.g., lead, tungsten) in various forms, such as bricks, blankets, or sheets will be available for use as portable shielding. (A safety evaluation checklist must be completed before shielding can be applied to any safety-related equipment or systems.)

A radiation work permit will be employed as the principal means of ensuring that proper precautions are taken and that adequate planning is effected before work is performed in any area that presents a real or potential radiological hazard. Prior to a worker's entry into an area in which the radiological conditions are unknown, a survey is made and a radiation work permit completed which lists the radiation protection requirements for the particular work to be accomplished.

Other handling methods and procedures for keeping external and internal exposures ALARA are discussed in subsections 12.1.5 and 12.2.5.

12.3.2.3 Respiratory Equipment

Respiratory equipment will be available for use in areas in which airborne radioactive material exceeds those concentrations given in Table 1, Appendix B to 10 CFR 20.1001 - 20.2401. Typical respiratory devices which will be made available at the plant include the following:

A. Full-face masks with high efficiency particulate and charcoal filters.

B. Full-face masks with air line.

C. Hoods and suits with air line.

D. Full-face masks with self-contained breathing apparatus.

E. Full-face masks with battery-powered high efficiency particulate filters.

The respiratory protection program is designed to comply with 10 CFR 20.1701 - 20.1704. Respiratory equipment is selected and protection factors are assigned in accordance with 10 CFR 20.1001 - 20.2401, Appendix A. An exemption from 10 CFR 20 which allows the use of a protection factor for radioiodine has been granted to Farley Nuclear Plant by the NRC. Any changes to the October 23, 1984, NRC exemption will be incorporated into the program.

12.3.2.4 Protective Clothing

Protective clothing will be required in contaminated areas. Typical protective clothing that will be made available at the plant is listed below:

A. Coveralls.

B. Laboratory coats.

C. Plastic suits.

D. Canvas caps.

E. Hoods.

F. Shoe covers.

G. Booties.

H. Gloves.

12.3.2.5 Portable Instrumentation

The majority of the portable radiation protection instrumentation will be located in the auxiliary building near the radiation protection office or the radiation protection calibration laboratory. For purposes of emergency monitoring, instruments will be kept at various places as designated by emergency preparedness procedures. A listing and description of some of the portable radiation protection instruments are given in table 12.3-1.

The Radiation Protection group will be responsible for writing and implementing procedures for the use and calibration of this equipment. Detailed records on the maintenance and calibration of this instrumentation will be maintained at the plant. Calibration will be performed using sources of known strength purchased from the National Institute of Standards and Technology (NIST) or other reputable vendors and/or using reference instruments having calibrations traceable to the NIST. In addition, reputable vendors will be used to calibrate and perform maintenance on some of the portable instruments. Vendors will implement their own calibration procedures but are subject to Southern Nuclear Operating Company (SNC) quality assurance requirements. Calibrations and preventive maintenance on portable radiation protection instrumentation will be performed semiannually or when required. Calibration will also be required after a piece of equipment has undergone repair work which affects calibration.

12.3.2.6 <u>Laboratory Equipment</u>

Major fixed laboratory instrumentation will generally be located at the radiation controlled area exit and the radiation protection counting room, but use is not limited to these areas. A listing of typical equipment, including location and description, is given in table 12.3-1 and table 12.3-2.

The Radiation Protection group will be responsible for writing and implementing procedures for the use and calibration of equipment. Detailed records on the calibration of this instrumentation will be maintained at the plant. Calibration will be performed using sources of known strength purchased from the NIST or other reputable vendors and/or using reference instruments having calibration traceable to the NIST. In addition, reputable vendors may be used to calibrate and perform maintenance on fixed laboratory instrumentation. Vendors will implement their own calibration procedures but are subject to SNC quality assurance requirements. Calibration will also be required after a piece of equipment has undergone repair work which affects calibration. The equipment and instrumentation listed in tables 12.3-1 and 12.3-2 are typical of the devices which will be purchased.

12.3.3 PERSONNEL DOSIMETRY

Where applicable, the personnel dosimetry program will be developed in accordance with Regulatory Guide 8.4, Revision 0, Direct Reading and Indirect Reading Pocket Dosimeters, and Regulatory Guide 8.13, Revision 2, Instructions Concerning Prenatal Radiation Exposure.

12.3.3.1 <u>External Dosimetry</u>

Plant employees, visitors, support personnel, and construction workers will be required to wear one or more personnel dosimeters when they enter the radiation control area if they are likely to receive, in 1 calendar year, from sources external to the body, a dose in excess of 10 percent of the limits in 10 CFR 20.1201(a). A third party may be used or a complete in-house program may be implemented for processing dosimetry badges (e.g., OSLDs).

