U.S. NUCLEAR REGULATORY
COMMISSION
DOCKET 50-410
LICENSE NPF-69

NINE MILE POINT NUCLEAR STATION UNIT 2

UPDATED SAFETY ANALYSIS REPORT

OCTOBER 2016

REVISION 22

Chapter 12

LIST OF EFFECTIVE FIGURES

Figure No.	Revision Number	Figure No.	Revision Number
12.3-1	R19	12.3-37	R15
12.3-2	R15	12.3-38	R15
12.3-3	R19	12.3-39	R20
12.3-4	R15	12.3-40	**R22**
12.3-5	R15	12.3-41	R21
12.3-6	R20	12.3-42	**R22**
12.3-7	R15	12.3-43	R17
12.3-8	R21	12.3-44	R15
12.3-9	**R22**	12.3-45	R15
12.3-10	R17	12.3-46	R21
12.3-11	R15	12.3-47	**R22**
12.3-12	R15	12.3-48	**R22**
12.3-13	R21	12.3-49	R15
12.3-14	**R22**	12.3-50	**R22**
12.3-15	**R22**	12.3-51	**R22**
12.3-16	R15	12.3-52	R15
12.3-17	**R22**	12.3-53	R15
12.3-18	**R22**	12.3-54	R03
12.3-19	R15	12.3-55	R02
12.3-20	R15	12.3-56	R02
12.3-21	R15	12.3-57	R15
12.3-22	R15	12.3-58	R15
12.3-23	R02	12.3-59	R15
12.3-24	R15	12.3-60	R10
12.3-25	R15	12.3-61	R15
12.3-26	R15	12.3-62	R15
12.3-27	R15	12.3-63	R03
12.3-28	R15	12.3-63a	R02
12.3-29	R15	12.3-64	R02
12.3-30	R15	12.3-65	R15
12.3-30a	R15	12.3-66	R20
12.3-31	**R22**	12.3-67	R00
12.3-32	R15	12.3-68	A00
12.3-33	R20	12.3-69 Sh 1	R00
12.3-34	R19	12.3-69 Sh 2	R19
12.3-35	R02	12.3-69 Sh 3	R00
12.3-36	R07	12.3-69 Sh 4	R05

CHAPTER 12

RADIATION PROTECTION

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

CHAPTER 12

TABLE OF CONTENTS (Cont'd.)

CHAPTER 12

TABLE OF CONTENTS (Cont'd.)

CHAPTER 12

LIST OF TABLES

LIST OF TABLES (cont'd.)

EXHIBIT 12.1-1

RESPONSIBILITIES OF ALL WORKERS

CHAPTER 12

LIST OF FIGURES (cont'd.)

CHAPTER 12

LIST OF FIGURES (cont'd.)

CHAPTER 12

LIST OF FIGURES (cont'd.)

CHAPTER 12

RADIATION PROTECTION

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

12.1.1 Policy Considerations

The Station policy is to commit sufficient resources to ensure that exposure of Station personnel, the general public, and the environment will be as low as reasonably achievable (ALARA).

The responsibility and authority to establish and implement a program to maintain occupational and environmental radiation exposure ALARA is vested with the Vice President Nine Mile Point.

Responsibility and authorities include:

1. Ensuring that a program that integrates management philosophy and regulatory requirements is established, including specific goals and objectives for implementation.

2. Ensuring that an effective measurement system is established and used to determine the degree of success achieved by Station operations with regard to the program goals and specific objectives.

3. Ensuring that measurement system results are reviewed on a periodic basis and that corrective actions are taken when attainment of the program goals or specific objectives appears to be jeopardized.

4. Ensuring that the authority is delegated for providing procedures and practices by which specific goals and objectives will be achieved.

5. Ensuring that the resources needed to achieve goals and objectives to maintain occupational radiation exposures ALARA are made available.

Implementation of the ALARA concept by means of a strong radiation protection program has been delegated to the Plant General Manager and General Supervisor Radiation Protection.

These positions have the authority to require adherence to Station procedures generated to accomplish objectives of this policy. Nine Mile Point Nuclear Site Administrative Procedures describe departmental and personnel responsibilities in detail and provide for direct recourse to corporate management. The site organization chart shown on Figure 13.1-2 shows the reporting of the radiation management personnel on site.

Unit 2 management is committed through its radiation protection program to maintaining occupational radiation exposures ALARA. Details of ALARA concepts are described in Regulatory Guides (RG) 8.8 and 8.10, many aspects of which are incorporated in Station procedures.

As part of the training program (which meets RG 1.8) described in administrative procedures for site employees and contractor employees, aspects of the radiation protection program dealing with maintaining exposures ALARA will be discussed. Employees will be made aware of their responsibilities (Exhibit 12.1-1) and management's commitment to achieve this end. Exhibit 12.1-2 outlines the Nine Mile Point respirator usage policy.

12.1.1.1 Management Audit and Review

The Station ALARA Committee will:

1. Review selected procedures as described in the administrative procedures; consider the radiation aspects of their procedures to ensure that radiation exposures are kept ALARA.

2. Review unusual radiation exposures or incidents, past exposure records, exposure trends, and major sources of radiation exposure throughout the site.

3. Hold discussions with the radiation protection staff regarding steps to be taken to reduce exposures.

4. Review the degree of success achieved in meeting specific ALARA goals, and take corrective action when the attainment of specific objectives appears to be jeopardized.

EXHIBIT 12.1-1

RESPONSIBILITIES OF ALL WORKERS

1. Obey promptly "stop work" and "evacuate" instruction of radiation protection personnel.

2. Follow all procedures.

3. Wear thermoluminescent dosimeter (TLD)/film badge and pocket or electronic dosimeter where required by procedures, signs, or radiation protection personnel.

4. Keep track of your own radiation dose status and avoid exceeding dose limits.

5. Remain in as low radiation areas as practicable to accomplish work.

6. Do not loiter in radiation areas or airborne radioactivity areas; use "wait areas" when designated.

7. Do not smoke, eat, drink, or chew in controlled surface contamination areas.

8. Wear anticontamination clothing and respirators properly and wherever required by signs, radiation work permits (RWP), radiation protection personnel, and procedures.

9. Remove anticontamination clothing and respirators properly to minimize spread of contamination.

10. Frisk yourself or be frisked for contamination as directed when leaving a controlled surface contamination area.

11. For a known or possible radioactive spill, minimize its spread and notify radiation protection personnel promptly.

12. Do not unnecessarily touch a contaminated surface or allow your clothing, tools, or other equipment to do so.

13. Place contaminated tools, equipment, and solid waste on disposable surfaces (e.g., place on sheet plastic when not in use and inside plastic bags when work is finished).

14. Limit the amount of material that has to be decontaminated or disposed of as radioactive waste.

15. Report the presence of treated or open wounds to radiation protection personnel prior to work in areas where radioactive contamination exists and immediately exit if a wound occurs while in such an area.

EXHIBIT 12.1-1 (Cont'd.)

EXHIBIT 12.1-1 (cont'd.)

RESPONSIBILITIES OF ALL WORKERS

16. Promptly report unsafe or noncompliance situations to plant management.

17. Report prior or concurrent occupational radiation exposure to the employer.

EXHIBIT 12.1-2

NINE MILE POINT RESPIRATORY PROTECTION

Sufficient resources shall be committed to limit the inhalation of airborne radioactive materials by all personnel at the site through the implementation of a respiratory protection program in accordance with 10CFR20 and Institute of Nuclear Power Operations (INPO) guidelines. The main objectives of this program shall be: 1) to minimize the use of respiratory protective equipment by using practical engineering controls where possible, 2) to provide necessary and sufficient respiratory equipment in emergency situations for workers, 3) to minimize the overall duration that respirators are used and minimize the time they are worn continuously, and 4) to provide respirator use relief in case of equipment malfunction, physical or psychological distress, procedural or communication failure, deterioration of operating conditions, and any other condition that might require such relief.

Prior to respiratory protective device use, all personnel shall: 1) receive a comprehensive respiratory physical accompanied by a psychological evaluation, 2) receive respiratory equipment specific training, 3) be clean shaven, and 4) be quantitatively-fit tested for specific respiratory protective devices.

1. The General Supervisor Radiation Protection is responsible for the maintenance and implementation of the respiratory protection program and to ensure that all personnel adhere to Station procedures and practices needed to accomplish such a program. The Plant General Manager is responsible to ensure that personnel receive necessary training in respirator use.

2. The Manager Training is responsible to develop and implement necessary training programs for respirator use.

3. The General Supervisor Radiation Protection is responsible for evaluating total hazards of a job, requesting engineering evaluation, if appropriate, prescribing appropriate respirator protective devices and, if conditions warrant, forbidding the use of respirators.
4. The Manager Engineering Services is responsible to provide design and other technical support for the respiratory protection program.

12.1.1.2 Responsibilities

Responsibility for implementing the ALARA philosophy and commitment and the radiation protection program will rest with the General Supervisor Radiation Protection and members of his staff. This General Supervisor reports to the Plant General Manager who is responsible for all aspects of Station operations including the radiation protection program. Departmental supervisors are also responsible to the Plant General Manager for the radiation protection program within the Station, and support the Radiation Protection Organization in formulating and implementing a Station program for maintaining radiation exposures ALARA.

12.1.1.3 Authority to Prevent Unsafe Practices

The General Supervisor Radiation Protection and his staff have direct recourse to the Vice President Nine Mile Point to prevent unsafe practices and halt unsafe operations. The General Supervisor Radiation Protection also contacts higher levels of management if required to resolve questions related to the radiation protection program.

12.1.1.4 Modifications to Operating, Surveillance, and Maintenance Procedures and Station Equipment

Based on the review performed (Section 12.1.1.1), operating, surveillance, and maintenance procedures will be reviewed by the ALARA Group when a significant reduction in radiation exposure appears to be possible. Station equipment and facilities will also be modified when such modification will substantially reduce radiation exposures at a reasonable cost. Site Engineering reviews modifications to the facility for ALARA concerns during the design process.

12.1.1.5 Vigilance by the Radiation Protection Staff

It will be the responsibility of the radiation protection staff to conduct surveillance programs and investigations to ensure occupational exposures are as far below specified limits as is reasonably achievable. Additionally, they will be vigilant in searching out new and better ways to perform all radiation work with less exposure.

12.1.1.6 General Supervisor Radiation Protection

To fulfill his responsibilities for implementation of the radiation protection program, the General Supervisor Radiation Protection will:

1. Review routine and special reports received from his supervisors and other members of his staff.

2. Make quarterly inspections of the Station to ensure that procedures dealing with radiation protection are being properly implemented. These include exposure control, training, survey and monitoring data collection, and recordkeeping.

3. Review origins of radiation exposures including locations, operations, and job conditions, and become aware of trends of exposures by specific job functions.

4. Investigate any unusual incident involving significant radiological hazards.

5. Make recommendations to management for changes to procedures and equipment that affect radiation safety, particularly when situations are identified in which exposures can be reduced.

6. Administer the radiation protection program to assure compliance with federal regulations.

7. Participate in design reviews for facilities and equipment that can affect potential radiation exposures.

8. Participate in development and approval of training programs related to work in radiation areas or involving radioactive materials.

9. Make routine and special reports to the Plant General Manager regarding status of all radiation protection operations, surveys, special problems, and unusual conditions.

10. Make, or cause to be made, weekly routine inspections of the Station to ensure that procedures dealing with radiation protection are being properly implemented. Submit this report to the Plant General Manager.

11. Make recommendations regarding changes in procedures and equipment, especially those pertaining to reduction of exposure or better methods of measuring exposure.

12.1.1.7 Supervisor Radiation Protection Operations

Under the general direction of the General Supervisor Radiation Protection, the Supervisors Radiation Protection Operations are responsible for performance of radiological assessments and surveys on a daily basis in support of Operations and Maintenance.

12.1.1.8 Supervisor Radiological Engineering

The Supervisor Radiological Engineering, who reports to the General Supervisor Radiation Protection, will review RWPs and man-Rem reports and will identify work functions that could benefit from an ALARA review. This individual is responsible for presenting annual man-Rem reports to the General Supervisor Radiation Protection, and such interim reports as may be required to document radiation exposure, and produce data useful to ALARA functions. In conjunction with appropriate department supervisors, and in accordance with established procedures, the Supervisor Radiological Engineering will provide evaluations of routine and special tasks and recommend appropriate revisions to procedures. In addition, the Supervisor Radiological Engineering is responsible for the internal and external dosimetry programs and all associated dosimetry services as required.

12.1.2 Design Considerations

This section discusses the general methods and features that implement the policy considerations of Section 12.1.1. Detailed provisions for maintaining personnel exposures ALARA are presented in Sections 12.3.1, 12.3.2, and 12.5.3.

12.1.2.1 General Design Considerations for ALARA Exposures

The general design considerations and methods employed to maintain in-plant radiation exposures ALARA have two objectives:

1. Minimizing the amount of time plant personnel spend in radiation areas.

2. Minimizing radiation levels in routinely occupied plant areas and in the vicinity of plant equipment expected to require the attention of plant personnel.

Both equipment design and arrangement are considered in maintaining exposures ALARA during plant operations, including: normal operation, radwaste handling, normal maintenance, corrective maintenance, refueling, inservice inspection (ISI), and other events of moderate frequency and certain infrequently occurring events.

Equipment layout, shielding, penetrations, and piping locations are reviewed by engineering personnel, including radiation protection personnel, during the development of design drawings for implementation of the ALARA philosophy for minimizing occupational radiation exposures to plant personnel during normal plant operations and maintenance.

Aspects of the design that are inconsistent with the ALARA philosophy are resolved on a case-by-case basis in accordance with the guidelines in RG 8.8.

This review assures that plant design is such that occupational radiation exposures are maintained ALARA.

12.1.2.2 Equipment General Design Considerations for ALARA

General equipment design considerations to minimize the amount of time plant personnel spend in a radiation area include:

1. Reliability, durability, construction, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance.

2. Servicing convenience for anticipated maintenance or potential repair.

3. Provisions, where practical, to remotely or mechanically operate, repair, service, monitor, or inspect equipment (including ISI in accordance with ASME Section XI).

4. Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high.

General equipment design considerations directed toward minimizing radiation levels in proximity to equipment or components requiring the attention of personnel include:

1. Provision for draining, flushing or, if necessary, remote cleaning of equipment and piping containing radioactive material.

2. Design of equipment, piping, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps.

3. Utilization of welded connections, where feasible, instead of flanged or threaded connections, valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials. In some cases, radwaste equipment is flanged in order to minimize removal time.

4. Provisions for isolating equipment from radioactive process fluids.

12.1.2.3 Plant Layout General Design Considerations for ALARA

Plant general design considerations to minimize the amount of time personnel spend in radiation areas include:

1. Locating equipment, instruments, and sampling stations, which require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas.

2. Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment.

3. Providing, where practical, for transportation of equipment or components requiring service to a lower radiation area.

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

1. Separating radiation sources and occupied areas where practical (e.g., routing pipes or ducts containing potentially highly radioactive fluids through unoccupied areas).

2. Providing adequate shielding between radiation sources and access and service areas.

3. Locating equipment, instruments, and sampling sites in the lowest practicable radiation zones.

4. Providing central control panels to permit remote operation of all essential instrumentation and controls from lowest radiation zones practicable.

5. Within a system, separating highly radioactive equipment, where practical, from less radioactive equipment, instruments, and controls.

6. Providing means and adequate space for utilizing movable shielding for sources within the service area when required.

7. Providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practical.

8. Providing means for decontamination of service areas.

9. Providing remotely operated backflushable filter systems for highly radioactive radwaste and cleanup systems.

10. Providing labyrinth entrances to radioactive pump, equipment, and valve rooms, as required.

11. Providing adequate space in labyrinth entrances for easy access.

12. Maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.

13. Providing drainage control such as curbing and floors sloping to local drains, or sumps to limit the spread of leakage from radioactive liquid systems.

14. Providing controlled access to High Radiation Areas, (e.g., lockable doors that require controlled keys or card reader approval to unlock, located at access points to High Radiation Areas).

12.1.2.4 Decommissioning Design Considerations for ALARA

The radiation protection aspects of the plant design also aid in the ALARA aspects of decommissioning. These include:

1. Accessibility for maintenance or removal of equipment.

2. Shielding to provide protection during maintenance or during storage after termination of plant operations.

3. Provisions for draining, flushing, or decontaminating equipment or piping.

4. Separation of more radioactive equipment from less radioactive equipment.

5. Features to minimize crud buildup.

6. Coatings applied to surfaces likely to become contaminated to facilitate cleanup.

12.1.2.5 Examples of ALARA Improvements

Design improvements, which indicate man-Rem reductions during operation and maintenance, have been recommended to the ALARA Task Force and Review Committee. A specific example of improvements based on dose assessment, operational experience, and the ALARA design review is described as follows.

<u>Improvement Based on Dose Assessment</u>

Information on crud deposition from operating plants and an ALARA dose assessment resulted in the addition of shielding to some of the reactor building equipment and floor drain system components.

The reactor building equipment drain tanks, coolers, and pumps, and the drywell equipment and floor drain tanks, cooler, and pumps are shielded to provide radiation protection to adjacent areas. The shield design also provides for segregation of the tanks from other components to limit exposures during maintenance.

<u>Improvement Based on Operational Experience</u>

Operational experience with radwaste filters, sludge processing, and radwaste solidification systems was utilized in the design and selection of components for systems. Use was also made of studies of operating plant experiences with similar systems to determine system failure rates, down times, number of personnel and man-hours required to repair the systems, and the man-Rems associated with repairs.

Operating experience from other units and the ALARA design features of Unit 2 are utilized in the development and revision of Station operating and maintenance procedures and instructions to ensure that occupational radiation exposure is maintained ALARA.

Improvement Based on the ALARA Design Review

An example is the ALARA design review of the condenser area which resulted in a shielding design with a twofold purpose: 1) the shielding provides radiation protection for adjacent areas; and 2) the shielding provides segregation of components such that maintenance operations can be performed on any one condenser while the other two are operating so that personnel would not receive excessive doses.

12.1.3 Operational Considerations

Operational radiation protection objectives deal with access to radiation areas, exposure to personnel, and decontamination. Working on or near radioactive components requires planning, special methods, and criteria to keep occupational radiation exposure ALARA. Special training and detailed preparation of necessary equipment prior to jobs and debriefing following jobs contribute toward reduced exposures. Decontamination and the use of portable shielding also helps to reduce exposure.

Implementation of procedures and techniques is based upon operational criteria and experience.

12.1.3.1 Operational Objectives

The operational radiation protection objectives include those given in Section 12.1.2.2 and the following:

1. Accurate knowledge of Station designs.

2. Sufficient experienced personnel to direct and train other personnel.

3. Detailed job planning for high exposure work.

4. Job simulations to improve productivity on the job, thereby keeping exposure ALARA.

5. Debriefings after jobs to identify time-consuming work and to identify problems.

6. Improving procedures and techniques (defined in the following sections) for future jobs.

12.1.3.2 Implementation of Procedures and Techniques

The criteria and/or conditions under which various operating procedures and techniques for ensuring that occupational radiation exposure are ALARA for systems associated with radioactive liquids, gases, and solids, along with the means for planning and developing procedures for radiation exposure-related operations, are given in the following sections:

Ensuring that Occupational Radiation Exposures are as Low as Reasonably Achievable	12.1
Radiation Protection Program	12.5

12.2 RADIATION SOURCES

12.2.1 Contained Sources

12.2.1.1 General

Three types of radiation sources that occur in a nuclear plant are discussed in this section: primary radiation from the reactor core, secondary radiation sources resulting from nuclear reactions between the primary radiation and the reactor environment (activation products), and possible release of radioactive material from the reactor core (fission products). During normal operations, secondary sources and released radioactive materials are transported by either the reactor water or the steam to process components in the plant. This section discusses the design sources which are grouped by location and equipment type (e.g., primary containment, core sources). The following sections present the source data for various pieces of equipment throughout the plant. General locations of equipment are shown on the general plant arrangement drawings of Section 1.2.

The biological shield wall (BSW) and the drywell wall are the principal shields for radiation from the reactor core. The maximum expected neutron flux is a factor in the design of the thickness of these walls. Consideration is also given to gamma rays emitted due to thermalization of neutrons in the shields.

With the exception of the BSW and drywell wall, shielding designs are based on fission product and activation product sources consistent with Section 11.1 and Table 12.2-16. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience indicates the annual average is considerably less than the design. Activation products, principally N-16, are the principal source for shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transit decay, while at the same time providing for transient increase of the noble gas source, daughter product formation, and energy level of emission. Areas where fission products are significant relative to activation products include: the condenser offgas system downstream of the steam jet air ejector (SJAE), liquid and solid radwaste equipment, portions of the reactor water cleanup (RWCU) system, and portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

A review of the systems that contain radioactive fluid was performed to determine whether an increase in source terms due to crud deposition was required. Based on Unit 2 calculated contact dose rates and measured data[3] (which summarizes operating plant data for various systems and equipment), it was determined that various system components required an increased source term. Table 12.2-16 provides the additional contact dose rates which

are included in the shielding design analysis of the corresponding components.

12.2.1.2 Primary Containment (Drywell)

12.2.1.2.1 Reactor Vessel Sources

This section defines the reactor vessel model for the associated gamma and neutron radiation sources and provides data required for calculations beyond the vessel. The data selected were not chosen for any given computer program, but were chosen to provide information for any of several shield analysis program types.

Physical Data

Table 12.2-1 presents the physical data required to form a reactor vessel model. The data include core material volume fractions, thermal power, average power density, power peaking factors, and noncore average water densities.

Gamma Ray Source Energy Spectra

The energy spectra (presented in this section) include fission gamma rays, fission product gamma rays, and gamma rays resulting from inelastic neutron scattering and thermal neutron capture. The total gamma ray energy release rate in the core is estimated to be accurate to within ±10 percent, while the energy release rate above 6 MeV may be in error by as much as a factor of ±2.

Core Spectrum After Operation

Table 12.2-2 gives the gamma ray energy spectrum in MeV/sec-watt in spent fuel at selected time intervals after operation. The data were prepared from tables of fission product decay gamma energies fitted to integral measurements for operation times of 10^8 sec (approximately 3.2 yr). Shutdown sources in the core are found in the same manner as operating sources, by combining the shutdown spectrum with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the spectrum and the thermal power that the element contained during operation.

Around the Vessel - Gamma Ray and Fast Neutron Fluxes

Table 12.2-3 presents the calculated gamma ray energy fluxes and fast neutron fluxes with energy greater than 1 MeV at three points outside of the reactor vessel. The calculated data can vary by a factor of ±3. Also, the gamma ray energy fluxes in this section do not include any provisions for scattering from points outside of the vessel, nor are there any provisions for gamma ray fluxes from depositions of radioactive isotopes within the vessel.

12.2.1.2.2 Radioactive Sources in the Reactor Water, Steam, and Offgas

The radioactive sources in the reactor coolant and steam are discussed in Section 11.1, and the offgas sources are discussed in Section 11.3.3. These sections provide concentrations, during normal operation, of the radioisotopes leaving the reactor vessel and gases leaving the condenser.

12.2.1.3 Reactor Building

12.2.1.3.1 Reactor Water Cleanup System Sources

The radioactive sources in the RWCU system (Section 5.4.8) are the result of the activity in the reactor water in transit through the system or accumulation of radioisotopes removed from the water. Components for this system include regenerative and nonregenerative heat exchangers, pumps, valves, and filter demineralizers. All major components of the cleanup system are located in the reactor building. Activities for the cleanup system are given in Table 12.2-4. The contact dose rates obtained from these activities are augmented by the crud contribution given in Table 12.2-16.

12.2.1.3.2 Main Steam System Sources

All radioactive materials in the main steam system (MSS) result from radioactive sources carried over from the reactor during plant operation. In most of these components the source is dominated by N-16; where the N-16 has decayed, the other isotopes carried by the steam become significant (Section 12.2.1.4.1).

12.2.1.3.3 Residual Heat Removal System Sources

The design gamma source strengths in the residual heat removal (RHR) system (Section 5.4.7) were evaluated for system operation in the reactor shutdown mode. In this mode, the system recirculates reactor coolant to remove reactor decay heat, operating from approximately 2 hr after shutdown until the end of the shutdown period. When necessary, this system also may supplement the spent fuel pool cooling (SFC) system capacity. The RHR system is described in Section 5.4.7. The source in the RHR system is the activity in the volume of reactor water contained in the system. The source strengths in the RHR equipment 8 hr after shutdown are shown in Table 12.2-5. The contact dose rates obtained from these activities are augmented by the crud contribution given in Table 12.2-16.

12.2.1.3.4 Reactor Core Isolation Cooling System Sources

The radioactive sources in the reactor core isolation cooling (RCIC) system (Section 5.4.6) were evaluated for the system operating in the test mode. This system may be utilized during reactor shutdown if the main condenser or feedwater systems are

unavailable. The system is operated from the time of reactor shutdown for approximately 2 hr until a reactor pressure of 150 psia is achieved. Below 150 psia the RHR system is initiated to achieve cold shutdown.

During routine testing of the system, the source in the equipment is the activity of the steam driving the system turbine. This activity is dominated by N-16. The radiation source data used in the shield design of this system are shown in Table 12.2-6. The contact dose rates obtained from these source data are augmented by the crud contribution given in Table 12.2-16.

12.2.1.3.5 Fuel Pool Cooling and Cleanup System Sources

The radiation source for spent fuel is given in Section 12.2.1.2.1, Gamma Ray Source Energy Spectra - Core Spectrum After Operation, and Table 12.2-2. There are two major sources of radioactivity in the spent fuel pool: the mixing of the reactor coolant with the fuel pool water at the start of refueling, and the release of crud from the surface of the spent fuel assemblies during movement.

The activities in the SFC filter/demineralizer are determined from initial reactor coolant activities. First, the corrosion product activities of the reactor coolant are multiplied by 3 to account for an increase in soluble species during the first week following shutdown. Next, the activities are decayed for 24 hr and then diluted by the fuel pool water. This spectrum is fed to the filter/demineralizer (removal efficiency assumed 100 percent) and built up for 7 days. The spectrum obtained here is added to the spectrum generated by the original coolant activities decayed for 7 days and built up in the filter/demineralizer for 23 days. This sum is the activity used for shielding; this two-part method results in a conservative shield design.

The radioactivity in the SFC heat exchanger is arrived at by assuming the SFC filter/demineralizer removal efficiency is 90 percent. The long-lived corrosion product spectrum is determined by the 10 percent passed through. This distribution is normalized to give dose rates consistent with data from other plants due to the plateout and buildup of these corrosion products in the heat exchanger. These and the spent fuel storage pool, surge tank, and other activities shown in Table 12.2-7, are based on liquid activity and crud buildup, and are used for shielding determination.

All SFC components are located within the reactor building. The SFC system is described in Section 9.1.3.

12.2.1.3.6 Liquid Radwaste System Sources (Reactor Building)

Components of the liquid radwaste system (LWS) located in the reactor building include the LWS RWCU phase separator tanks and pumps (el 289 ft), the LWS spent fuel pool phase separator tank

and pump (el 289 ft), and various pipes and valves, including drain discharge piping. All other components are located in the radwaste building (Section 12.2.1.5.1).

12.2.1.3.7 Startup Sources

The four reactor startup sources are tubular source holders each containing a Californium-252 (Cf-252) capsule. Cf-252 produces neutrons by spontaneous fission. The Cf-252 capsules are shipped to the site in a special shielded cask to the source holders. They are then loaded into the reactor while being kept underwater. They provide a sufficient source of neutrons in the core to help test the neutron flux monitors (of different ranges) for proper operation and response before and during fuel loading. There are seven alternate locations for various tests. These sources and source holders may be removed from the core at the end of the first fuel cycle.

12.2.1.3.8 Traversing In-core Probe System Sources

Five traversing in-core probes (TIPs) provide neutron flux readings in the core for calibration of other instruments. The maximum radiation source for this system is given in Table 12.2-8. The radiation source is based upon the probe's location within the core and its residence time. As indicated in the tables, the system is divided into three sections for shielding calculations: the probe's fissionable material, nonfissionable material, and signal cable. Sources are provided for 100-sec irradiation and zero-sec decay.

12.2.1.4 Turbine Building

12.2.1.4.1 Turbine System Sources

Piping and equipment that contain main steam are sources of radiation due to the presence of N-16, the predominant source of activity during operation. Fission product gases (xenon and krypton), water activation products (O-19 and N-17), and the carryover of iodine and other fission products present in steam and condensate are considered additional activity sources. The carryover is conservatively assumed to be 2 percent by weight for halogens and 0.1 percent by weight for other fission products. The N-16 concentrations in the turbine system equipment are listed in Table 12.2-9 for EPU conditions. Transit decay of the short-lived N-16 has been included in the source terms.

12.2.1.4.2 Condensate and Feedwater System Sources

The sources in the condensate system are based on decayed main steam activities. These systems are described in Section 10.4.

The condensate demineralizer (CND) system is designed to maintain high purity of the condensate to assure high-quality feedwater to the reactor by removal of various contaminants. Each CND is a

tank filled with resins that serves to remove ionic impurities and to filter out suspended solids from the condensate. Resin can be reused if the collected crud is removed. This is accomplished by ultrasonic resin cleaning.

Almost all noncondensables entering the condenser are removed by the air ejectors. Because of this and the relatively long holdup time, the N-16 and other gaseous activity is very low in the hotwell and negligible in the remainder of the condensate system. The activities caused by activated corrosion and fission products are shown in Table 12.2-10 for the condensate system.

12.2.1.4.3 Offgas System Sources

The gaseous effluent treatment system (Section 11.3.2) is designed to limit offsite doses from routine Station releases. The offgas system contains sources of radiation based on the holdup of noncondensable gases from the main condenser. These gases are removed by the condenser air removal system and are treated through the use of equipment including catalytic recombiners and ambient temperature charcoal adsorption decay beds. The N-16 activity in the air removal system is shown in Table 12.2-9 for EPU conditions. The N-16 activity in the offgas system (downstream of the holdup pipe) is negligible due to decay. Therefore, the predominant radiation sources are the fission product gases, xenon and krypton, and their daughter products. The source terms used for shielding are given in Table 12.2-11.

12.2.1.5 Radwaste Building

12.2.1.5.1 Liquid Radwaste System Sources (Radwaste Building)

The LWS (Section 11.2.2) is composed of four subsystems designed to collect, treat, and then recycle or discharge different categories of potentially radioactive waste water. The activities from sources in the LWS such as pipes, tanks, filters, demineralizers, and evaporators used in shielding calculations are listed in Table 12.2-12.

12.2.1.5.2 Solid Radwaste System Sources

The solid radwaste system is designed to collect, hold, monitor, process, package, and provide temporary storage facilities for solid radioactive wastes prior to shipment for offsite disposal. This system is described in Section 11.4. Radioisotopic inventories of major components in this system that are used in shielding calculations are listed in Table 12.2-13.

12.2.2 Airborne Radioactive Material Sources

Nine Mile Point Nuclear Station - Unit 2 (Unit 2) is designed so that the airborne radionuclide concentration in normally occupied areas is well below the limits discussed in 10CFR20. Radiation

zoning is discussed in Section 12.3.1.2. Areas that are designated Radiation Zone IV, V, or VI are considered to be normally occupied areas. Table II, column 1, of Appendix B to 10CFR20 provides the criteria used for Zone V and VI areas. Table I, column 1, of Appendix B to 10CFR20 provides the criteria used for a Zone IV radiation area.

Radioactive materials become airborne through evaporation and attachment to suspended water droplets and water vapor. The water vapor comes from leaks in high-energy lines (pressurized hot water). Suspended water droplets are created by sprays and splashing. Evaporation occurs wherever there is standing water exposed to air. The level of airborne radioactivity is periodically determined by the radiation protection staff to ensure that radiation exposure is ALARA.

12.2.2.1 Method for Computing the Airborne Radionuclide Concentration in a Plant Area

The method used for computing the airborne radionuclide concentrations for areas in the reactor, turbine, and radwaste buildings is based on data given in NUREG-0016, Revision 1, and EPRI-495[1,2]. The activity releases (in uCi/yr) from a building are distributed throughout the various building areas specified as follows by evaluating the ventilation system design along with the distributions recommended in NUREG-0016 and EPRI-495. These radionuclide concentrations result in the expected annual releases specified in Table 11.3-1 after credit is taken for filtration of radioiodine and particulates.

For both power operations and shutdown, 100 percent of the reactor building release is assumed to come from the areas containing the RHR, RWCU, and emergency core cooling system (ECCS) equipment and components.

For full-power operations, 85 percent of the activity released from the turbine building is from the main condenser area, which is not normally occupied. The remaining 15 percent of the turbine building release is from miscellaneous areas, including the SJAE area, the turbine operating floor, the feedwater pump area, and the mechanical vacuum pump area. Noble gas concentrations during full-power operations in the turbine operating floor, feedwater pump area, and the mechanical vacuum pump area are expected to be negligible.

The radwaste building is divided into three major areas contributing to the total activity release. These are the waste collection system areas, the floor drain collection system areas (both of which are LWS areas), and the solid waste system areas.

The LWS consists of four subsystems: waste collection, floor drain, regenerant waste, and phase separator (as discussed in Section 11.2.2). The calculations of airborne radionuclide concentrations are made with the regenerant waste system areas

and processing capability considered as part of the waste collection system, and with the phase separator system areas and processing capability considered as part of the solid waste system.

For power operations, 72 percent of the radwaste building releases come from the waste collection system areas; 18 percent, from the floor drain collector system areas; and 10 percent, from the solid waste system areas. The breakdown of releases from the LWS areas is based on the expected processing capability of the subsystems.

12.2.2.2 Production of Airborne Radioactive Material Sources

The primary potential sources of airborne radioactivity during normal operations are:

1. Leakage from process equipment in radioactive systems, such as valves, flanges, and pumps.

2. Evaporation from sumps, drains, tanks, and filter/demineralizer vessels that contain radioactive fluid, except where vapor is hard-piped to a heating, ventilating and air conditioning (HVAC) system.

3. Exhaust from relief valves.

4. Gases released during removal of reactor pressure vessel (RPV) head and associated internals.

5. Evaporation and gases released from sampling.

6. Airborne radioactivity released from the spent fuel pool water and spent fuel movement.

7. Maintenance activities such as decontamination.

Sections 12.2.2.2.1 to 12.2.2.2.6 discuss each of these sources and their effect on the airborne radionuclide concentrations in normally-occupied areas. Design features that serve to reduce these concentrations are also discussed. Tables 12.2-15a and 12.2-15b present the airborne radioactivity concentrations expected in reactor building, radwaste building, and turbine building areas for both power operations and shutdown corresponding to EPU conditions.

Abnormal occurrences that can cause airborne radiation include: 1) spills (i.e., overflows and splashing), 2) failure of a ventilation system, 3) cracks in piping, 4) failures of pump and valve seals, and 5) malfunctioning equipment.

Airborne radioactivity is expected in the decontamination area, occasionally in labs, and during refueling on the refueling floor. The airborne radioactivity is caused by leaks, spills,

venting, decontamination, etc.; concentrations are calculated for the occurrences that are the most common, leaks and venting.

12.2.2.2.1 Effect of Leakage from Process Equipment in Radioactive Systems

Normally there is no leakage from equipment in radioactive systems. If a leak occurs, its effect is determined by three items.

If a leak occurs from a component located in a high radiation area, it does not contribute to the airborne radionuclide concentration in a normally-occupied area because the plant HVAC systems are designed to provide airflow from areas of lesser to progressively greater potential radioactive contamination prior to final exhaust. Separate HVAC systems are provided for each building to aid in the isolation of contamination. Areas where radiation levels vary are regarded as having a high potential for airborne radioactivity. Air contained in these areas is treated as if the highest potential of radioactivity exists, so that it will not affect the airborne radionuclide concentration of a normally-occupied area.

The system operating pressure affects the possible leakage rate. A system such as the MSS, which operates at high pressure, is expected to have a higher leakage rate than a system that operates at atmospheric pressure. However, radioactive systems that operate at high pressure are located in high radiation areas. Thus, these systems do not significantly contribute to the airborne radioactivity level in normally-occupied areas due to the HVAC airflow discussed earlier.

A system that can leak highly-radioactive fluid, such as reactor coolant or main steam, is of greater concern, initially, than a system that can leak a less contaminated fluid such as condensate storage tank (CST) water. Systems that can leak highly-radioactive fluid are located in high radiation areas, and this activity is not transported to lower radiation areas by HVAC airflow, as discussed earlier.

There are many systems containing radioactive fluids that are located in low radiation areas. Each system has been evaluated to determine the effective potential of creating an airborne radiation hazard. Design features have been incorporated to reduce the possibility of hazard. These measures include hard piping all relief valves on the auxiliary boiler steam system to a contact condenser. Gaseous effluents from this tank are vented to the auxiliary boiler building ventilation system, which is ducted to a monitored release point.

12.2.2.2.2 Effect of Sumps, Drains, and Tanks

The floor and equipment drains in the reactor, radwaste, and turbine buildings are designed to collect and transport various

types of waste to the LWS for processing. These drain systems convey waste by gravity to their respective sumps; waste is pumped from the sumps to the radwaste building. Drains and sumps in the systems noted are not significant sources of airborne radionuclides for the following reasons:

1. Each sump is covered with a steel plate. The free volume in the sump is maintained at a negative pressure with respect to the surrounding area by the use of a riser vent, which is connected to the HVAC system in the building of concern. The steel plate covering the sump does not provide an airtight seal. Air is drawn into the sump around the edges of the steel cover and exhausted through the riser into the HVAC system. Any radionuclides that escape into the free volume of the sump are discharged to the HVAC system and do not escape into the area surrounding the sump.

2. To prevent crud buildup, the drains empty by gravity with no water traps or level pipe runs. Air is drawn through the drains to the sump by the same riser vent discussed in the previous paragraph and out to the HVAC system.

Sump 2DFT-SUMP2H receives condensate effluent intermittently from the main steam line drain to the condenser. This sump is not vented to the building ventilation system because there is no flashing concern which could result in airborne radioactivity due to the temperature being less than 212°F. Also, noble gases, which could become airborne even in the absence of flashing, are expected to be negligible.

The holding tanks and filter/demineralizer units that contain significant inventories of radionuclides are hard piped to the HVAC system. These tanks and filter/demineralizer vessels are located in high radiation areas. Even if any airborne radionuclides were released from these components, there would be no effect on normally-occupied areas due to the HVAC system design features discussed in Section 12.2.2.3.1.

12.2.2.2.3 Effect of Relief Valve Exhaust

The relief valves found in the various plant systems which can exhaust radioactive fluids are not considered a significant source of airborne radioactivity in normally-occupied areas for the following reasons:

1. The exhaust of many relief valves is piped directly to the condenser, with no access to the atmosphere.

2. The exhaust of other relief valves is either piped directly to the suppression pool or to the floor or equipment drains. These drains lead to the LWS (Section 12.2.2.2.2).

12.2.2.2.4 Effect on Removing RPV Head and Associated Internals

Experience at boiling water reactor (BWR) plants has shown that an inventory of radioactive gases will accumulate inside the reactor vessel head between the time of shutdown and head removal. These gases consist primarily of the longer-lived radiohalogens and noble gases. To prevent these gases from being released to the refueling area, provisions are made for the venting of the gases to the HVAC system prior to RPV head removal. These gases are vented through an 8-in diameter duct connected between the reactor head and reactor head evacuation filter assembly containing particulate and charcoal filter units.

Experience at Dresden and Quad-Cities stations has shown that some airborne radioactivity can result from the following:

1. When the reactor water in the reactor cavity goes above 100°F, noticeable increases in the I-131 airborne activity result, increasing with temperature.

2. When the reactor water level in the vessel is low, previously covered metallic surfaces dry out. If cobalt dioxide (CoO_2) is plated out on these surfaces, air moving across these surfaces can dislodge fine particles of CoO_2. The dryer and separator are also susceptible to this phenomenon.

These two airborne activity problems have been solved by maintaining water temperatures below 100°F and by making provisions for clean water services to the RPV cavity area.

This permits wetting of the RPV cavity and components. By following these procedures, it is anticipated that RPV head and reactor internals removal will have a minimal effect on the airborne radionuclide level in the spent fuel area.