To minimize congestion during outages and other peak activity periods, issuance and storage of contractor dosimetry badges and subsequent monitoring may be established at an alternate location other than the primary access point.

Personnel dosimetry used at the plant will include a dosimetry badge and either a digital alarming dosimeter or pocket ion chamber. The dosimetry badge must be sensitive to beta-gamma radiation and the dosimeter must be sensitive to gamma radiation. The dose received on dosimeters and dosimetry badges will be tracked by plant personnel. Extremity dosimeters will be issued on a case-by-case basis, and neutron dosimetry will be accomplished by setting dose rates and time keeping, which must be performed by a qualified individual, or by issuing neutron dosimetry.

12.3.3.2 Internal Dosimetry

Whole body counting and bioassay will be used to supplement the dosimetry program. If internal dose assessment is deemed necessary, calculations will meet the intent of Regulatory Guide 8.34.

12.3.3.3 Records

Exposure data of all personnel will be collected and recorded on form NRC-5, Occupational Exposure Received for a Monitoring Period, or the equivalent. Occupational exposures incurred by individuals prior to working at the Farley Nuclear Plant, bioassay data, and whole body counting data will be summarized on form NRC-4, Cumulative Occupational Exposure History, or the equivalent. Records retained on form NRC-4 or its equivalent will be retained until the license is terminated. The records used in preparing form NRC-4 or its equivalent will be kept for 3 years, after which time they may be disposed of. Exposure data recorded on form NRC-5 or its equivalent will be retained until the license is terminated.

TABLE 12.3-1

PORTABLE AND SEMIORTABLE RADIATION PROTECTION INSTRUMENTS

Instrument	Radiation Detected	Range	Accuracy	Number	Location	Remarks
GM survey meter (portable)	Beta, gamma	0-70,000 cpm 0-50 mR/h	±10% full scale	15	Instrument locker	Equipped with end window, side window, or pancake probe
GM survey meter (semiportable)	Beta, gamma	0-500,000 cpm	±10% full scale	10	Various areas in plant	Equipped with pancake probe for smear checks and personnel frisking
GM survey meter	Gamma	0-1000 R/h	±25% full scale	2	Instrument locker	Extendible probe
Neutron survey meter	Thermal through fast neutrons	0-5000 mR/h	±10% full scale	2	Instrument locker	Detection of neutrons up to 10 MeV
Ion chamber survey meter	Beta, gamma	0-20,000 R/h	±10% full scale	2	Instrument locker	Ionization chamber
Proportional alpha counter	Alpha	0-500,000 cpm	±10% full scale	2	Instrument locker	Scintillation counter
Ion chamber survey meter	Beta, gamma	0-5 R/h	±10% full scale	9	Instrument locker	Ionization chamber
Ion chamber survey meter	Beta, gamma	0-50 R/h	±10% full scale	5	Instrument locker	Ionization chamber
High volume air samplers	-	0-30 cfm	±10%	6	Instrument locker	Air sampling
Low volume air samplers	-	0-100 cfm	±10%	10	Instrument locker	Air sampling

TABLE 12.3-2

FIXED INSTRUMENTATION

Instrument	Radiation Detected	Sensitivity(a)	Number	Location	Remarks
Automatic smear counter (gas proportional)	Alpha, beta, gamma		1	Radiation protection designated area	Used primarily for counting smears to determine contamination levels
Multiple channel analyzer	Gamma		1	Radiation protection designated area	Used for identification of isotopes
Small articles monitor/tool monitor	Gamma	Alarm ≥ 5000 dpm/100 cm^2	2	Radiation protection designated area	Plastic scintillator detectors, RCA exit survey
Contamination monitors - whole body	Beta, gamma	Alarm ≥ 5000 dpm/100 cm^2	2	RCA exit	Flow proportional detectors for RCA exit survey
Portal monitor	Gamma	Alarm 75 nCi of Cs-137^2	1	RCA exit	Plastic scintillator detectors for RCA exit survey
Portal monitor	Gamma	Alarm 75 nCi of Cs-137^2	2	PESB exit point	Plastic scintillator detectors for MPBPA exit
Contamination monitor - hand, cuff, and foot surface	Alpha, beta	Alarm ≥ 5000 dpm/100 cm^2	1	RCA exit toilets	Plastic scintillator detector, used to survey prior to entrance to the RCA exit restrooms

a. Instrument sensitivities will comply with measuring and reporting requirements of NRC IE Circular 81-07.