12.2.2.2.5 Effect of Sampling

The possibility of releasing radionuclides that could become airborne during sampling operations is recognized. Design features are incorporated into the sample system to limit the radionuclide release. Radioactive fluids that require frequent grab sampling are piped via sample lines to fume hoods located in sample rooms. Grab sampling will be accomplished in the fume hoods. During sampling, an inflow air velocity of approximately 100 ft/min will be maintained to sweep any airborne radioactive particles to the exhaust duct. Administrative control is used by following procedures when process fluids are sampled. This minimizes the release of radioactive fluids and, hence, exposure to personnel during the sampling process.

12.2.2.2.6 Effect of Spent Fuel Movement

Experience at operating BWR plants has shown that fuel movement normally does not present any unusual radiological problems. The expected level of radioactivity in the spent fuel pool water is listed in Table 12.2-7; this includes activity due to crud buildup. Evaporation of the spent fuel pool water is the major possible contributor to airborne activity, but is not expected to be significant.

12.2.2.2.7 Effect of Solid Radwaste Handling Areas

The solid radwaste handling equipment located in the radwaste building is designed for semiremote operation. Entry for maintenance activities will normally entail shutdown and flushing of systems and equipment.

The ventilation supply for the radwaste building is filtered outside air. The airflow around all components that are possible sources of airborne radioactive contamination is ducted directly to the HVAC system. Expected airborne radioactivity concentrations in the solid radwaste handling areas are provided in Table 12.2-15.

12.2.2.2.8 Effect of Liquid Radwaste Handling Areas

Low maintenance-type equipment is designed for the LWS, located in the radwaste building. All components that contain radioactive materials and are located near normally occupied areas are shielded. The ventilation of the liquid radwaste handling areas is similar to that of the solid radwaste system. The supply air is filtered outside air, and the airflow around components that are possible sources of airborne radioactivity is return ducted either individually or by cubicle directly to the HVAC system. Expected airborne radioactivity concentrations in liquid waste handling areas are provided in Table 12.2-15.

12.2.2.3 References

1. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code), NUREG-0016, Revision 1, January 1979.

2. Sources of Radioiodine at Boiling Water Reactors, EPRI 495, February 1978.

3. Hazzan, M. J.; Stocknoff, M. S.; Barcomb, D.; Irving, T. Radiation Levels Due to Crud Deposition in Boiling Water Reactors. Presented at American Nuclear Society 1983 Winter Meeting, San Francisco, CA.

TABLE 12.2-1

BASIC REACTOR DATA

A. Reactor thermal power* 3,536 MWt

B. Average power density 50.7 watts/cm^3

C. Core power peaking factors:

1. At core center: $\left.\frac{P_{avg}}{P_{max}}\right|_Z$ (axial) 1.5

   $\left.\frac{P_{avg}}{P_{max}}\right|_R$ (radial) 1.4

2. At core boundary: $\left.\frac{P_{avg}}{P_{max}}\right|_Z$ (axial) 0.5

   $\left.\frac{P_{avg}}{P_{max}}\right|_R$ (radial) 0.7

D. Core volume fractions:

Material	Density (g/cc)	Volume Fraction
UO_2	10.4	0.254
Zr	6.4	0.140
H_2O	1.0 (liquid)	0.274
Void	0	0.332

E. Average water density between core and vessel and below the core 0.74 g/cc

F. Average water-steam density above core:

1. In the plenum region 0.23 g/cc
2. Above the plenum (homogenized) 0.6 g/cc

G. Average steam density 0.036 g/cc

* A power level of 102 percent of rated power (3,467 MWt) is chosen to ensure that the calculated design bases sources bound the expected sources in the reactor at the licensed rated power condition.

TABLE 12.2-2

POSTOPERATION GAMMA SOURCES IN CORE*

(MeV/sec-watt)

Energy Bounds (MeV)	Time After Shutdown			
	0 sec	1 day	1 week	1 month
4.0-6.0	8.2+9	<1.0+6	<1.0+6	<1.0+6
3.0-4.0	1.8+10	7.0+6	4.6+6	<1.0+6
2.6-3.0	1.1+10	5.7+6	3.7+6	<1.0+6
2.2-2.6	1.7+10	2.9+8	1.7+8	<2.0+7
1.8-2.2	2.1+10	4.5+8	4.0+7	4.0+7
1.4-1.8	3.3+10	3.1+9	2.1+9	6.4+8
0.9-1.4	3.7+10	2.3+9	1.6+9	1.1+9
0.4-0.9	5.1+10	7.5+9	3.8+9	2.1+9
0.1-0.4	1.2+10	1.8+9	8.7+8	3.6+8

* Operating history of 3.2 yr.

NOTE: 8.2+9 = 8.2×10^9

TABLE 12.2-3

RADIATION LEVELS INSIDE DRYWELL

	Gamma Flux (Mev/cm^2-sec)*						Fast Neutron Flux (>1 Mev) (n/cm^2-sec)
Location	1 Mev	1.5 Mev	2.3 Mev	3 Mev	5 Mev	7 Mev	
A	9.4+2	6.9+3	9.4+4	9.3+5	1.2+6	4.7+6	4.3-1
B	3.0+7	1.7+8	1.1+9	7.4+9	2.2+9	4.4+9	1.0+7
C	7.4-4	3.2-1	1.2+2	4.5+3	4.0+4	3.1+5	4.7-7

* The flux levels represent direct core fluxes. Contributions caused by scattering from walls and surfaces outside the reactor vessel are not included.

NOTE: 9.4+2 = 9.4×10^2

KEY (All on outer surface):

A = Top of RPV
B = Side of RPV (at highest core axial flux height)
C = Bottom of RPV

TABLE 12.2-4

REACTOR WATER CLEANUP SYSTEM SOURCES

(uCi/cc)

Isotope	RWCU Regenerative Heat Exchanger	RWCU Nonregenerative Heat Exchanger	RWCU Filter/ Demineralizer
Kr-83m	1.23E-04	1.50E-04	4.20E+00
Xe-131m	4.93E-09	6.00E-09	4.99E+01
Xe-133m	8.68E-07	1.06E-06	6.00E+00
Xe-133	1.22E-05	1.48E-05	2.06E+02
Xe-135m	9.92E-04	1.20E-03	2.61E+01
Xe-135	1.56E-04	1.90E-04	8.70E+01
N-17	1.84E-06	2.88E-07	-
O-19	1.87E-01	1.40E-01	-
Nb-98	1.48E-02	1.48E-02	9.64E-01
Tc-104	3.19E-01	3.17E-01	7.14E+00
Br-83	2.29E-02	2.29E-02	4.20E+00
Br-84	2.85E-02	2.84E-02	1.14E+00
Br-85	1.38E-02	1.32E-02	4.39E-02
I-129	2.76E-18	3.37E-18	2.61E-13
I-131	1.30E-02	1.30E-02	1.36E+02
I-132	2.19E-01	2.19E-01	3.16E+02
I-133	1.60E-01	1.60E-01	2.56E+02
I-134	3.86E-01	3.85E-01	2.55E+01
I-135	1.70E-01	1.70E-01	8.70E+01
Np-239	2.40E-01	2.40E-01	1.03E+03
Rb-89	2.12E-02	2.10E-02	3.96E-01
Sr-89	3.10E-03	3.10E-03	5.06E+01
Sr-90	2.30E-04	2.30E-04	4.11E+00
Sr-91	6.89E-02	6.89E-02	5.13E+01
Sr-92	1.10E-01	1.10E-01	2.26E+01
Y-90	3.54E-08	4.31E-08	3.02E+00
Y-91m	4.72E-04	5.74E-04	3.08E+01
Y-91	1.10E-04	1.10E-04	6.17E+01
Y-92	1.93E-02	1.93E-02	2.27E+01
Y-93	1.20E-02	1.20E-02	7.72E-02
Y-94	-	-	2.73E-08
Zr-97	4.00E-05	4.00E-05	4.13E-02
Nb-95	4.20E-05	4.20E-05	7.53E-01
Nb-97m	1.69E-05	1.95E-05	3.96E-02
Nb-97	9.34E-08	1.30E-07	4.13E-02
Mo-99	2.20E-02	2.20E-02	1.09E+00
Tc-99m	2.80E-01	2.80E-01	1.30E+02
Tc-101	3.65E-01	3.61E-01	6.35E+00
Ru-103	5.40E-05	5.40E-05	8.59E-01
Ru-105	6.19E-03	6.18E-03	2.10E+00
Ru-106	8.40E-06	8.40E-06	1.49E-01

TABLE 12.2-4 (Cont'd.)

Isotope	RWCU Regenerative Heat Exchanger	RWCU Nonregenerative Heat Exchanger	RWCU Filter/ Demineralizer
Rh-103m	5.10E-07	6.20E-07	8.56E-01
Rh-105m	9.45E-04	1.07E-03	2.12E+00
Rh-105	1.39E-06	1.72E-06	4.74E-03
Rh-106	5.83E-06	6.42E-06	1.49E-01
Te-129m	1.10E-04	1.10E-04	1.72E+00
Te-129	6.04E-07	7.35E-07	1.70E+00
Te-131m	2.80E-04	2.80E-04	6.46E-01
Te-131	1.44E-06	1.75E-06	1.29E-01
Te-132	4.90E-02	4.90E-02	2.78E+02
Cs-134	1.60E-04	1.60E-04	1.61E+00
Cs-136	1.10E-04	1.10E-04	2.76E-02
Cs-137	2.40E-04	2.40E-04	4.28E+00
Cs-138	1.87E-01	1.86E-02	1.84E-03
Ba-137m	4.69E-05	5.58E-05	3.94E+00
Ba-139	1.59E-01	1.59E-01	1.69E+01
Ba-140	9.00E-03	9.00E-03	1.13E+02
Ba-141	1.65E-01	1.63E-01	3.73E+00
Ba-142	1.61E-01	1.59E-01	2.09E+00
La-140	2.20E-06	2.68E-06	9.82E+01
La-141	4.19E-04	5.08E-04	3.99E+00
La-142	1.80E-02	1.81E-02	1.99E+00
Ce-141	8.40E-05	8.40E-05	2.32E+00
Ce-143	8.40E-05	8.40E-05	2.13E-01
Ce-144	3.50E-05	3.50E-05	6.17E-01
Pr-143	1.10E-04	1.10E-04	1.51E+00
Pr-144	1.18E-06	1.43E-06	6.15E-01
Nd-147	1.40E-05	1.40E-05	1.67E-01
Pm-147	5.99E-12	7.30E-12	9.65E-04
N-13	4.71E-02	4.65E-02	-
F-18	3.98E-03	3.97E-03	7.12E+00
Na-24	4.10E-03	4.10E-03	4.73E+00
P-32	7.80E-05	7.80E-05	1.02E+00
Cr-51	2.30E-03	2.30E-03	3.48E+01
Mn-54	4.00E-05	4.00E-05	7.06E-01
Mn-56	4.98E-02	4.98E-02	9.85E+00
Fe-55	3.90E-04	3.90E-04	6.95E+00
Fe-59	8.00E-05	8.00E-05	1.29E+00
Co-58	5.00E-03	5.00E-03	8.38E+01
Co-60	5.00E-04	5.00E-04	8.93E+00
Ni-65	2.99E-04	2.99E-04	5.85E-02
Cu-64	1.20E-02	1.20E-02	1.19E+01
Zn-65	7.80E-05	7.80E-05	1.37E+00
Zn-69m	8.19E-04	8.19E-04	8.67E-01
Ag-110m	6.00E-05	6.00E-05	1.05E+00
Ag-110	5.96E-07	6.45E-07	2.10E-02

TABLE 12.2-4 (Cont'd.)

Isotope	RWCU Regenerative Heat Exchanger	RWCU Nonregenerative Heat Exchanger	RWCU Filter/ Demineralizer
W-187	3.00E-03	3.00E-03	5.51E+00
N-16	4.78E-03	1.61E-03	-
Zr-95	-	-	7.40E-01
Nb-95m	-	-	3.44E-03
Zn-69	8.46E-06	1.03E-05	-
Kr-85m	3.37E-05	4.02E-05	-
Kr-85	3.86E-13	5.65E-13	-

NOTE: 1.23E-04 = 1.23×10^{-4}

The values shown in Table 12.2-4 are based on a reactor thermal power of 3,489 MWt. Power uprate will increase the values by no more than 10%, with the exceptions of F-18 (1100%) and 0-19 (57%).

TABLE 12.2-5

RESIDUAL HEAT REMOVAL SYSTEM PUMP AND HEAT EXCHANGER SOURCE TERMS

Isotope	Activity[1] (uCi/cc)
Br-83	2.1-3[2]
I-131	1.2-2
I-132	5.7-2
I-133	1.2-1
I-134	6.2-4
I-135	7.0-2
Nb-95	3.9-5
Nb-97	2.3-5
Nb-97m	2.1-5
Nb-98	2.1-5
Ag-110	1.1-6
Ag-110m	5.6-5
W-187	2.2-3
Np-239	2.0-1
La-140	1.1-3
La-141	3.3-3
La-142	4.4-4
Ce-141	1.2-4
Ce-143	6.6-5
Ce-144	3.2-5
Pr-143	1.0-4
Pr-144	3.2-5
Nd-147	1.3-5
F-18	1.9-4
Na-24	2.6-3
P-32	7.1-5
Cr-51	2.1-3
Mn-54	3.7-5
Mn-56	5.4-3
Fe-55	3.7-4
Fe-59	7.4-5
Co-58	4.6-3
Co-60	4.6-4
Ni-65	3.2-5
Cu-64	7.2-3
Zn-65	7.2-5
Zn-69m	5.1-4
Zr-95	3.7-5
Zr-97	2.2-5
Sr-89	2.9-3
Sr-90	2.1-4
Sr-91	3.6-2
Sr-92	1.3-2

TABLE 12.2-5 (Cont'd.)

Isotope	Activity[1] (uCi/cc)
Y-90	1.7-5
Y-91	2.8-4
Y-91m	2.4-2
Y-92	3.0-2
Y-93	6.4-3
Mo-99	1.8-2
Tc-99m	1.1-1
Ru-103	5.0-5
Ru-105	1.6-3
Ru-106	7.8-6
Te-129	9.9-5
Te-129m	9.9-5
Te-131	4.4-5
Te-131m	2.2-4
Te-132	4.2-2
Cs-134	1.5-4
Cs-136	9.8-5
Cs-137	2.2-4
Cs-138	5.8-6
Ba-137m	2.0-4
Ba-139	2.7-3
Ba-140	8.2-3
Rh-103m	5.0-5
Rh-105m	1.6-3
Rh-106	7.8-6

(1) Values less than $1x10^{-6}$ are assumed to be negligible.
(2) 2.1-3 = $2.1x10^{-3}$

TABLE 12.2-6

REACTOR CORE ISOLATION COOLING SYSTEM
DESIGN ACTIVITIES

Isotope	Activity (uCi/cc)
Kr-83m	2.17-04
Kr-85m	3.74-04
Kr-85	1.19-06
Kr-87	1.32-03
Kr-88	1.33-03
Kr-89	7.58-03
Kr-90	1.16-02
Xe-131m	9.52-07
Xe-133m	1.80-05
Xe-133	5.10-04
Xe-135m	1.67-03
Xe-135	1.43-03
Xe-137	8.94-03
Xe-138	5.34-03
Xe-139	1.26-02
Xe-140	6.80-03
Br-83	1.19-05
Br-84	1.89-05
Br-85	1.06-05
I-131	8.84-06
I-132	1.12-04
I-133	8.84-05
I-134	2.64-04
I-135	9.52-05
Rb-88	1.75-05
Rb-89	1.22-04
Rb-90	1.29-03
Sr-89	4.11-08
Sr-90	3.00-09
Sr-91	1.09-06
Sr-92	2.24-06
Y-90	1.85-13
Y-91m	3.00-09
Y-91	3.74-09
Y-92	6.83-07
Y-93	4.08-07
Zr-95	7.82-10
Zr-97	6.12-10
Nb-95m	2.49-16
Nb-95	7.82-10
Nb-97m	1.23-10
Nb-97	3.17-13
Mo-99	3.03-07
Tc-99m	2.11-06

TABLE 12.2-6 (Cont'd.)

Isotope	Activity (uCi/cc)
Tc-101	1.31-05
Ru-103	1.90-09
Ru-105	2.17-07
Ru-106	2.89-10
Rh-103m	7.24-12
Rh-105m	1.65-08
Rh-105	1.85-11
Rh-106	1.09-10
Te-129m	4.42-09
Te-129	9.76-12
Te-131m	9.52-09
Te-131	1.98-11
Te-132	1.90-07
Cs-134	2.89-09
Cs-136	1.87-09
Cs-137	7.62-09
Cs-138	4.49-05
Cs-139	3.77-04
Cs-140	2.36-03
Ba-137m	6.36-10
Ba-139	4.32-06
Ba-140	1.34-07
Ba-141	5.37-06
Ba-142	5.34-06
La-140	1.14-11
La-141	5.44-09
La-142	6.26-07
Ce-141	2.89-09
Ce-143	2.89-09
Ce-144	4.42-10
Pr-143	3.74-09
Pr-144	6.02-12
Nd-147	2.89-10
Pm-147	4.96-17
N-13	2.33-04
N-16	4.59-01
N-17	4.05-05
O-19	1.18-02
F-18	1.36-04
Na-24	1.39-07
P-32	2.69-09
Cr-51	8.16-08
Mn-54	1.50-09
Mn-56	1.87-06
Fe-55	1.33-08
Fe-59	2.99-09
Co-58	1.87-07

TABLE 12.2-6 (Cont'd.)

Isotope	Activity (uCi/cc)
Co-60	1.87-08
Ni-65	1.12-08
Cu-64	4.42-07
Zn-65	2.69-09
Zn-69m	3.74-08
Zn-69	1.55-10
Ag-110m	2.24-09
Ag-110	1.28-11
W-187	1.12-07
Nb-98	3.81-09
Tc-104	1.14-05
Np-239	3.23-06

NOTE: 2.17-04 = 2.17×10^{-4}.

TABLE 12.2-7

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM SOURCE ACTIVITIES
(uCi/cc)

Isotopes	Activity in Piping Upstream of Filter/Demineralizers	Activity Filter/ Demineralizers	Activity in Piping Downstream of Filter/Demineralizers	Activity in Heat Exchanger
Np-239	1.86-02	1.75+02	1.86-03	1.86-03
Br-83	2.34-06	8.65-04	2.34-07	2.34-07
I-131	1.24-03	1.73+01	1.24-04	1.24-04
I-132	4.25-03	5.58+01	4.25-04	4.25-04
I-133	7.48-03	2.50+01	7.48-04	7.48-04
I-135	1.47-03	1.52+00	1.47-04	1.47-04
Sr-89	3.18-04	2.10+01	3.18-05	3.18-05
Sr-90	2.39-05	1.91+00	2.39-06	2.39-06
Sr-91	1.29-03	1.87+00	1.29-04	1.29-04
Sr-92	2.35-05	1.06-02	2.35-06	2.35-06
Y-90	5.48-06	1.71+00	5.48-07	5.48-07
Y-91m	8.44-04	1.20+00	8.44-05	8.44-05
Y-91	5.08-05	4.01+00	5.08-06	5.08-06
Y-92	2.71-04	1.67-01	2.71-05	2.71-05
Y-93	2.43-04	3.85-01	2.43-05	2.43-05
Zr-95	4.12-06	2.83-01	4.12-07	4.12-07
Zr-97	1.23-06	3.37-03	1.23-07	1.23-07
Nb-95m	1.44-08	1.72-03	1.44-09	1.44-09
Nb-95	4.36-06	3.36-01	4.36-07	4.36-07
Nb-97m	1.19-06	3.19-03	1.19-07	1.19-07
Nb-97	1.33-06	3.64-03	1.33-07	1.33-07
Mo-99	1.78-03	1.96+01	1.78-04	1.78-04
Tc-99m	3.40-03	2.06+01	3.40-04	3.40-04
Ru-103	5.52-06	3.46-01	5.52-07	5.25-07
Ru-105	1.52-05	1.08-02	1.52-06	1.52-06
Ru-106	8.72-07	6.77-02	8.72-08	8.72-08
Rh-103m	5.52-06	3.12-01	5.52-07	5.52-07
Rh-105m	1.52-05	3.03-03	1.52-06	1.52-06
Rh-106	8.72-07	6.77-02	8.72-08	8.72-08
Te-129m	1.12-05	6.76-01	1.12-06	1.12-06
Te-129	1.12-05	4.41-01	1.12-06	1.12-06
Te-131m	1.67-05	8.14-02	1.67-06	1.67-06
Te-131	3.39-06	1.82-02	3.39-07	3.39-07
Te-132	4.11-03	5.42+01	4.11-04	4.11-04
Cs-134	1.66-05	1.31+00	8.32-06	8.32-06
Cs-136	1.09-05	4.48-01	5.46-06	5.46-06
Cs-137	2.50-05	1.99+00	1.25-05	1.25-05
Ba-137m	2.30-05	1.88+00	2.30-06	2.30-06
Ba-140	8.87-04	3.61+01	8.87-05	8.87-05
La-140	3.08-04	3.68+01	3.08-05	3.08-05

TABLE 12.2-7 (Cont'd.)

Isotopes	Activity in Piping Upstream of Filter/Demineralizers	Activity Filter/ Demineralizers	Activity in Piping Downstream of Filter/Demineralizers	Activity in Heat Exchanger
La-141	2.06-05	1.35-02	2.06-06	2.06-06
Ce-141	1.52-05	9.19-01	1.52-06	1.52-06
Ce-143	5.28-06	2.83-02	5.28-07	5.28-07
Ce-144	3.63-06	2.79-01	3.63-07	3.63-07
Pr-143	1.12-05	4.94-01	1.12-06	1.12-06
Pr-144	3.63-06	2.80-01	3.63-07	3.63-07
Nd-147	1.36-06	5.11-02	1.36-07	1.36-07
Pm-147	1.02-09	7.45-04	1.02-10	1.02-10
F-18	5.68-08	1.70-05	5.68-09	5.68-09
Na-24	4.23-04	1.02+00	4.23-05	4.23-05
P-32	2.32-05	5.79-01	2.32-06	2.32-06
Mn-56	2.48-05	1.01-02	2.48-06	2.48-06
Fe-55	1.22-04	4.70+00	1.22-05	1.22-05
Ni-65	1.37-07	5.11-05	1.37-08	1.37-08
Cu-64	1.03-03	2.04+00	1.03-04	1.03-04
Zn-69m	7.63-05	1.67+01	7.63-06	7.63-06
Ag-110	1.25-07	9.19-03	1.25-08	1.25-08
W-187	4.68-04	1.78+00	4.68-05	4.68-05
Zn-65	3.96-02	9.17-01	3.96-03	5.71-01
Cr-51	1.14+00	2.13+01	1.14-01	1.65+01
Mn-54	2.05-02	4.75-01	2.05-03	2.94-01
Ag-110m	1.02-02	7.08-01	1.02-03	1.47-01
Fe-59	4.02-02	8.17-01	4.02-03	5.79-01
Co-58	2.54+00	5.45+01	2.54-01	3.65+01
Co-60	2.56-01	6.06+00	2.56-02	3.68+00
Ba-139	1.01-07	2.04-05	1.01-08	1.01-08
La-142	3.61-08	2.05-05	3.61-09	3.61-09
Rh-105	-	3.32-01	-	-
Zn-69	-	1.80-01	-	-
Pu-239	-	4.86-04	-	-

NOTES:

1. 1.86-02 = 1.86×10^{-2}
2. Values less than 1.0×10^{-09} are assumed to be negligible.

TABLE 12.2-8

TRAVERSING IN-CORE PROBE SYSTEM MATERIALS
AND RADIATION SOURCES

A. Material Composition of TIP System Components as Used in Activation Calculations

Material	Weight (g)
Detector Region	
AISI 304 stainless steel	4.0
Commercially pure titanium	3.0
Fosterite ceramic	0.5
Nichrome	0.02
Uranium-235	0.001

Cable Region	Weight/Length (g/in)
AISI 3041 stainless steel	0.12
AISI C1070 carbon steel	2.10
Magnesium oxide	0.12

B. TIP Detector Decay Gamma Activities in Mev/sec of 0.001 g of U-235

Decay Time = 0 sec
Activation Time = 10^{2} sec

Energy-MeV	MeV/Sec
0.1-0.4	3.4+9
0.4-0.9	1.5+10
0.9-1.35	1.2+10
1.35-1.8	1.1+10
1.8-2.2	8.0+9
2.2-2.6	6.4+9
2.6-3.0	5.6+9
3.0-3.5	5.1+9
3.5-4.0	4.3+9
4.0-4.5	2.5+9
4.5-5.0	1.5+9
5.0-5.5	7.9+8

TABLE 12.2-8 (Cont'd.)

C. TIP Detector Decay Gamma Activities of Materials in Detector (Excluding U-235) in uCi in Irradiated Detector

Decay Time = 0 sec
Activation Time = 10^2 sec

Activated Isotopes	uCi
Fe-59	1.1+1
Mn-56	1.7+5
Cr-51	7.0+1
Mn-54	2.1+0
Co-58m	3.5+3
Co-58	2.2-2
Ni-57	1.1-1
Co-57	6.0-7
Ni-65	4.0+2
Co-60m	7.6+3
Co-60	1.8-3
Co-61	9.6+0
Si-31	2.9+1

D. Decay Gamma Activities of Materials in the Cable in uCi/in of Irradiated Cable

Decay Time = 0 sec
Activation Time = 10^2 sec

Activated Isotopes	uCi/in
Fe-59	8.2+0
Mn-56	7.4+4
Cr-51	3.7+0
Mn-54	1.6+0
Co-58m	1.0+2
Co-58	6.5-4
Ni-57	3.3-3
Co-57	1.8-8
Ni-65	1.2+1
Co-60m	2.2+2
Co-60	5.1-5
Co-61	2.8-1
Si-31	8.7-1

NOTE: 3.4+9 = 3.4×10^9

TABLE 12.2-9

AVERAGE N-16 ACTIVITIES IN EQUIPMENT
IN THE TURBINE BUILDING

Component	Activity (uCi/g)
Equalizer tube	8.74+01
High-pressure turbine	8.07+01
Moisture separator/reheater, tube side	5.73+01
Moisture separator/reheater, shell side	7.13+01
Moisture separator drain tank	3.13-01
Reheater drain tank	8.60-02
Low-pressure turbine	7.36+01
Condenser	3.44+01
Air removal piping	3.46+00*
Second-point heater drain cooler	3.30-03
Third-point heater drain cooler	1.73-03
First-point heater, shell side	6.96+01
Second-point heater, shell side	6.81+01
Third-point heater, shell side	8.69+00
Fourth-point heater, shell side	1.43+00
Fifth-point heater, shell side	4.59+00
Sixth-point heater, shell side	5.86+00
Clean steam reboiler	2.64+01

NOTE: 8.56+01 = 8.56×10^{1}

*uCi/cc

TABLE 12.2-10

CONDENSATE SYSTEM SOURCE TERMS

Isotope	Hotwell (5-min Delay) and Piping Upstream of Condensate Demineralizer Activity (uCi/cc)	Condensate Demineralizer Activity (uCi/cc)*
Kr-83m	0.0	1.25-01
Kr-85m	0.0	7.69-04
Kr-85	0.0	1.59-06
Xe-131m	0.0	7.83-02
Xe-133m	0.0	2.47-01
Xe-133	0.0	8.49+00
Xe-135m	0.0	4.31-01
Xe-135	0.0	2.80+00
Br-83	3.32-04	1.25-01
Br-84	4.84-04	4.05-02
Br-85	1.02-04	7.69-04
I-129	0.0	1.35-11
I-131	2.60-04	7.86+00
I-132	3.12-03	1.74+00
I-133	2.59-03	8.50+00
I-134	7.21-03	9.96-01
I-135	2.68-03	2.80+00
Rb-89	1.75-05	7.00-04
Sr-89	3.10-06	3.18-01
Sr-90	2.30-07	3.38-02
Sr-91	6.86-05	1.03-01
Sr-92	1.08-04	4.62-02
Y-90	2.07-10	3.14-02
Y-91m	2.67-06	5.99-02
Y-91	1.11-07	6.15-02
Y-92	2.05-05	5.76-02
Y-93	1.19-05	1.89-02
Zr-95	4.00-08	4.41-03
Zr-97	3.19-08	8.50-05
Nb-95m	1.86-13	2.87-05
Nb-95	4.20-08	5.62-03
Nb-97m	2.92-08	8.05-05
Nb-97	1.10-09	8.53-05
Mo-99	2.20-05	2.29-01
Tc-99m	2.78-04	4.65-01
Tc-101	2.98-04	1.11-02
Ru-103	5.40-08	5.05-03
Ru-105	6.12-06	4.29-03

TABLE 12.2-10 (Cont'd.)

Isotope	Hotwell (5-min Delay) and Piping Upstream of Condensate Demineralizer Activity (uCi/cc)	Condensate Demineralizer Activity (uCi/cc)*
Ru-106	8.40-09	1.17-03
Rh-103m	2.90-09	4.55-03
Rh-105m	1.70-06	1.20-03
Rh-105	9.42-09	4.34-03
Rh-106	8.39-09	1.17-03
Te-129m	1.10-07	9.59-03
Te-129	3.46-09	6.23-03
Te-131m	2.79-07	1.32-03
Te-131	7.94-09	2.91-04
Te-132	4.90-05	6.05-01
Cs-134	1.60-07	2.29-02
Cs-136	1.10-07	5.18-03
Cs-137	2.40-07	3.52-02
Cs-138	1.71-04	1.45-02
Ba-137m	1.68-07	3.33-02
Ba-139	1.53-04	3.32-02
Ba-140	9.00-06	4.15-01
Ba-141	1.41-04	6.78-03
Ba-142	1.23-04	3.46-03
La-140	1.29-08	4.12-01
La-141	2.25-06	8.18-03
La-142	2.17-05	8.74-03
Ce-141	8.41-08	1.29-02
Ce-143	8.39-08	4.37-04
Ce-144	3.50-08	4.81-03
Pr-143	1.10-07	5.74-03
Pr-144	6.33-09	4.81-03
Nd-147	1.40-08	5.69-04
Pm-147	3.50-14	1.69-05
Na-24	4.08-06	9.69-03
P-32	7.80-08	3.95-03
Cr-51	2.30-06	1.82-01
Mn-54	4.00-08	5.53-03
Mn-56	4.89-05	1.99-02
Fe-55	3.90-07	5.62-02
Fe-59	8.00-08	7.89-03
Co-58	5.00-06	5.66-01
Co-60	5.00-07	7.28-02
Ni-65	2.93-07	1.17-04
Cu-64	1.20-05	2.40-02

TABLE 12.2-10 (Cont'd.)

Isotope	Hotwell (5-min Delay) and Piping Upstream of Condensate Demineralizer Activity (uCi/cc)	Condensate Demineralizer Activity (uCi/cc)*
Zn-65	7.80-08	1.06-02
Zn-69m	8.17-07	1.77-03
Ag-110m	6.00-08	8.17-03
Ag-110	7.80-10	1.06-04
W-187	2.99-06	1.13-02
Nb-98	1.40-05	1.88-03
Tc-104	2.72-04	1.29-02
Np-239	2.40-04	2.14+00
Zn-69	4.81-08	1.78-03
Pu-239	-	8.88-06
H-3	1.00-02	1.46+03
N-13	4.95-03	1.30-01
F-18	3.88-03	1.15+00

* Per cc of resin volume.

NOTE: 1.25-01 = 1.25×10^{-1}

TABLE 12.2-11

OFFGAS SYSTEM SOURCE ACTIVITIES

(Ci)

Isotope	Charcoal Adsorber Activities
Xe-131m	6.30+01
Xe-133m	2.78+02
Xe-133	1.87+04
Xe-135m	1.17+02
Xe-135	3.92+03
Xe-137	1.21+02
Xe-138	3.41+02
Cs-138	3.41+02
Kr-83m	1.15+02
Kr-85m	4.74+02
Kr-85	9.97+00
Kr-87	4.65+02
Kr-88	1.04+03
Kr-89	8.37+01
Rb-88	1.04+03

NOTE: 6.30+01 = 6.30×10^{1}

TABLE 12.2-12
LIQUID RADWASTE SYSTEM COMPONENTS DESIGN ACTIVITIES (uCi/cc)

Isotope	Spent Fuel Pool Phase Sepa-rator Tank	Rad-waste Filter	Back-wash Tank	Rad-waste Demin.	RWCU Phase Sep-arator Tank	Floor Drain Col-lector Tank	Floor Drain Filter	Waste Col-lector Tank	Waste Evapo-rator Sub-System	Recovery Sample Tank	Spent Resin Tank	Regen-erant Waste Tank	Regen-erant Evapo-rator Sub-System	Evapo-rator Bottoms Tank	Waste Dis-charge Sample Tank
Br-83	1.56-1	-	-	1.91-2	1.56-1	4.72-3	-	4.72-3	1.80-2	3.55-5	1.56-1	6.75-3	1.80-2	1.80-2	1.39-6
Br-84	4.22-2	-	-	1.18-3	4.22-2	1.33-3	-	1.33-3	2.73-4	4.51-6	4.22-2	4.62-4	2.73-4	2.73-4	2.11-8
Br-85	1.63-3	-	-	6.18-6	1.63-3	7.03-5	-	7.03-5	-	2.51-8	1.63-3	-	-	-	-
I-129	1.09-11	-	-	1.92-12	1.09-11	-	-	-	6.32-13	3.30-18	1.09-11	0.00+0	6.32-13	6.32-13	1.93-18
I-131	1.94+1	-	-	4.11+0	1.94+1	1.27-2	-	1.27-2	6.36+1	1.26-4	1.94+1	2.22+0	6.36+1	6.36+1	2.16-3
I-132	2.12+1	-	-	6.24+0	2.12+1	8.06-2	-	8.06-2	4.38+0	7.10-4	2.12+1	2.21-1	4.38+0	4.38+0	5.53-5
I-133	1.01+1	-	-	4.29+0	1.01+1	1.23-1	-	1.23-1	2.99+1	1.19-3	1.01+1	1.76+0	2.99+1	2.99+1	1.35-3
I-134	9.43-1	-	-	4.30-2	9.43-1	2.96-2	-	2.96-2	4.31-1	1.42-4	9.43-1	4.44-1	4.31-1	4.31-1	3.34-5
I-135	3.22+0	-	-	9.08-1	3.22+0	8.05-2	-	8.05-2	2.45+0	7.27-4	3.22+0	3.35-1	2.45+0	2.45+0	1.59-4
Rb-89	1.47-2	-	-	2.05-3	1.47-2	4.83-4	-	4.83-4	3.78-7	4.17-4	1.47-2	1.34-6	3.78-7	3.78-7	2.93-12
Sr-89	2.68+1	-	-	4.55+0	2.68+1	3.10-3	-	3.10-3	2.82+0	3.17-5	2.68+1	9.28-2	2.82+0	2.82+0	9.24-6
Sr-90	3.49+0	-	-	5.83-1	3.49+0	2.31-4	-	2.31-4	3.06-1	2.30-6	3.49+0	9.94-3	3.06-1	3.06-1	9.94-7
Sr-91	1.90+0	-	-	6.54-1	1.90+0	3.99-2	-	3.99-2	1.59-1	3.74-4	1.90+0	1.57-2	1.59-1	1.59-1	9.15-7
Sr-92	8.35-1	-	-	1.14-1	8.35-1	2.54-2	-	2.54-2	8.02-3	1.95-4	8.35-1	2.68-3	8.02-3	8.02-3	6.13-8
Y-90	3.43+0	-	-	5.60-1	3.43+0	1.97-5	-	1.97-5	2.89-1	2.20-7	3.43+0	9.27-3	2.89-1	2.89-1	9.33-7
Y-91m	1.15+0	-	-	4.25-1	1.15+0	2.22-2	-	2.22-2	1.04-1	2.30-4	1.15+0	1.01-2	1.04-1	1.04-1	6.00-7
Y-91	3.49+1	-	-	1.72+2	3.49+1	2.93-4	-	2.93-4	5.57-1	1.10-3	3.49+1	1.82-2	5.57-1	5.57-1	1.82-6
Y-92	8.41-1	-	-	2.81-1	8.41-1	2.84-2	-	2.84-2	3.28-2	2.69-4	8.41-1	6.30-3	3.28-2	3.28-2	2.40-7
Y-93	2.87-3	-	-	1.23-1	2.87-3	7.15-3	-	7.15-3	3.09-2	6.69-5	2.87-3	2.92-3	3.09-2	3.09-2	1.74-7
Y-94	1.01-9	-	-	-	1.01-9	-	-	-	-	-	1.01-9	-	-	-	-
Zr-95	-	-	-	6.52-2	-	3.99-5	-	3.99-5	3.93-2	3.99-7	-	1.29-3	3.93-2	3.93-2	1.29-7
Zr-97	1.58-3	-	-	6.49-4	1.58-3	2.32-5	-	2.32-5	2.44-4	2.21-7	1.58-3	1.62-5	2.44-4	2.44-4	1.17-9
Nb-95m	-	-	-	1.26-3	-	1.79-8	-	1.79-8	7.51-4	5.70-10	-	2.42-5	7.51-4	7.51-4	2.43-9
Nb-95	2.91-1	-	-	3.91-2	2.91-1	4.21-5	-	4.21-5	5.03-2	3.01-9	2.91-1	1.64-3	5.03-2	5.03-2	1.64-7
Nb-97m	1.51-3	-	-	6.24-4	1.51-3	2.19-5	-	2.19-5	2.34-4	2.13-7	1.51-3	1.56-5	2.34-4	2.34-4	1.12-9
Nb-97	1.58-3	-	-	6.93-4	1.58-3	2.14-5	-	2.14-5	2.63-4	2.18-7	1.58-3	1.74-5	2.63-4	2.63-4	1.26-9
Mo-99	6.93-2	-	-	2.26+0	6.93-2	2.02-2	-	2.02-2	1.52+0	2.00-4	6.93-2	6.03-2	1.52+0	1.52+0	5.54-6
Tc-99m	4.85+0	-	-	3.32+0	4.85+0	1.35-1	-	1.35-1	1.63+0	1.21-3	4.85+0	8.46-2	1.63+0	1.63+0	6.49-6
Tc-101	2.35-1	-	-	3.10-3	2.35-1	7.78-3	-	7.78-3	4.40-6	1.26-5	2.35-1	1.67-5	4.40-6	4.40-6	3.41-11
Ru-103	4.06-1	-	-	6.85-2	4.06-1	5.38-5	-	5.38-5	4.46-2	5.37-7	4.06-1	1.47-3	4.46-2	4.46-2	1.46-7
Ru-105	7.78-2	-	-	1.65-2	7.78-2	2.21-3	-	2.21-3	1.85-3	1.88-5	7.78-2	3.76-4	1.85-3	1.85-3	1.33-8
Ru-106	1.18-1	-	-	1.96-2	1.18-1	8.41-6	-	8.41-6	1.05-2	8.40-8	1.18-1	3.41-4	1.05-2	1.05-2	3.41-8
Rh-103m	4.06-1	-	-	6.86-2	4.06-1	4.46-5	-	4.46-5	4.46-2	5.15-7	4.06-1	1.47-3	4.46-2	4.46-2	1.46-7
Rh-105m	7.84-2	-	-	1.65-2	7.84-2	6.18-4	-	6.18-4	1.86-3	1.89-5	7.84-2	3.76-4	1.86-3	1.86-3	1.34-8
Rh-105	2.26-4	-	-	5.51-4	2.26-4	4.44-4	-	4.44-4	7.19-5	8.97-8	2.26-4	3.22-6	7.19-5	7.19-5	2.86-10
Rh-106	1.18-1	-	-	1.96-2	1.18-1	8.41-6	-	8.41-6	1.05-2	8.35-8	1.18-1	3.41-4	1.05-2	1.05-2	3.41-8
Te-129m	7.44-1	-	-	1.26-1	7.44-1	1.10-4	-	1.10-4	6.53-2	1.09-6	7.44-1	2.16-3	6.53-2	6.53-2	2.15-7
Te-129	7.44-1	-	-	1.26-1	7.44-1	6.41-5	-	6.41-5	6.62-2	1.03-6	7.44-1	2.79-3	6.62-2	6.62-2	2.21-7
Te-131m	2.80-2	-	-	1.17-2	2.80-2	2.33-4	-	2.33-4	6.07-3	2.27-6	2.80-2	3.02-4	6.07-3	6.07-3	2.51-8
Te-131	5.60-3	-	-	2.37-3	5.60-3	4.97-5	-	4.97-5	1.23-3	4.54-7	5.60-3	6.13-5	1.23-3	1.23-3	5.08-9
Te-132	1.95+1	-	-	5.93+0	1.95+1	4.57-2	-	4.57-2	4.11+0	4.51-4	1.95+1	1.59-1	4.11+0	4.11+0	1.48-5
Cs-134	1.32+0	-	-	3.90-1	1.32+0	1.60-4	-	1.60-4	2.06-1	8.00-5	1.32+0	6.71-3	2.06-1	2.06-1	6.71-7

TABLE 12.2-12 (Cont'd.)