12.4 RADIOACTIVE MATERIALS SAFETY

12.4.1 MATERIALS SAFETY PROGRAM

Sealed and unsealed sources may be used at the Farley Nuclear Plant to calibrate reactor excore detectors, process and effluent radiation monitoring systems, area radiation monitoring systems, portable survey instruments, and fixed laboratory equipment. Storage and handling of these sources will be in accordance with 10 CFR 20, 30, 37, 40, and 70, with the radiation protection group being responsible for the control of such sources. A Nuclear Regulatory Commission license will be obtained for byproduct, source, or special nuclear material, as appropriate, prior to the procurement of radioactive sources.

High level sources such as those listed in table 12.4-1 will normally be housed in lockable, shielded containers and stored in an area that has been approved by the radiation protection group. Low level sources that are primarily used for calibration and quality control checks of fixed laboratory instruments and for portable survey instrument check sources are normally stored in locked cabinets located in radiation protection approved areas (e.g. on el 139 ft in the radiochemistry laboratory and counting room, or on el 155 ft in the radiation protection calibration laboratory at the training center).

Other information pertinent to the handling and use of radioactive sources is contained in subsections 12.3.1.4, 12.3.2.2, and 12.3.3.

12.4.2 FACILITIES AND EQUIPMENT

A discussion of the facilities utilized by the radiation protection groups is given in subsection 12.3.2.1. The radiochemistry laboratory, where unsealed radioactive sources would normally be stored, is equipped with two exhaust hoods that exhaust to the plant vent. The sampling room and gas analysis room are also equipped with one such hood each.

A discussion of portable radiation protection instrumentation is given in paragraph 12.3.2.5, with a listing and description of each instrument given in table 12.3-1. A discussion of fixed laboratory instrumentation is given in paragraph 12.3.2.6, with a listing and description of each instrument given in table 12.3-2.

12.4.3 PERSONNEL AND PROCEDURES

The radiation protection manager, the radiation protection support superintendent, and the plant health physicist are the key personnel responsible for handling and monitoring radioactive materials at the plant. The qualifications of these personnel are given in paragraph 13.1.3.1.

The qualification requirements for radiation protection supervisors, who direct/oversee the work activities of the radiation protection technicians, meet or exceed the minimum requirements set forth in American National Standards Institute (ANSI) N18.1-1971. The minimum qualification requirements for radiation protection supervisor are given in paragraph 13.1.3.1.

The qualification requirements for the health physicist/radiation protection technician, who reports to the radiation protection support superintendent and is responsible for activities related to radioactive material and radioactive waste shipments, meet or exceed the minimum requirements set forth in ANSI N18.1-1971. The minimum qualification requirements for the health physicist/radiation protection technician are given in paragraph 13.1.3.1.

The qualification requirements for radiation protection technicians, who handle and monitor radioactive materials under the direction of the radiation protection support superintendent or supervisors, meet or exceed the minimum requirements set forth in ANSI N18.1-1971. The minimum qualification requirements for radiation protection technicians are given in paragraph 13.1.3.1.

Procedures have been developed by the radiation protection group to cover the receipt, storage, and use of radioactive sources. These procedures are discussed in the group training sessions to ensure that all technicians who are required to handle radioactive sources are thoroughly familiar with the procedures.

12.4.4 REQUIRED MATERIALS

A list of sources that are likely to be purchased is given in table 12.4-1. This table includes a listing of isotope, quantity, form, and use for byproduct source and special nuclear material sources. At the time of procurement of radioactive sources that exceed the quantity listed in table 12.4-1, an amendment will be made to the table, if necessary, per Farley Operating License, Section 2.B. and as per 10CFR20.

TABLE 12.4-1

RADIOACTIVE SOURCES

Material	Isotope	Quality	Form	Use
Byproduct	Cs-137	300 μCi	Sealed source	Low range calibration of gamma radiation monitoring equipment
	Cs-137	700 Ci	Sealed source	Intermediate and high range calibration of gamma radiation monitoring equipment
	Co-60	1 μCi	Sealed source	Low level calibration of portal monitors
	Co-60	36 μCi	Sealed source	Intermediate and high range calibration of gamma radiation monitoring equipment
Special nuclear	Pu-239	8.5 Ci	Pu-Be sealed source	Calibration of neutron radiation monitoring equipment
	Pu-239	5 μCi	Pu-Be sealed	Calibration of alpha radiation monitoring equipment
	Am-241	3.5 Ci	Am-Be	Source for in-line boron analysis instrumentation (one source per unit)