Isotope	Spent Fuel Pool Phase Sepa- rator Tank	Rad- waste Filter	Back- wash Tank	Rad- waste Demin.	RWCU Phase Sep- arator Tank	Floor Drain Col- lector Tank	Floor Drain Filter	Waste Col- lector Tank	Waste Evapo- rator Sub- System	Recovery Sample Tank	Spent Resin Tank	Regen- erant Waste Tank	Regen- erant Evapo- rator Sub- System	Evapo- rator Bottoms Tank	Waste Dis- charge Sample Tank
Cs-136	6.19-3	-	-	5.90-2	6.19-3	1.08-4	-	1.08-4	4.54-2	5.39-5	6.19-3	1.54-3	4.54-2	4.54-2	1.51-7
Cs-137	3.64+0	-	-	6.08-1	3.64+0	2.41-4	-	2.41-4	3.17-1	1.20-4	3.64+0	1.03-2	3.17-1	3.17-1	1.03-6
Cs-138	6.82-5	-	-	7.94-3	6.82-5	8.82-3	-	8.82-3	1.01-4	1.51-3	6.82-5	1.69-4	1.01-4	1.01-4	7.82-10
Ba-137m	3.35+0	-	-	5.59-1	3.35+0	2.27-4	-	2.27-4	2.91-1	1.07-4	3.35+0	9.47-3	2.91-1	2.91-1	9.48-7
Ba-139	6.25-1	-	-	4.46-2	6.25-1	1.91-2	-	1.91-2	1.76-3	1.18-4	6.25-1	1.14-3	1.76-3	1.76-3	1.36-8
Ba-140	2.40+1	-	-	4.53+0	2.40+1	8.85-3	-	8.85-3	3.47+0	8.82-5	2.40+1	1.18-1	3.47+0	3.47+0	1.16-5
Ba-141	1.38-1	-	-	2.30-3	1.38-1	4.48-3	-	4.48-3	7.88-6	9.26-6	1.38-1	2.32-5	7.88-6	7.88-6	6.10-11
Ba-142	7.73-2	-	-	7.84-4	7.73-2	2.62-3	-	2.62-3	3.12-7	3.18-6	7.73-2	1.57-6	3.12-7	3.12-7	2.41-12
La-140	2.60+1	-	-	4.60+0	2.60+1	1.17-3	-	1.17-3	3.64+0	1.30-5	2.60+1	1.20-1	3.64+0	3.64+0	1.20-5
La-141	1.48-1	-	-	2.99-2	1.48-1	4.23-3	-	4.23-3	2.98-3	3.77-5	1.48-1	6.89-4	2.98-3	2.98-3	2.19-8
La-142	7.39-2	-	-	5.87-3	7.39-2	4.89-3	-	4.89-3	2.55-4	1.46-5	7.39-2	1.49-4	2.55-4	2.55-4	1.98-9
Ce-141	9.88-1	-	-	1.68-1	9.88-1	1.27-4	-	1.27-4	1.13-1	1.31-6	9.88-1	3.74-3	1.13-1	1.13-1	3.72-7
Ce-143	9.53-3	-	-	3.93-3	9.53-3	7.10-5	-	7.10-5	2.13-3	6.94-7	9.53-3	1.02-4	2.13-3	2.13-3	8.60-9
Ce-144	4.79-1	-	-	7.97-2	4.79-1	3.51-5	-	3.51-5	4.30-2	3.50-7	4.79-1	1.40-3	4.30-2	4.30-2	1.40-7
Pr-143	3.40-1	-	-	6.30-2	3.40-1	1.10-4	-	1.10-4	4.85-2	1.09-6	3.40-1	1.64-3	4.85-2	4.85-2	1.61-7
Pr-144	4.79-1	-	-	7.97-2	4.79-1	3.42-5	-	3.42-5	4.30-2	3.48-7	4.79-1	1.40-3	4.30-2	4.30-2	1.40-7
Nd-147	3.13-2	-	-	6.08-3	3.13-2	1.37-5	-	1.37-5	4.70-3	1.37-7	3.13-2	1.61-4	4.70-3	4.70-3	1.58-8
Pm-147	2.05-3	-	-	3.31-4	2.05-3	3.48-9	-	3.48-9	1.56-4	3.91-11	2.05-3	4.98-6	1.56-4	1.56-4	5.02-10
F-18	2.64-1	-	-	-	2.64-1	6.46-4	-	6.46-4	-	-	2.64-1	-	-	-	-
Na-24	1.79-1	5.67+1	2.50-1	7.72-2	1.79-1	2.87-3	8.34-1	2.87-3	2.51-2	2.73-5	1.79-1	1.78-3	2.51-2	2.51-2	1.24-7
P-32	2.37-1	1.53+0	7.37-3	4.37-2	2.37-1	7.69-5	2.65-2	7.69-5	3.34-2	7.66-7	2.37-1	1.13-3	3.34-2	3.34-2	1.11-7
Cr-51	1.34+1	4.56+1	2.19-1	2.29+0	1.34+1	2.29-3	7.92-1	2.29-3	1.58+0	2.28-5	1.34+1	5.24-2	1.58+0	1.58+0	5.20-6
Mn-54	5.53-1	8.00-1	3.84-3	9.20-2	5.53-1	4.01-5	1.39-2	4.01-5	4.94-2	4.00-7	5.53-1	1.61-3	4.94-2	4.94-2	1.61-7
Mn-56	3.65-1	3.36-1	6.33-1	7.67-5	3.65-1	1.10-2	1.58+0	1.10-2	3.27-3	1.35-7	3.65-1	1.14-3	3.27-3	3.27-3	2.51-8
Fe-55	5.76+0	7.80+0	3.75-2	9.60-1	5.76+0	3.91-4	1.36-1	3.91-4	5.04-1	3.90-6	5.76+0	1.64-2	5.04-1	5.04-1	1.64-6
Fe-59	6.46-1	1.59+0	7.66-3	1.09-1	6.46-1	7.98-5	2.77-2	7.98-5	6.95-2	7.96-7	6.46-1	2.29-3	6.95-2	6.95-2	2.28-7
Co-58	5.04+1	9.98+1	4.78-1	8.43+0	5.04+1	4.99-3	1.74+0	4.99-3	5.03+0	4.99-5	5.04+1	1.65-1	5.03+0	5.03+0	1.64-5
Co-60	7.50+0	1.00+1	4.79-2	1.25+0	7.50+0	5.01-4	1.74-1	5.01-4	6.55-1	5.00-6	7.50+0	2.13-2	6.55-1	6.55-1	2.13-6
Ni-65	2.16-3	1.24+0	3.70-3	2.80-4	2.16-3	6.48-5	9.11-3	6.48-5	1.88-5	4.98-7	2.16-3	6.64-6	1.88-5	1.88-5	1.44-10
Cu-64	4.47-1	1.57+2	6.77-1	1.72-1	4.47-1	7.88-3	2.22+0	7.88-3	5.29-2	7.51-5	4.47-1	4.17-3	5.29-2	5.29-2	2.75-7
Zn-65	1.05+0	1.56+0	7.50-3	1.75-1	1.05+0	7.81-5	2.72-2	7.81-5	9.51-2	7.80-7	1.05+0	3.10-3	9.51-2	9.51-2	3.10-7
Zn-69m	3.26-2	1.10+1	4.80-2	1.28-2	3.26-2	5.56-4	1.59-1	5.56-4	4.15-3	5.27-6	3.26-2	3.13-4	4.15-3	4.15-3	2.12-8
Zn-69	-	-	4.95-2	-	-	5.24-4	1.67-1	5.24-4	-	-	-	-	-	-	-
Ag-110m	8.10-1	1.20+0	5.76-3	1.35-1	8.10-1	6.01-5	2.09-2	6.01-5	7.33-2	6.00-7	8.10-1	2.39-3	7.33-2	7.33-2	2.39-7
Ag-110	1.62-2	2.40-2	7.50-5	2.70-3	1.62-2	7.81-7	2.72-4	7.81-7	1.47-3	1.20-8	1.62-2	4.78-5	1.47-3	1.47-3	4.78-9
W-187	2.24-1	4.73+1	2.15-1	9.54-2	2.24-1	2.38-3	7.40-1	2.38-3	4.45-2	2.31-5	2.24-1	2.44-3	4.45-2	4.45-2	1.93-7
Nb-98	3.57-2	1.90+1	3.01-2	1.59-3	3.57-2	1.10-3	5.95-2	1.10-3	3.81-5	5.30-6	3.57-2	4.03-5	3.81-5	3.81-5	2.95-10
Tc-104	2.64-1	-	-	4.31-3	2.64-1	8.62-3	-	8.62-3	1.40-5	1.74-5	2.64-1	4.20-5	1.40-5	1.40-5	1.09-10
Np-239	5.92+1	-	-	2.07+1	5.92+1	2.18-1	-	2.18-1	1.34+1	2.14-3	5.92+1	5.51-1	1.34+1	1.34+1	4.99-5
Pu-239	-	-	-	-	-	6.17-9	-	6.17-9	-	-	-	-	-	-	-
N-13	-	-	-	-	-	7.19-4	-	7.19-4	-	-	-	-	-	-	-

NOTE: 1.56-1 = 1.56×10^{-1}

TABLE 12.2-13

SOLID RADWASTE SYSTEM COMPONENTS
DESIGN ACTIVITIES
(uCi/cc)

Isotope	Waste Sludge Tank
Br-83	1.56-1
Br-84	4.22-2
Br-85	1.63-3
I-129	1.09-11
I-131	1.94+1
I-132	2.12+1
I-133	1.01+1
I-134	9.43-1
I-135	3.22+0
Rb-89	1.47-2
Sr-89	2.68+1
Sr-90	3.49+0
Sr-91	1.90+0
Sr-92	8.35-1
Y-90	3.43+0
Y-91m	1.15+0
Y-91	3.49+1
Y-92	8.41-1
Y-93	2.87-3
Y-94	1.01-9
Zr-95	-
Zr-97	1.58-3
Nb-95m	-
Nb-95	2.91-1
Nb-97m	1.51-3
Nb-97	1.58-3
Mo-99	6.93-2
Tc-99m	4.85+0
Tc-101	2.35-1
Ru-103	4.06-1
Ru-105	7.78-2
Ru-106	1.18-1
Rh-103m	4.06-1
Rh-105m	7.84-2
Rh-105	2.26-4
Rh-106	1.18-1
Te-129m	7.44-1
Te-129	7.44-1
Te-131m	2.80-2
Te-131	5.60-3
Te-132	1.95+1

TABLE 12.2-13 (Cont'd.)

Isotope	Waste Sludge Tank
Cs-134	1.32+0
Cs-136	6.19-3
Cs-137	3.64+0
Cs-138	6.82-5
Ba-137m	3.35+0
Ba-139	6.25-1
Ba-140	2.40+1
Ba-141	1.38-1
Ba-142	7.73-2
La-140	2.60+1
La-141	1.48-1
La-142	7.39-2
Ce-141	9.88-1
Ce-143	9.53-3
Ce-144	4.79-1
Pr-143	3.40-1
Pr-144	4.79-1
Nd-147	3.13-2
Pm-147	2.05-3
F-18	2.64-1
Na-24	1.79-1
P-32	2.37-1
Cr-51	1.34+1
Mn-54	5.53-1
Mn-56	3.65-1
Fe-55	5.76+0
Fe-59	6.46-1
Co-58	5.04+1
Co-60	7.50+0
Ni-65	2.16-3
Cu-64	4.47-1
Zn-65	1.05+0
Zn-69m	3.26-2
Ag-110m	8.10-1
Ag-110	1.62-2
W-187	2.24-1
Nb-98	3.57-2
Tc-104	2.64-1
Np-239	5.92+1
Nb-98	3.57-2
Tc-104	2.64-1
Np-239	5.92+1

NOTE: 1.56-1 = 1.56×10^{-1}

TABLE 12.2-14

THIS TABLE HAS BEEN DELETED.

TABLE 12.2-15a

EXPECTED AIRBORNE RADIOACTIVITY CONCENTRATIONS
AT IN-PLANT AREAS DURING NORMAL POWER OPERATION
(uCi/cc)

		Radwaste Building			Turbine Building			
					Miscellaneous Areas			
Isotope	Reactor Building	Floor Drain System	Waste Collector System	Solid Waste System	Main Condenser	Total	Steam Jet Air Ejector	Mechanical* Vacuum Pump
I-131	8.53E-11	3.49E-11	7.73E-11	1.07E-11	5.19E-10	7.13E-11		
I-132	6.87E-10	2.79E-10	6.18E-10	8.59E-11	4.23E-09	5.81E-10		
I-133	5.62E-10	2.33E-10	5.15E-10	7.16E-11	3.46E-09	4.75E-10		
I-134	1.12E-09	4.65E-10	1.03E-09	1.43E-10	7.30E-09	1.00E-09		
I-135	7.70E-10	3.26E-10	7.23E-10	1.00E-10	4.80E-09	6.60E-10		
Kr-85M	6.24-09				3.96-08		2.76-07	3.73-07
Kr-87	4.37-09				7.45-08		1.20-06	6.50-07
Kr-88	7.91-09				1.56-07		6.98-07	1.55-06
Kr-89	4.99-09			2.08E-07	8.00-07		1.03-05	2.47-06
Xe-133	2.29-07			1.57E-06	2.17-07		2.16-06	1.92-06
Xe-135M	1.27-07			3.79E-06	4.88-07		6.33-06	1.11-05
Xe-135	2.71-07			2.00E-06	5.15-07		2.53-06	1.06-05
Xe-137	3.95-07			5.94E-07	1.58-06		1.44-05	0
Xe-138	1.52-08			1.43E-08	1.20-06		2.21-05	2.13-06
Cr-51	2.00-12	1.31E-14	2.88E-14	4.01E-15	1.38-12	1.90-13		
Mn-54	2.71-12	7.45E-14	1.65E-13	2.29E-14	9.22-13	1.27-13		
Fe-59	7.28-13	5.58E-15	1.24E-14	1.72E-15	1.54-13	2.11-14		
Co-58	5.83-13	3.72E-15	8.23E-15	1.15E-15	1.54-12	2.11-13		
Co-60	1.04-11	1.31E-13	2.88E-13	4.01E-14	1.54-12	2.11-13		
Zn-65	8.32-12	5.58E-15	1.24E-14	1.72E-15	9.22-12	1.27-12		
Sr-89	3.54-13				9.22-12	1.27-12		
Sr-90	1.58-14				3.07-14	4.22-15		
Zr-95	1.27-12	1.49E-14	3.30E-14	4.58E-15	6.15-14	8.45-15		
Nb-95	2.29-11	7.45E-17	1.65E-16	2.29E-17	9.22-15	1.27-15		
Mo-99	1.31-10	5.58E-17	1.24E-16	1.72E-17	3.07-12	4.22-13		
Ru-103	1.21-12	1.86E-17	4.12E-17	5.73E-18	7.69-14	1.06-14		
Ag-110M	5.20-15							
Sb-124	4.16-14	1.31E-15	2.88E-15	4.01E-16	1.54-13	2.11-14		
Cs-134	8.95-12	4.42E-14	9.81E-14	1.36E-14	3.07-13	4.22-14		
Cs-136	9.99-13				1.54-13	2.11-14		
Cs-137	1.17-11	7.45E-14	1.65E-13	2.29E-14	1.54-12	2.11-13		

TABLE 12.2-15a (Cont'd.)

		Radwaste Building			Turbine Building			
					Miscellaneous Areas			
Isotope	Reactor Building	Floor Drain System	Waste Collector System	Solid Waste System	Main Condenser	Total	Steam Jet Air Ejector	Mechanical* Vacuum Pump
Ba-140	2.91-11	7.45E-17	1.65E-16	2.29E-17	1.54-11	2.11-12		
Ce-141	1.62-12	1.31E-16	2.88E-16	4.01E-17	1.54-11	2.11-12		
H-3	6.45E-08				5.96E-08	8.18E-09		

* The mechanical vacuum pump is used to purge the main condenser and to maintain condenser vacuum only during the transition between normal power operation and reactor shutdown, as well as from shutdown to power operation. Noble gases are only generated when the reactor's core is undergoing fission.

TABLE 12.2-15b

EXPECTED AIRBORNE RADIOACTIVITY CONCENTRATIONS
AT IN-PLANT AREAS DURING REACTOR SHUTDOWN
(uCi/cc)

Isotope	Reactor Building	Radwaste Building: Floor Drain System	Radwaste Building: Waste Collector System	Radwaste Building: Solid Waste System	Turbine Building: Main Condenser	Turbine Building: Miscellaneous Areas
I-131	3.54E-11	1.07E-11	2.37E-11	2.71E-12	3.28E-11	2.55E-11
I-132	2.91E-10	8.84E-11	1.96E-10	2.24E-11	2.71E-10	2.11E-10
I-133	2.29E-10	7.21E-11	1.60E-10	1.82E-11	2.15E-10	1.67E-10
I-134	4.79E-10	1.45E-10	3.19E-10	3.65E-11	4.63E-10	3.61E-10
I-135	3.33E-10	9.77E-11	2.16E-10	2.47E-11	3.05E-10	2.38E-10
Cr-51	2.71-13	3.26E-15	7.23E-15	8.24E-16	4.07E-14	3.17E-14
Mn-54	5.41-13	1.86E-14	4.12E-14	4.71E-15	2.71E-14	2.11E-14
Fe-59	7.91-14	1.40E-15	3.09E-15	3.53E-16	4.52E-15	3.52E-15
Co-58	1.71-13	9.31E-16	2.06E-15	2.35E-16	4.52E-14	3.52E-14
Co-60	1.58-12	3.26E-14	7.23E-14	8.24E-15	4.52E-14	3.52E-14
Zn-65	2.29-12	1.40E-15	3.09E-15	3.53E-16	2.71E-13	2.11E-13
Sr-89	2.71-14				2.71E-13	2.11E-13
Sr-90	4.79-15				9.04E-16	7.04E-16
Zr-95	2.08-13	3.72E-15	8.23E-15	9.42E-16	1.81E-15	1.41E-15
Nb-95	4.16-13	1.86E-17	4.12E-17	4.71E-18	2.71E-16	2.11E-16
Mo-99	7.70-14	1.40E-17	3.09E-17	3.53E-18	9.04E-14	7.04E-14
Ru-103	2.50-13	4.65E-18	1.03E-17	1.18E-18	2.26E-15	1.76E-15
Ag-110M	1.44-15					
Sb-124	2.29-14	3.26E-16	7.23E-16	8.24E-17	4.52E-15	3.52E-15
Cs-134	7.07-13	1.12E-14	2.48E-14	2.83E-15	9.04E-15	7.04E-15
Cs-136	1.44-13				4.52E-15	3.52E-15
Cs-137	1.25-12	1.86E-14	4.12E-14	4.71E-15	4.52E-14	3.52E-14
Ba-140	5.62-13	1.86E-17	4.12E-17	4.71E-18	4.52E-13	3.52E-13
Ce-141	1.71-13	3.26E-17	7.23E-17	8.24E-18	4.52E-13	3.52E-13

TABLE 12.2-16

CRUD RADIATION LEVELS

Component/System	Additional Contact Dose Rate Due to Crud (mrem/hr)
RCIC piping/pump	60
RHR piping/heat exchanger	600
CRD	
discharge header	600
pumps/filter	50
RWCU	
piping/pumps	2,000
heat exchanger	2,500
Reactor building equipment drain*	
tank	6,000
cooler	3,000
pump	2,000
Drywell equipment drain*	
tank	30,000
cooler	15,000
pump	10,000

* The analyses performed for these systems are based entirely on the crud levels quoted.

12.3 RADIATION PROTECTION DESIGN FEATURES

Radiation protection design features are provided to reduce direct radiation, control airborne radioactivity, identify and control access to radiation areas, minimize personnel and equipment contamination, and detect area and airborne radiation levels in order to maintain personnel radiation exposure ALARA.

12.3.1 Facility Design Features

The design objectives and the design feature guidance given in RG 8.8 are incorporated into the Unit 2 plant to the extent discussed in Section 12.1.2. The general design and requirements of the Unit 2 access control system are covered in Section 12.3.1.2.

Items aiding plant radiation protection included in Unit 2 or provided by Nine Mile Point Nuclear Station - Unit 1 (Unit 1) include: 1) classification of all areas of the Station into radiation zones, 2) adequate laboratory facilities, 3) decontamination capabilities, and 4) adequate laundry facilities.

12.3.1.1 Plant Design Description

The equipment and plant design features employed to maintain radiation exposures ALARA are outlined in this section for several general classes of equipment (Section 12.3.1.1.1) and several typical layout situations (Section 12.3.1.1.2).

12.3.1.1.1 Common Equipment and Component Designs

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

Filters

Filters that accumulate radioactivity carried in liquids are supplied with the means to remotely backflush the filter. Filters not having remote backflush capability (e.g., manual backflush, replaceable or cleanable filters) are cleaned and replaced in accordance with ALARA practices to minimize personnel exposures.

Demineralizers

Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to the radwaste system for solidification. Fresh resin can be remotely loaded into the demineralizer. Underdrains and downstream strainers are designed for full system pressure drop. The

demineralizers and piping are designed with provisions to flush with condensate. Depending on the use received, resin can be reused if the collected crud is removed.

This can be achieved by ultrasonic resin cleaning, which removes the main radioactive source, the crud. This is performed as required in the CNDs.

Evaporators

Evaporators are provided with chemical addition connections through normal process piping for descaling operations. Space is provided to allow efficient removal of heating tube bundles.

Pumps

Where practicable, pumps are purchased with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for maintenance. Small pumps are installed in a manner that allows easy removal, if necessary. All pumps in radioactive waste systems have flanged connections for ease in removal. Pump casings have drain connections for draining the pump for maintenance.

Tanks

Whenever practicable, tanks are provided with sloped or conical bottoms and bottom outlet connections. Overflow lines are directed to the floor and equipment drain system in order to control any contamination within plant structures. Provisions such as compartmentation of tanks and compartment floor drains are made to contain overflows and accidental spills.

Instruments

Instrument devices are located in low radiation zones and away from radiation sources whenever practical. Primary instrument devices, which for functional reasons are located in high radiation zones, are designed for easy removal to a lower radiation zone for calibration, if possible. Transmitters and readout devices, whenever practical, are located for servicing in low radiation zones, such as corridors and the main control room. Some instruments (such as thermocouples) in high radiation zones are provided in duplicate to reduce the access and service time required.

Instrument sensing lines on process piping that may contain highly radioactive liquids and solids have diaphragm seals with backflushing capability to reduce the servicing time required to keep the lines free of solids. The diaphragm seal connections are arranged to permit remote calibration of instruments whenever possible. Instrument and sensing line connections are located so as to avoid corrosion product and radioactive gas buildup.

Compression-type fittings are used between instruments and their sensing lines to allow for easy removal.

Valves

To minimize personnel exposures from valve operations, motor-operated or other remotely-actuated valves are used to the maximum extent practicable. Solenoids for air-operated valves (AOVs) in radwaste systems are located on remote racks in low radiation zones.

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

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

All manually-operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are remotely operated. Personnel do not enter the valve gallery during flushing operations. The valve gallery shield walls are designed for maximum expected filter backflush activities during flushing operations.

Most large valves in the radwaste system are of the packless design allowing no leakage around the valve stem. Other large valves in lines carrying radioactive liquid have packing designs which minimize stem leakage. Full-ported valves are used in systems expected to contain radioactive solids.

Valve designs with minimum internal crevices are used where crud trapping could become a problem, especially for piping carrying spent resin or evaporator bottoms. Valves are flanged when necessary in radwaste systems for ease of removal from piping systems in high radiation areas.

Piping

Piping containing radioactive material is routed to minimize radiation exposure to personnel. Generally, there are no valves or instrumentation located in the pipe chase. Wherever radioactive piping is routed through areas where routine maintenance is required, shielding or pipe chases are provided to reduce the radiation contribution from these pipes to levels appropriate for the inspection and/or maintenance requirements. Radioactive fluid piping is not routed through radioactive equipment areas of unrelated systems, or is routed through shielded pipe chases in these areas, to minimize exposure during

maintenance by segregation of systems. Piping carrying radioactive solids is designed with five-diameter radius bends and crevice-free construction to minimize plugging and crud deposition.

Floor Drains

Floor drains and properly sloped floors are provided in each room or cubicle containing serviceable components containing radioactive liquids. Local gas traps or porous seals are not used on radwaste floor drains.

Lighting

Multiple electric lights are used to provide a sufficient margin of illumination for each cell or room containing highly radioactive components so that entry is not required each time a single lamp fails. The fluorescent lights used in most areas of the plant do not require frequent service due to the increased life of the tubes.

Heating, Ventilating, and Air Conditioning System

The HVAC system is designed so that filter elements may be quickly replaced.

Sample Stations

Sample stations for routine sampling of process fluids are located in accessible areas. Shielding is provided at the local sample stations as required to maintain radiation levels ALARA. The counting room and laboratory facilities are described in Section 12.5.

Clean Services

Whenever practicable, clean services and equipment such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipe chases.

Design Features That Reduce Cobalt/Nickel Buildup

Design features and considerations are included to reduce radioactive nickel and cobalt production and buildup. For example, the primary coolant system consists mainly of austenitic stainless steel, carbon steel, and low-alloy steel components. The nickel content of these materials is low. Nickel and cobalt contents are controlled in accordance with applicable ASME material specifications. A small amount of nickel base material (Inconel 600) is employed in the reactor vessel internal components. Inconel 600 is required where components are attached to the reactor vessel shell, and the coefficient of expansion must match the thermal expansion characteristics of the low-alloy vessel steel. Inconel 600 was selected because it

provides the proper thermal expansion characteristics and adequate corrosion resistance, and can be readily fabricated and welded. Hardfacing and wear materials having a high percentage of cobalt are restricted to applications where no alternative materials with equally good characteristics are available. Also, components where radio-cobalt buildup is probable are designed to accommodate decontamination.

12.3.1.1.2 Common Facility and Layout Designs

This section describes the design features utilized for Unit 2 processes and layout arrangements. These features are employed in conjunction with the general equipment designs described in Section 12.3.1.1.1 and are discussed in the following paragraphs.

Piping

Pipes carrying radioactive materials are routed through controlled access areas properly zoned for that level of activity. Each piping run is individually analyzed to determine the potential radioactivity level and surface dose rate. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, appropriate shielding is provided in accordance with the ALARA philosophy. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Whenever practicable, equipment compartments are used as pipeways only for those pipes associated with equipment in that compartment.

Radioactive and nonradioactive piping are separated when practicable to minimize personnel exposure. Should maintenance be required, provisions are made to isolate and drain the radioactive piping and associated equipment. Process piping is monitored to ensure that access is controlled to limit exposure (Section 12.5).

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Thermal expansion loops are raised rather than dropped, where possible. In radioactive systems, the use of nonremovable backing rings and socket welds in the pipe weld joints is minimized to eliminate potential crud traps for radioactive materials. Piping for systems carrying radioactive solids is run downward from the point of origin whenever possible, and large radius bends are used instead of elbows.

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

Penetrations

To minimize radiation streaming through penetrations, as many penetrations as practicable are located with an offset between

the source and the accessible areas or provided with appropriate shielding. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. In addition, the annular clearance between pipe and penetration sleeve is kept as small as possible consistent with construction requirements. Radiation surveys during preliminary operation will be made to determine if supplementary shielding, i.e., collars, sleeve packing, etc., are required at any shield penetration in order to remain below design dose rates in the given area and maintain exposures ALARA.

Contamination Control

Access control and traffic patterns are considered in basic plant layout to minimize the spread of contamination. Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to floor drains. All-welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at joints.

To facilitate decontamination, smooth surface coatings are applied to those areas that have potential for high levels of contamination. When assigning decontamination priorities, areas are evaluated based on traffic of personnel in the area, the accessibility of the area, the radiation levels in the area, the impact of the area contamination on the facility environment, and whether the ALARA principle is served by cleaning the area and keeping it cleaned.

Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant.

For dry, contaminated areas, other contamination control techniques are used such as step-off pads and roped and posted areas. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

Systems that become highly radioactive, such as radwaste slurry piping systems, are provided with flush and drain connections. Certain systems have provisions for chemical and mechanical cleaning prior to maintenance.

Concrete shield wall thicknesses are dimensioned for the turbine, reactor, and radwaste buildings on Figures 12.3-1 through 12.3-33. These figures also show controlled access areas, personnel and equipment decontamination areas, contamination control areas, location of the health physics facilities, location of area radiation monitors, location of control panels for radwaste equipment and components, and location of the counting room.

Personnel decontamination facilities are supplied for prompt decontamination of plant personnel. Features include showers,

drying areas, and monitoring instruments, while emergency equipment (used for chemical as well as radioactive decontamination) includes showers, eye wash stations, and sinks. These are installed at appropriate locations throughout the Station. The locations of the personnel decontamination rooms are shown on Figures 12.3-1 to 12.3-33. The laundry facilities located in Unit 1 are shared by Unit 2 for the decontamination of radiation protection apparel and breathing apparatus. Laundry services may also be provided by outside vendors.

Equipment Layout

In those systems where process equipment is a major radiation source (e.g., fuel pool cleanup filter/demineralizers, radwaste and CNDs), pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of these items in reduced radiation zones. Control panels are located in low radiation areas.

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

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

Labyrinth entranceway shields or shielding doors are provided for each compartment from which radiation could stream to access areas and exceed the radiation zone dose limits for those areas.

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

Design Features That Reduce Occupational Doses During Decommissioning

Many of the design facilities that presently exist in the Unit 2 plant can be used to minimize occupational exposure during decommissioning, whether decommissioning is accomplished through mothballing, entombment, removal/dismantling, or any combination of these alternatives. Such facilities include those used for

handling and for offsite shipment of fresh fuel, spent fuel, contaminated filter elements and resins, and other radioactive wastes. The radioactively contaminated spent fuel pool water can be removed after all spent fuel has been removed from the site and any decommissioning use of the pool is finished. The fuel pool cooling and cleanup system can decontaminate the fuel pool water.

The number of man-Rems due to airborne radioactivity that may be introduced by the handling of radioactively contaminated systems, as well as the number of man-Rems due to direct contact with the same systems, can be reduced by first decontaminating them. Means exist in the present design where radioactively contaminated systems can be decontaminated chemically and flushed.

An ISFSI is used for interim storage of NMPNS Unit 1 and Unit 2 spent nuclear fuel assemblies utilizing the Transnuclear, Inc., Standardized NUHOMS® modular storage system for irradiated nuclear fuel with NUHOMS®-61BT and NUHOMS®-61BTH Dry Shielded Canisters (DSCs). Each DSC can contain 61 spent fuel assemblies and each HSM contains only on DSC.

Decommissioning of NMPNS ISFSI will be performed in a manner consistent with that for decommissioning of the plant itself. It is anticipated that the DSCs will be transported intact to a Federal repository offsite when such a facility is operational. However, should the storage facility not accept the DSCs intact, the NUHOMS® system allows the DSCs to be brought back into the spent fuel pool and the fuel off-loaded to racks for subsequent loading in to transport casks provided by the Department of Energy.

All components of the NUHOMS® system are manufactured of materials similar to those found at the NMPNS (e.g., reinforced concrete, stainless steel, lead). These components will be decommissioned by the same methods in place to handle those materials within the plant. Any of the components that may be contaminated will be cleaned and/or disposed of using the decommissioning technology available at the time of decommissioning.

The NUHOMS® system is a dry containment system that effectively confines all contamination within the DSC. When the DSC is removed from the HSM, the freestanding HSM can be manually decontaminated for any trace activity, dismantled and removed from the site. It is possible that a thin layer of material comprising the inner wall of the HSM could become activated by the neutron flux from the fuel after an extended period of service. Estimates of the potential for activation are difficult due to the variability of rare earths which may be present in the local aggregate. The specific activity of the HSM inner wall surfaces may be measured at the time of decommissioning and compared with the existing guidelines to determine whether the

values are below regulatory concern. Disposal procedures will be developed which comply with existing guidelines at the time of decommissioning.

Removal of fuel assemblies from the DSC can be accomplished in the plant's spent fuel pool. The DSC is also being qualified for offsite shipment in a compatible transportation cask licensed to 10 CFR 71. If such transport is made, the DSC may be disposed of as-is at the permanent geologic repository in a suitable overpack container. If the DSC is not compatible with the repository handling or packaging systems, fuel transfer to a suitable container can be performed in a large hot cell or offsite fuel pool.

12.3.1.2 Radiation Zoning and Access Control

The following terms are used to designate the various areas within the site boundary:

<u>Controlled Area</u> As defined in 10CFR20, "means an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason." At Unit 2, the Controlled Area includes the area between the site boundary and the security protected area fence.

<u>Restricted Area</u> As defined in 10CFR20, "means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials..." At Unit 2, the Restricted Area includes the area within the security protected area fence.

<u>Radiologically Controlled Area (RCA)</u> As defined at Unit 2, means major plant areas, access to which is limited for the purpose of protecting personnel from exposure to radiation and contamination. The RCA includes the reactor, turbine, radwaste, offgas buildings and auxiliary services buildings.

Other RCAs may be established with protective requirements specified by radiation protection supervision in accordance with approved Station procedures.

Access to areas inside the plant structures and yards is regulated by radiation zoning and access control. Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding.

All areas within the unit are identified by different radiation zones in accordance with the expected occupancy as listed in the following table:

Numerical Designation	Zone Designation and Expected Maximum Occupancy (hr/week)	Design Radiation Dose Rate (mRem/hr)
I	High Radiation Area (not normally accessible)*	$\geq$100
II	Radiation Area - Infrequent Access (<1)*	<100
III	Radiation Area - Periodic Access (1-5)*	<20
IV	Radiologically-Controlled Area - Controlled Access (20)*	<5
V	Restricted Area - Occupational Access (50)	<2**
VI	Restricted Area - Continuous Access (>50)	<0.2
IV/I	Zone Dependent on Operational Mode - Radiologically-Controlled Area - Controlled Access (20)* during normal operation/High Radiation Area (not normally accessible)* during specific operations	<5/$\geq$100

High Radiation Areas are posted. Access is controlled by doors or gates and is permitted by RWP. Areas above 1,000 mRem/hr shall be provided with a locked door, gate, or guard to prevent unauthorized entry, and the keyed access shall be maintained under the administrative control of the Shift Manager (SM) or the designee on duty, and/or the General Supervisor Radiation Protection or designee. During the time when access is permitted into these areas, there is positive control over each individual entry.

---

* Personnel monitoring equipment required to be worn when required by 10CFR20 or Station procedures.

** The dose rate of 1.5 mRem/hr has been used as a design value to ensure that the 2-mRem/hr requirement has been met. Two mRem/hr used as the operational limit.

Areas denoted Radiation Area-Infrequent Access are posted as Radiation Areas. Access to these areas is permitted in accordance with established radiation protection procedures at the Station.

Areas where the design dose rate is in excess of 5 mRem/hr but less than 100 mRem/hr are posted as Radiation Areas. Personnel exposures in radiation areas are kept to a minimum by use of administrative procedures based on accumulated doses and by keeping time spent in radiation areas as short as possible.

Areas where design dose rates are less than 5 mRem/hr but greater than 2 mRem/hr are denoted as Radiologically-Controlled Areas. Access to these areas is controlled administratively through the use of conspicuously posted signs. Radiation exposure is measured for all personnel entering a RCA or higher level radiation area when required by 10CFR20 or Station procedures.

Restricted-Occupational Access Areas within the Station are available for use by personnel on a limited occupancy basis, consistent with security requirements. This occupancy is controlled administratively. Access to these areas from a RCA is through a contamination control point where personnel and material are monitored for contamination in accordance with approved Station procedures. Any other access is through Restricted-Continuous Access Areas or the Controlled Area.

Restricted-Continuous Access Areas within the protected area are available for use by personnel on an essentially unlimited basis consistent with security requirements. Access to these areas from areas outside the protected area is through locked doors and gates. Only authorized personnel are allowed entry to the protected area. Access to the Restricted-Continuous Access Areas from RCAs in the Station is through a contamination control point where personnel and material are monitored for contamination in accordance with approved Station procedures.

Areas where the zone designation is Zone Dependent on Operational Mode are RCAs during normal operation/High Radiation Areas during specific operations. Access controls and expected maximum occupancy for each mode of operation are as described above for RCAs and High Radiation Areas.

Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal operation and shutdown, for maintenance occupancy requirements, for general access requirements, and for material exposure limits to determine appropriate zoning.

Specific zoning for Zone 1 plant areas is shown on Figures 12.3-34 through 12.3-66. All frequently occupied areas are shielded for less than 5 mRem/hr (Zone IV). Whenever practicable, the measured radiation level and the location of the

source are posted at the entry to any Radiation or High Radiation Area.

Access control to the Unit 2 protected area is controlled by security personnel by a computerized cardreader system. In order to be permitted unescorted access within the Station buildings housing radioactive materials, a person must be qualified in radiation protection. Access to RCAs is controlled administratively through the use of conspicuously posted signs. Radiation area access is permitted in accordance with established radiation protection procedures. The SM or designee on duty, and/or the General Supervisor Radiation Protection or designee, maintain positive access control over High Radiation Areas (>1000 mRem/hr) at all times. All doors controlling ingress to areas by means of a cardreader/hardkeys are equipped such that exit from Radiation Areas, High Radiation Areas, and RCAs is not prohibited. Magnetic switches provided on all cardreader-controlled doors will cause an alarm if the doors are being held or propped open during system operation.

Magnetic switches are also provided to monitor certain controlled doors without cardreaders that are not normally accessed. In this case, there will be an alarm whenever the door is opened during system operation. When the system is inoperative all doors controlled by electric locks fail to the locked position, and the alarm function is defeated. During this period, keys and procedures for alternate methods of access control are provided to backup this system.

Procedures are employed to assure that radiation levels and allowable working time are within the limits prescribed by 10CFR20. These procedures, as well as the control of ingress or egress of plant operating personnel to controlled access areas, are described in Section 12.5.

12.3.1.3 Post-accident Access and Shield Design Review

A post-accident access and shield design review was performed in accordance with NUREG-0737, Item II.B.2, to ensure personnel accessibility to vital areas following a design basis accident (DBA). The DBA considered in this analysis is the loss-of-coolant accident (LOCA). The source terms used are those specified in RG 1.3 and discussed in Section 15.6.5.5.2.

The plant is designed so that access after an accident is essential in only a limited number of areas. All Unit 2 post-accident vital access areas are listed as follows:

1. Main control room - control building, el 306 ft

2. Relay and computer room - control building, el 288 ft 6 in

3. Health physics/counting room - Unit 1 turbine building, el 261 ft

4. Radwaste sample room (post-accident sampling) - turbine building, el 261 ft

5. Off-line gaseous effluent monitors - turbine building, el 306 ft and main stack, el 261 ft

6. Radwaste control room - turbine building, el 279 ft

7. Technical support center (TSC) - Unit 1 administration building

8. Chemistry laboratory - Unit 1 turbine building, el 261 ft

9. Associated connecting access paths

Other post-accident vital access areas suggested by NUREG-0737 either do not apply to Unit 2, or access to them is not required at Unit 2.

The doses received by individuals working in or traveling between the various vital areas in performing necessary tasks are presented in Table 12.3-3. The tasks to be performed in the area, the occupancy times in the area including travel time to and from the area, and the doses received in performance of each task are presented for each vital area. The following radiation sources contribute to the doses received for each task:

1. Direct shine from secondary containment

2. Airborne releases (described in Section 15.6.5.5.3)

3. Air-scattered radiation from secondary containment (sky shine)

Additional dose contributions from localized sources (e.g., post-accident samples) are accounted for on a case-specific basis.

Dose rates as a function of time at various areas requiring possible occupancy following an accident are presented in Table 12.3-4. Post-accident pathways to vital areas are presented on Figure 12.3-69.

The calculated doses received in performing vital post-accident functions were determined based on the following:

1. Unless otherwise specified, tasks are assumed to be performed at the time post-accident at which the highest dose rates occur in order to provide a maximum possible dose for the task.

2. Allowable dose limits are based on 10CFR50 Appendix A, General Design Criterion (GDC) 19, as specified by NUREG-0737.

3. Personnel transit times are based on:

   a. A constant walking speed of 3 ft/sec, or

   b. A constant driving speed of 15 miles/hr (22 ft/sec)

4. Areas requiring continuous occupancy are analyzed to ensure that the 30-day average dose rates are less than 15 mRem/hr, specified by NUREG-0737.

5. The source terms used to calculate the dose contribution due to the samples during operation of the post-accident sampling system (PASS) are as follows:

Source	Source Term (% of core inventory)
Pressurized reactor coolant	100 noble gases 50 halogens 1 cesium 1 remaining isotopes
Depressurized reactor coolant	0 noble gases 50 halogens 1 cesium 1 remaining isotopes
Containment atmosphere	100 noble gases 25 halogens

6. Other than the main control room and the TSC, no vital area requires access within the first hour after the accident.

7. The starting and ending point for all post-accident activities is the Operational Support Center (OSC) located in the Unit 1 administration building.

Descriptions of the post-accident vital areas and tasks to be performed are provided as follows. Area numbers correspond with those provided above.

1&2. Main Control Room/Relay and Computer Room - Together, these two areas make up the control room emergency zone. Continuous occupancy for 30 days is required to execute safe shutdown of the plant. Shielding and ventilation system designs ensure habitability for 30

days within the dose limits of GDC 19. See Section 6.4 for details of this habitability analysis.

3. Health Physics/Counting Room - Intermittent occupancy is required to perform routine health physics functions. Since a specific stay time in this area is not defined, the maximum dose is calculated based on full-time occupancy for a standard 8-hr workday.

4&8. Radwaste Sample Room/Unit 1 Chemistry Lab - Intermittent occupancy is required to obtain, transport, and analyze post-accident samples. The samples are assumed to be taken at t=1 hr post-LOCA. See Section 1.10, Item II.B.3, for details of the sampling and analysis procedure.

5a. Deleted.

5b. Main Stack Off-line Monitor - Due to the increased radioactivity concentration in the stack effluent after an accident, access could be required as frequently as every 6 hr throughout the accident to retrieve iodine and particulate sample cartridges. It is assumed that the person servicing the stack monitor will drive from the administration building to the stack to perform these functions. Assuming both tasks must be performed during the same trip, the stay time is 15 min plus 5 min for the cartridge retrieval, for a total of 20 min.

Access is required to manually obtain samples at the main stack in the event that the main stack off-line monitor becomes inoperative or does not function properly. Sample taps are provided in the inlet and outlet monitor sample lines to manually obtain particulate and air samples. It is assumed that the individual obtaining the samples will walk from the administration building to the stack to perform this function. The time for a round trip to the main stack is 14 min. The stay time at the main stack (including 5 min for a radiation survey) is 17 min. The total time for analyzing the particulate and air samples is 22 min (11 min for each sample). The total time required to complete this task is 53 min. See Section 1.10, Item II.B.3, for details of the sampling and analysis procedure.

6a. Radwaste Control Room - One-time access is required to turn off reactor building equipment and floor drain pumps in order to prevent the discharge of post-LOCA fluids to the radwaste building. Although this task will probably be performed early in the accident, the dose is calculated using worst-case dose rates to provide a conservative dose. The stay time for this task is assumed to be 5 min.

6b. Access is also required at t=1 hr and again at t=12 hr post-LOCA to service the Emergency Response Facility (ERF) computer system. Again, one dose is calculated using worst-case dose rates to provide a conservative dose. The stay time for each task is 15 min.

7. Technical Support Center - Continuous occupancy for 30 days is required to:

   a. Provide plant management and technical support to plant operations personnel during emergency conditions.

   b. Relieve the Reactor Operators of peripheral duties and communications not directly related to reactor system manipulations.

   c. Prevent congestion in the control room.

   d. Perform Emergency Operations Facility (EOF) functions for the alert emergency class, the site emergency class, and the general emergency class events until the EOF is functional.

9. Associated Connected Access Paths - All pathways used to perform vital post-accident functions are shown on Figure 12.3-69. Calculated doses, except for those continuously occupied areas, include the dose received for a round trip between the OSC and the vital area based on the average dose rate for the path at the appropriate time post-LOCA.

12.3.2 Shielding

12.3.2.1 Design Objectives

12.3.2.1.1 Radiation Exposure of Individuals

The major objective of the shielding is to protect the operating personnel and the general public from radiation emanating from the reactor, power conversion process, and auxiliary systems, including equipment and piping. Radiation shielding and other unit features are designed to satisfy the dose criteria specified in the following documents:

Station Condition	Regulatory Document	Group Affected
Normal operation	10CFR20	Unit operating personnel and general public
Normal operation	10CFR50,	General public

Station Condition	Regulatory Document	Group Affected
	Appendix I, 40CFR190	
Accident conditions	10CFR50, Appendix A, GDC 19	Unit control room personnel and general public
Accident conditions	10CFR100	General public

All areas of the unit are subject to 10CFR20 regulations and are zoned according to their expected occupancy by Station personnel and the radiation exposure level anticipated under normal operating conditions. Refer to the zone maps, Figures 12.3-34 through 12.3-66, for Zone 1 designated areas.

12.3.2.1.2 Radiation Exposure of Materials and Components

A secondary shielding objective is the protection of materials and equipment from excessive radiation. Further information is found in Section 3.11.5.

12.3.2.2 Design Description

12.3.2.2.1 General Design Guides

To meet the design objectives, the following general design guides are used in the shielding analysis of the plant.

All systems containing radioactivity are identified and shielded, based on the access requirements of adjacent areas. The radiation zone designation for each area and the amounts of shielding needed to ensure these zones are calculated. Radiation zones are shown on Figures 12.3-34 through 12.3-66.

An effort is made to locate process systems in such a manner as to minimize shielding. Use of labyrinths is made to eliminate any streaming radiation from equipment. Segregation of radioactive equipment is provided, wherever practicable, in such a way that shielding reduces the dose rate from adjacent cubicles to less than 20 mRem/hr.

When possible, penetrations are placed so that they do not pass through the shield wall in a direct line with the radiation source in order to prevent streaming at locations hazardous to personnel. If this is not feasible, adequate shielding is provided.

12.3.2.2.2 Shielding Materials

The primary shielding material used to meet shielding design criteria is concrete with a density of 135 lb/cu ft. The

majority of the concrete shields are poured in place, and the remainder are made up of mortared solid block. Lead, steel, heavy aggregate concrete, and water are used in certain applications, e.g., the reactor shield, wall plugs, the fuel pools, and temporary shielding. Where access must be provided for periodic inspection and maintenance, removable concrete, lead, or steel shields (e.g., plugs, block walls) are used.

12.3.2.2.3 Plant Shielding Description

Plant building layouts, which provide locations of equipment containing radioactive fluids and indicate relative shield wall thickness, radiation zone designations, access control, and radiation monitor locations, are shown on Figures 12.3-1 through 12.3-33.

The general description of plant shielding in the different plant buildings is as follows.

Reactor Building

Shielding for the reactor building includes the biological shield, drywell (primary containment), and reactor building walls. The BSW is a double cylindrical steel wall connected by internal horizontal and vertical stiffeners and filled with heavy-density fill material which surrounds the reactor vessel. The BSW is used to reduce gamma and neutron flux outside the vessel. It reduces radiation heating in the drywell concrete wall; reduces activation of, and radiation effects on, materials and equipment in the drywell; and provides limited access to the annular region for periodic shutdown inspection and maintenance. The drywell is Zone I during normal operation.

The drywell wall provides additional shielding of neutrons and gamma rays in order to permit access to the reactor building during normal operation. The reactor building radiation level in most areas outside the drywell is less than 5 mRem/hr.

Within the reactor building, there are several shielded rooms. The rooms enclose RWCU, RHR, TIP, RCIC, SFC, and other miscellaneous equipment and piping. In addition, the main steam lines are within a shielded tunnel.

The reactor building wall is a 2 1/2-ft thick minimum, reinforced concrete structure that completely surrounds the nuclear steam supply system (NSSS). This wall attenuates the system radiation to ensure that levels outside the building are less than 0.2 mRem/hr (Zone VI). In addition, in the unlikely event of an accident, the reactor building wall shields personnel and the public from radiation sources inside the containment. Details on the construction of this building are presented in Section 3.8.

Turbine Building

The anticipated major radiation source in the turbine building is the primary steam containing activation gases, principally N-16, and fission products. Radiation shielding is provided around the following equipment in order to ensure that the zoned radiation dose rate limits are not exceeded:

1. Main condenser.
2. Feedwater heaters and drain receiver tanks.
3. Steam jet air ejectors (SJAEs).
4. Steam and extraction piping.
5. Offgas equipment and piping.
6. CNDs and regeneration facilities.
7. Steam-packing evaporators and drain tank.
8. Turbines.
9. Clean steam reboilers.
10. Moisture separator/reheaters.

Areas within these shields are high radiation zones and have limited access.

Radwaste Building

Concrete walls, removable blocks, labyrinths, and pipe chases are used to shield the process equipment in the radwaste building, including valves and piping, in accordance with the general guides of Section 12.3.2.2.1. The radioactive waste cask room is shielded above by the 2-ft concrete building roof.

Main Control Room

The main control room shielding design is based on the requirements of 10CFR50 Appendix A, GDC 19. This requires that personnel can occupy and have access to the main control room following a hypothetical maximum accident, maintain full control, and shut down the plant. Direct shielding of the main control room from the fission product inventory in the containment is provided by the concrete walls between them. Shielding of the main control room is described in Section 6.4.2.5 and shown on Figure 12.3-30. The emergency control room ventilation systems are provided for post-accident conditions in the main control room (Sections 6.4 and 9.4.1).

General Plant Yard Access

Areas adjacent to the radwaste, reactor, turbine, SGTS, CSTs, offgas, and auxiliary boiler buildings that are accessible on an unlimited basis are designed to have dose rates less than 0.2 mRem/hr. Protection for these areas is afforded by concrete building walls and by access-controlling fences. Occasionally, dose rates outside of the buildings may exceed 0.2 mRem/hr due to operational practices, such as operating the auxiliary boilers for an extended period of time using CNS as the supply, storage of radioactive materials in the yard, processing of radwaste in the radwaste truck bay, etc. Yard areas found to have dose rates in excess of 0.2 mRem/hr are evaluated, and access controlled, as necessary, to maintain exposure ALARA and conform to 10CFR20 requirements.

12.3.2.3 Method of Shielding Design

The shielding approach and methods are consistent with those described in RP-8A[1]. Where shielded entryways to compartments containing high radiation sources are necessary, labyrinths are designed using a general purpose Monte Carlo radiation transport computer code. The shielding provides a total dose rate contribution at the entryway below the upper limit of the radiation zone specified for the area.

12.3.3 Ventilation

12.3.3.1 Design Objectives

The primary function of the plant ventilation systems is to provide:

1. Safe and comfortable environmental conditions, and effective protection against possible airborne radioactive contamination, for operating personnel.

2. Adequate environmental conditions for the functioning of equipment.

The systems are designed to operate in such a way that the in-plant airborne radioactivity levels for normal operation (including anticipated abnormal operational occurrences) for RCAs within plant structures containing radioactive components are within the limits of 10CFR20 Appendix B, Table 1. For other buildings, airborne radioactivity levels are maintained within the limits of 10CFR20 Appendix B, Table II. Section 12.2.2 discusses airborne radionuclide sources. Exposures within the plant are discussed in Section 12.4.1.

The main control room ventilation system is designed to provide a suitable environment for equipment and continuous personnel occupancy in the main control room under normal and accident conditions in accordance with 10CFR50 Appendix A, GDC 19.

12.3.3.2 Design Guidelines

In order to limit and reduce the airborne radioactivity in accordance with the design objectives, the following general design guidelines are employed to the maximum extent practicable.

<u>Guidelines to Minimize Airborne Radioactivity</u>

1. For radioactive systems, equipment vents and drains are piped directly to a collection device connected to the collection system instead of allowing any contaminated fluid to flow across the floor to a floor drain.

2. All-welded piping systems are employed on contaminated systems to the maximum extent practicable. Gasketed flanged connections are used for equipment where maintenance considerations require flanged connections to facilitate removal of components and reduce personnel exposure.

3. To minimize the amount of airborne radioactivity as a result of valve leakage, valves containing radioactive fluids are designed to minimize packing leakage. For example, plug valves are used extensively in the liquid waste management system (Section 11.2.3.2). Valves are positioned whenever possible so that the radioactive fluid pressure is normally against the seat rather than the packing.

4. Contaminated equipment is designed to minimize the potential for airborne contamination during maintenance operations. These features may include flush connections on pump casings for draining and flushing the pump prior to maintenance or flush connections on piping systems that could become highly radioactive.

5. All sinks and chemical laboratory work areas where radioactive samples or materials are handled are provided with exhaust hoods to protect operating personnel from airborne contaminants.

6. Transient airborne contamination may result due to maintenance. Special procedures such as system isolation, the use of flexible vent duct or portable air handling units, and the use of glove boxes or plastic tents are instituted to minimize the contamination on a case-by-case basis.

7. Filters in all systems are changed based upon the air flow and the pressure drop across the filter bank.

8. While the majority of the accumulated radioactivity in the filtration units is removed by replacing the contaminated filters, further decontamination of the internal structure is facilitated by the proximity of

electrical outlets and water supply for operation of decontamination equipment, if necessary. Drains are provided on the filter housings for removal of contaminated water.

Guidelines to Control Airborne Radioactivity

1. Building HVAC systems are designed to provide air movement from areas of low potential radioactivity to areas of progressively higher potential radioactivity prior to final exhaust.

2. The turbine, radwaste, auxiliary boiler, and reactor buildings are kept under negative pressure to minimize exfiltration of contamination. Positive pressure is maintained in the main control room during normal operation and after postulated accidents to minimize infiltration of potential contaminants.

3. High-efficiency particulate air (HEPA) filters are provided on the radwaste building and shops exhaust to remove airborne radioactivity and to reduce onsite and offsite radiation levels. The HEPA filter units for the control building and the SGTS are designed and constructed in accordance with RG 1.52 with the exceptions listed in Section 1.8.

4. The fresh air supply to the main control room is from the two air intakes and is designed to be operable during a loss of offsite power (LOOP) and a LOCA. Upon detection of outside airborne radioactivity the air is filtered and passed through HEPA filters and charcoal adsorbers to prevent contamination of the control room.

5. The control room ventilation system and SGTS have redundant Category I equipment to control the spread of airborne radioactivity. The extent to which redundant components are provided is discussed in Sections 6.4, 6.5, and 9.4.

6. The containment purge system reduces airborne radioactivity within the drywell to acceptable levels prior to personnel entry. The purge air is processed through the SGTS prior to exhaust to remove airborne iodine and particulates.

Guidelines to Minimize Personnel Exposure from HVAC Equipment

1. The ventilation fans and filters have adequate access space to permit servicing with minimum personnel exposure (Figures 12.3-67 and 12.3-68). The HVAC system is designed to allow rapid replacement of components.

2. Air is recirculated in clean areas only during normal operation.

3. Access and service of ventilation systems in radioactively contaminated areas is expedited by component location to minimize personnel exposure during maintenance, inspection, and testing as follows:

   a. In most cases, the outside air supply units and building exhaust system components are located in ventilation equipment rooms. These units are located to maintain personnel exposure ALARA and are accessible during plant operation. Work space is provided around each unit for anticipated maintenance, testing, and inspection.

   b. Local cooling equipment servicing the normal building requirements is generally located in areas of low contamination potential, Radiation Zones IV, V, and VI. The primary containment unit coolers as well as several turbine building unit coolers are located in Zone I areas; these are expected to require a minimum of maintenance which can be scheduled during shutdown.

12.3.3.3 Design Descriptions

The ventilation systems that have been designed in accordance with the guidelines in Section 12.3.3.2 are described in Sections 9.4 and 6.4.

The expected gaseous effluents for radiologically significant areas, such as the radwaste building, reactor building, and turbine building, are provided in Section 11.3.

12.3.3.4 Air Cleaning System Description

The air cleaning systems that utilize special filtration equipment to limit airborne radioactive contaminants are:

1. SGTS (Section 6.5).

2. Control room ventilation system (Sections 9.4.1 and 6.4).

3. Containment purge system (Section 9.4.2).

4. Radwaste building exhaust filtration system (Section 9.4.3).

5. Mechanical vacuum pump system (Section 10.4.2.2.1).

The guidance and recommendations of RG 1.52 concerning maintenance, in-place testing provisions for atmospheric cleanup

systems, and air filtration and adsorption units have been used as a reference in the design of the various safety-related charcoal filter systems. The extent to which RG 1.52 has been followed is discussed in Section 6.5. Nonsafety-related charcoal filters are designed to meet the requirements of RG 1.140 as discussed in Section 9.4.3.

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

1. The filter adsorption of radioactive material during normal plant operation is a slow process; therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity buildup and the filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard.

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

3. The safety-related filter units are located in separate independent rooms. Adequate access to active elements is provided to simplify element handling. Ample space is provided in the filter rooms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling facilities, and during any in-place testing operation. The provision for in-service inspection, testing, and decontamination of charcoal filter units is conducted in accordance with the requirements discussed in Technical Specifications.

4. Typical layout with minimum distances for access and servicing is shown on Figures 12.3-67 and 12.3-68. No filter bank is more than three filter units high; each filter unit is 2 ft by 2 ft.

5. The clear space for doors throughout the plant is a minimum of 2 ft 6 in by 7 ft, providing for easy bulk movement of filter sections.

6. The filters are designed with replaceable 2-ft by 2-ft units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with access panels that are closed against compression gaskets.

7. The safety-related charcoal adsorber cells are gasketless type designed for ease of installation and replacement. The cell module is designed so that charcoal can be added to or removed from the cell as necessary without violating the integrity of the cell.

Internally-mounted test canisters are provided for laboratory testing of charcoal.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Objectives

The area radiation monitoring system (ARMS) is provided to supplement the personnel and area radiation survey provisions of the radiation protection program (Section 12.5) to ensure compliance with the personnel radiation protection guidelines of 10CFR20, 10CFR50, and 10CFR70.

Consistent with this purpose, the area radiation monitors function to:

1. Alert plant personnel entering or working in nonradiation or low radiation areas of increasing or abnormally high radiation levels.

2. Provide the main Control Room Operators with a record and indication of the occurrence and the approximate location of an abnormal radiation increase in nonradiation or low radiation areas.

3. Comply with the requirements of 10CFR50 Appendix A, GDC 63, for monitoring fuel and waste storage and handling areas.

4. Monitor the new fuel storage area for criticality in compliance with requirements of 10CFR70.24.

5. Comply with the requirements of RG 1.97 and NUREG-0737, Section II.F.1, to monitor post-accident radiation levels in the reactor containment.

6. In general, assist in maintaining personnel exposures ALARA.

The ARMS has no active function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment.

12.3.4.1.1 Area Radiation Monitoring System Design Criteria

The following design criteria are applicable to the ARMS:

1. <u>Range</u> To cover the various ranges anticipated in the plant, five different models of the basic instrument with the following scales are provided:

   10^{-2} to 10^{3} mRem/hr
   10^{-1} to 10^{4} mRem/hr

10^0 to 10^5 mRem/hr
10^{-1} to 10^4 Rem/hr
10^0 to 10^7 Rem/hr (for containment post-accident monitoring)

2. Alarms Each area radiation monitor has a red beacon to alarm on either an alert or high radiation condition, and a horn for an audible alarm on either alert or high radiation. A channel failure alarm light, which is on during normal operation, turns off upon detection of channel failure. All alarms are annunciated in the main control room. Alarm setpoints are adjustable over the range of the detector.

3. Sensitivity Area monitors are sensitive to gamma radiation of photon energies 100 keV and above.

4. Environmental Conditions The area monitors are designed to operate in the normal environmental conditions for the areas in which they are located for the design life of the plant. The post-accident area monitors are designed to remain functional for conditions described in Section 3.11.

12.3.4.1.2 Criteria for Location of Area Monitors

Generally, area radiation monitors are provided in areas to which personnel normally have access and for which there is a potential for personnel to receive radiation doses in excess of the radiation zone designations. Plant areas that meet one or more of the following criteria are monitored:

1. Areas which, during normal plant operations including refueling, could exceed radiation limits due to system failure or personnel error.

2. Areas that are continuously occupied following an accident to perform plant shutdown.

3. Areas where new fuel is received and stored. The five refuel floor area radiation monitors are located such that a minimum of two detectors serve as criticality alarms, as specified by 10CFR70.24.

4. Areas which may require access post-accident (see Section 12.3.4.3).

5. Area monitors are located in accordance with the requirements in GDC 63 of 10CFR50 Appendix A.

6. Post-accident containment monitors are located in accordance with the requirements in GDC 64 of 10CFR50 Appendix A, RG 1.97, and NUREG-0737, Section II.F.1.

12.3.4.1.3 System Description (Area Radiation Monitoring)

The ARMS detects and measures ambient gamma radiation levels at various locations. It also provides audible and visual alarms in monitored areas and the main control room if gamma radiation exceeds a specified limit. It provides visual indication in the area monitored and at a main control room annunciator if there is a malfunction in any area monitor.

Each area channel consists of a detector assembly, a check source assembly, a data acquisition unit, indicators, and alarms. All monitors are independent, and failure of one monitor has no effect on any others.

The area radiation monitors are powered from the 120-V ac regulated bus. Standby power to this bus is provided by the station battery through an inverter for balance of plant equipment. The redundant post-accident containment monitors are powered by 120-V ac divisional buses.

The location of each area radiation detector is indicated on Figures 12.3-1 through 12.3-33 and is listed in Table 12.3-1.

12.3.4.1.4 Safety Evaluation

The ARMS is not essential for the safe shutdown of the plant, and serves only to warn plant personnel of high radiation levels in various plant areas. Except for the four high range containment monitors and vital area monitors present for post-accident monitoring, the system serves no active post-emergency function.

The ARMS is designed to operate unattended for extended periods of time. A visual display of ambient radiation dose rate and trend information for any detector is available on demand in the main control room. These monitors provide audible and visual alarms at the detector and annunciate in the main control room if the radiation levels exceed preset limits. Strip chart recorders located on the radiation monitoring panel system (2CEC*P880) in the main control room provide a permanent record of the radiation levels for the four post-accident safety-related containment monitors. Also, LED indication is provided on the radiation monitor panel in the main control room for quick Operator assessment of post-accident monitoring conditions.

12.3.4.1.5 Calibration and Testing

Each of the monitors is calibrated by the instrument manufacturer prior to shipment using sources certified by or traceable to the National Institute of Standards and Technology (NIST). In-plant calibration is performed using a standard radioactive source traceable to NIST.

The proper functioning of each monitor is verified periodically by checking the instrument response to portable survey instrument response in the course of performing routine surveys. Specific requirements for some monitors are described in Technical Requirements Manual (TRM) Section 3.3.7.1.

12.3.4.2 Airborne Radioactivity Monitoring Objectives

Airborne radioactivity monitoring is provided in compliance with 10CFR20 and RG 1.45. The purpose of the airborne radioactivity monitoring system is to monitor the air within an enclosure by direct measurement of either the enclosure atmosphere or the exhaust air from the enclosure. The system indicates and records the levels of airborne radioactivity and, if abnormal levels occur, actuates alarms. Alarms are provided to alert personnel that airborne radioactivity is at or above the selected setpoint level to ensure that personnel are not subjected to airborne radioactivity above the limits in 10CFR20. The system provides a continuous record of airborne radioactivity levels, which aids operating personnel in maintaining airborne radioactivity at the lowest, reasonably achievable level.

The main objective of the in-plant airborne radioactivity monitoring systems required for safety is to initiate appropriate manual or automatic protective actions to limit the potential release of radioactive materials from the reactor vessel, primary or secondary containment, or the intake of radioactivity into the main control room if predetermined radiation levels are identified in major HVAC streams. Additional objectives are to have those systems available under all operating conditions including accidents, and to provide main control room personnel with an indication of the radiation levels in the major HVAC streams plus alarm annunciation if high radiation levels are detected.

The safety-related airborne radioactivity monitoring systems provided to meet these objectives are:

1. Reactor building ventilation exhaust (above refueling floor).

2. Reactor building ventilation exhaust (below refueling floor).

3. Main control room air intakes.

4. Drywell atmosphere monitoring (drywell area monitors cover post-accident monitoring).

The main objective of the in-plant airborne radioactivity monitoring systems that are not required for safety is to monitor major building exhaust airborne radioactivity levels and to alarm in the main control room when predetermined levels are exceeded.

The nonsafety-related radiation monitoring systems include:

1. Main stack exhaust (isotopic monitoring).

2. Containment purge discharge.

3. Turbine building ventilation.

4. Radwaste building ventilation.

5. Radwaste tank exhausts.

6. Combined reactor/radwaste building ventilation exhaust (isotopic monitoring).

12.3.4.2.1 Airborne Radioactivity Monitoring System Design Criteria

The criteria for determining the type of airborne radioactivity monitoring systems are based on the nature and type of radioactive releases expected and the location being monitored. The guidance of ANSI N13.1 and RG 1.21 is followed for the airborne radioactivity monitoring system design.

In the case of the drywell radioactivity monitoring system, which is used to detect leakage from the reactor coolant pressure boundary (RCPB), the guidance of RG 1.45 is followed.

The primary design criteria for the safety-related in-plant airborne radioactivity monitoring systems are to:

1. Withstand the effect of natural phenomena (e.g., earthquakes) without the loss of capability to perform their functions.

2. Perform their intended safety functions under normal, abnormal, and postulated accident conditions (Section 3.11).

3. Meet the reliability, testability, independence, and failure mode requirements of engineered safety features (ESF).

4. Provide continuous display on main control room panel.

5. Permit the checking of the operational availability of each channel during reactor operation with provision for calibration function and instrument checks.

6. Assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Additional criteria are found in Section 11.5.1.2.

The primary design criteria for the nonsafety-related in-plant airborne radioactivity monitoring systems are to:

1. Provide continuous data output in the main control room of radiation levels in selected building exhaust systems.

2. Permit checking the operational availability of each channel during reactor operation with provision for calibration function and instrument checks.

3. Perform their intended functions under normal operating conditions for the design life of the plant.

Additional criteria are found in Section 11.5.1.2.

12.3.4.2.2 Criteria for Airborne Radioactivity Monitor Locations

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

1. Airborne radioactivity monitors sample the drywell atmosphere for reactor pressure boundary leak detection.

2. The outside air intake ducts for the main control room area are monitored to measure the possible introduction of radioactive materials into the main control room to ensure habitability of those areas requiring personnel occupancy for safe shutdown.

3. Exhaust ducts servicing an area containing processes which, in the event of a major leakage, could result in concentrations within the plant approaching the limits established by 10CFR20 for plant workers are monitored.

Monitor sensitivity criteria are noted in Section 12.3.4.2.5.

Airborne process and effluent radiation monitor locations and functions are summarized in Table 12.3-2. ANSI N13.1 was used as a guide in locating monitors and sample points. Monitor locations are shown on the shielding arrangement and facilities drawings, Figures 12.3-1 through 12.3-33.

12.3.4.2.3 System Description (Airborne Radioactivity Monitors)

<u>Monitors Required for Safety</u>

Drywell Atmosphere Monitoring The drywell atmosphere radiation monitors are designed for early RCPB leak detection in accordance with RG 1.45.

Redundant off-line gas and particulate monitors located in the reactor building are dedicated to sampling the drywell atmosphere. Samples are drawn from the various elevations of drywell air by the use of sampling trees, are pumped through the monitoring system, and then are returned to the drywell. Each sample is continuously monitored for particulate and gaseous activities. A complete monitor description is found in Section 11.5.2.1.1. A removable iodine cartridge filter, which may be used for laboratory analysis, is provided between the moving particulate filter and the gas sample chamber. Alarms are provided for alert or high radiation levels for each channel. Alarms are also provided for channel or sampling system component failure. All alarms are annunciated locally at the monitor and in the main control room. Recorders are provided in the main control room to maintain a permanent record of drywell radiation levels.

Reactor Building Ventilation Exhaust One off-line gas and particulate and one off-line gas monitor (Section 11.5.2.1.1) are provided on the reactor building ventilation exhaust air ductwork, both above and below the refueling floor. Their function is to indicate the airborne levels of activity in the reactor building. Sampling is performed by an isokinetic sampling system. On a high radiation alarm signal the reactor building ventilation intake and exhaust air is isolated and the reactor building air is recirculated, with a small fraction being diverted through the SGTS and exhausted after treatment.

Main Control Room Ventilation The main control room ventilation radiation monitors are designed to measure the radiation levels in the junction of the two inlet ducts of the main control room ventilation system, and automatically divert the air through the emergency filter system on detection of high radiation. Dual redundant off-line gas monitors (Section 11.5.2.1.1) are provided in the main control room outside air intake ducts' junction. Sampling is performed by a sampling system with probes and returns located near the intakes. The monitors for this intake are located in the control building.

The main control room radiation monitors provide a single channel for gaseous activity only. Fixed-particulate and iodine filters are located upstream of the gas sample chamber and can be removed for analysis. Alert and high radiation levels, channel failure, and sampling system failure are alarmed locally and in the main control room.

Nonsafety-Related Process and Effluent Airborne Radiation Monitoring Systems

The nonsafety-related airborne radiation monitoring systems are designed to comply with RG 1.21 for monitoring all effluent streams and major contributing process streams for a more accurate determination of the airborne radiation levels in the final effluent, and for identification of the sources of airborne radiation in the plant.

Main Stack Exhaust

The following process ventilation systems exhaust through the main stack:

1. Containment purge discharge (SGTS exhaust).
2. Condenser mechanical vacuum pump discharge.
3. Turbine building ventilation system exhaust air.
4. Gland seal condenser gaseous discharge.
5. Offgas system exhaust.

The main stack is monitored by an off-line effluent monitor (Section 11.5.2.1.1). Sampling of the main stack exhaust is performed by an automatic isokinetic sampling system with the probe and return located in the stack. High and alert noble gas radiation levels are alarmed in the main control room. The main stack monitor has sufficient range to cover releases during and following an accident.

Additional requirements for main stack noble gas effluent radiation monitoring will be provided in accordance with Task II.F.1.1 in Section 1.10.

Containment Purge Discharge This effluent flows through the SGTS and is subsequently monitored by the SGTS effluent off-line gas radiation monitor.

Turbine Building Ventilation Exhaust A connection tap is provided in the system ductwork for a continuous air monitor (Section 11.5.2.1.3).

Combined Radwaste/Reactor Building Ventilation Exhaust The reactor/radwaste building ventilation exhaust off-line effluent monitor is designed to measure radionuclide releases to be reported and evaluated in accordance with RG 1.21. Since the effluent from general areas of the reactor building is unfiltered, this monitor also indicates general airborne levels of radiation in the reactor building.

Sampling is performed by an automatic isokinetic sampling system with the probe and return located in the combined exhaust duct. The building ventilation monitor is an off-line gaseous type. Alert or high noble gas radiation levels are alarmed in the

control room. Major process streams exhausted through the reactor building vent include the reactor building and radwaste building ventilation exhaust air. The reactor/radwaste vent monitor has extended range to cover post-accident monitoring requirements.

12.3.4.2.4 Safety Evaluation

The reactor building ventilation exhaust and the main control room ventilation airborne radioactivity monitors have the safety-related functions of isolating their particular ventilation systems and actuating the associated filtered emergency systems, as discussed in Sections 12.3.4.2.1 through 12.3.4.2.3. These monitors are redundant, Category I, and powered from the emergency power system (Chapter 8).

The combination of the airborne radioactivity monitoring system in conjunction with the administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns through the plant, and plant equipment layout is sufficient to minimize personnel radiation exposure throughout all areas of the plant where access is required.

12.3.4.2.5 Sensitivities and Ranges

Each particulate monitoring system, with the exception of the drywell atmosphere contamination monitors and the radwaste building equipment exhaust continuous air monitor (CAM), requires a minimum detectable concentration such that 10 MPC-hr (25 percent of maximum permissible concentration as defined in 10CFR20) of airborne particulate and iodine radioactivity can be detected in any compartment, given a minimum cubicle flow rate which has the possibility of airborne contamination. Any cubicle that has the possibility of airborne contamination and has an exhaust flow rate less than the minimum to detect 10 MPC-hr requires access control and surveys to ensure regulatory requirements are met. Drywell atmosphere contamination monitors require a minimum detectable activity for leak detection capability while 2HVW-RE196 monitor high alarm setpoint is based on offsite dose limits. Monitor sensitivities, ranges, and limiting isotopes are indicated in Table 12.3-2.

A strip chart recorder located on the radiation monitor panel in the main control room provides a permanent record of the radiation levels for each safety-related channel. Also, LED indication is provided by the digital radiation monitoring system (DRMS) (Section 11.5.1.1.2) in the main control room for quick Operator assessment of the control room ventilation.

The continuous monitoring system consists of radiation monitors initially calibrated by the manufacturer. Calibration standards are traceable to the NIST.

The count rate response of each continuous monitor to remotely positionable check sources supplied with each monitor is recorded by the manufacturer after the primary calibration, again after installation, and together with the instrument background count rate at intervals during reactor operation to ensure proper functioning of the monitors.

Decay curves are provided for the sources to permit correction for source decay. An electronic circuit check is performed through the use of an internal oscillator or pulse-generating circuit.

12.3.4.3 Accident Consideration

Areas requiring access after an accident are discussed in Section 12.3.1.3. Permanent radiation monitors are located in all vital areas on the Unit 2 site. Radiation monitoring for the two vital areas located at Unit 1, the TSC in the administration building and the chemistry laboratory in the turbine building, is provided by portable dose rate instrumentation.

Information on post-accident radiation levels is available from safety-related monitors specifically designed for this purpose. In addition, information may be available from nonsafety-related area and airborne radiation monitors.

The nonsafety area monitors and nonsafety-related airborne monitors are supplied with power from the Station battery system through an inverter. The safety-related airborne radioactivity monitors and area monitors are powered by the emergency diesel generator sets (Chapter 8).

The ranges of the nonsafety-related monitors include the maximum anticipated operational radiation levels. High or upscale radiation or radioactivity is indicated by a local audible and visible alarm, alerting personnel to leave and not to enter the affected area until a survey with portable equipment establishes that personnel will not be subjected to doses in excess of established limits. Safety-related post-accident drywell monitors have extended ranges to cover postulated accident radiation levels.

Areas other than those discussed in Section 12.3.1.3 that were reviewed as vital post-accident areas, as suggested in NUREG-0737, either do not apply or access is not required. The hydrogen recombiner system is controlled remotely from the control room. The containment isolation reset control area is located in the main control room. Unit 2 has no manual ECCS alignment area; all vital ECCS valves are automatic and remotely operable from the control room. Motor control centers (MCCs) do not need to be accessed, nor do the instrument panels located outside the control room. Access is not needed for emergency power supplies since they are remotely operable from the control room.

Access to the radwaste control panel is not required for safe shutdown of the plant; however, operation of the panel is necessary to limit the spread of contaminated fluids. Switches to stop some reactor building floor and equipment drain pumps are situated on panel 513, located in the radwaste control room. Shutdown of the drain pumps is essential to prevent discharge of collected contaminated fluids to the radwaste building. For this reason, the radwaste control room is designated as a vital area requiring only intermittent access.

Access to the security center is not needed to gain access to the control room, the TSC, the post-accident sampling station, the control panel, or the counting room area. Since the control room is always manned, entrance could be gained manually at any time, should the automatic security systems fail. Keys are available to authorized personnel to open manually the necessary doors to the TSC and post-accident sampling and analysis area if the automatic security systems should fail.

12.3.4.4 Portable Monitors

Portable, moving- or fixed-filter paper CAMs are provided and normally operated at fixed locations in the plant to provide trend indication. These locations include the radwaste tank vents, radwaste building ventilation exhaust, decontamination area, reactor building recirculation system, RHR heat exchanger cubicle ventilation, and turbine building ventilation. CAMs are upstream of any filters present. However, during a period of extensive maintenance in an area with a potential for airborne radioactivity, a CAM may be moved to the maintenance area.

Monitor sensitivity criteria are noted in Section 12.3.4.2.5. Intermittent sampling using a high-volume portable air sampler verifies and/or supplements CAM equipment. These portable air samplers are also used in areas with a potential for some airborne radioactivity. Portable instrumentation is described in Section 12.5.

12.3.5 Reference

1. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plant RP-8A. Topical Report, Stone & Webster Engineering Corporation, May 1975.

TABLE 12.3-1
AREA RADIATION MONITOR LOCATIONS

Equipment No.	Area Monitors: Description	Area Monitors: QA Category/ Seismic Category	Range	High Alarm Setpoint	Location
2RMS*RE1A(-G)	Drywell high range monitor	1/1	1 - 10^7 R/hr	(1)	Reactor building, el 261'
2RMS*RE1B(-Y)	Drywell high range monitor	1/1	1 - 10^7 R/hr	(1)	Reactor building, el 261'
2RMS*RE1C(-G)	Drywell high range monitor	1/1	1 - 10^7 R/hr	(1)	Reactor building, el 261'
2RMS*RE1D(-Y)	Drywell high range monitor	1/1	1 - 10^7 R/hr	(1)	Reactor building, el 261'
2RMS-RE101	RHS heat exchanger equipment room	N/A	10^{-1} - 10^4 mR/hr	(2)	Auxiliary bay N, el 175'
2RMS-RE102	Equipment drains sumps and pumps E	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 175'
2RMS-RE103	RHS heat exchanger equipment room	N/A	10^{-1} - 10^4 mR/hr	(2)	Auxiliary bay S, el 175'
2RMS-RE104	Equipment drains sumps and pumps W	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 175'
2RMS-RE105	TIP drive mechanism equipment area	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 240'
2RMS-RE106	Entrance area	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 261'
2RMS-RE108	CRD maintenance facility	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 289'
2RMS-RE109	Contaminated equipment storage	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 328'10"
2RMS-RE111	Spent fuel refueling area	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 353'10"
2RMS-RE112	Spent fuel refueling area	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 353'10"
2RMS-RE113	New fuel storage vault area	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 353'10"
2RMS-RE114	Operating floor equipment	N/A	10^{-1} - 10^4 mR/hr	(2)	Reactor building, el 353'10"
2RMS-RE116	Condensate pump area	N/A	10^{-1} - 10^4 mR/hr	(2)	Turbine building, el 250'
2RMS-RE117	Resin regeneration area	N/A	10^{-1} - 10^4 mR/hr	(2)	Turbine building, el 250'
2RMS-RE118	Radwaste sample room - vital area monitor	N/A	10^{-1} - 10^4 mR/hr	(2)	Turbine building, el 261'
2RMS-RE119	Truck aisle area north	N/A	10^{-1} - 10^4 mR/hr	(2)	Turbine building, el 250'
2RMS-RE120	URC flow adjustment panel	N/A	10^{-1} - 10^4 mR/hr	(2)	Turbine building, el 277'6"
2RMS-RE121	Radwaste control room - vital area monitor	N/A	10^{-2} - 10^3 mR/hr	(2)	Turbine building, el 279'

TABLE 12.3-1 (Cont'd.)

Equipment No.	Area Monitors: Description	Area Monitors: QA Category/ Seismic Category	Range	High Alarm Setpoint	Location
2RMS-RE123	Turbine operating floor entrance	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 306'
2RMS-RE125	Truck loading dock	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 265'
2RMS-RE129	Main control room - vital area monitor	N/A	10^{-2} - 10^{3} mR/hr	TRM	Control building, el 306'
2RMS-RE130	Remote shutdown panel area	N/A	10^{-2} - 10^{3} mR/hr	(2)	Control building, el 261'
2RMS-RE132	Compacted waste storage area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 265'
2RMS-RE133	Distillate roughing filters area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 279'
2RMS-RE134	Evaporator sampling area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 261'
2RMS-RE135	Air removal pumps area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 250'
2RMS-RE136	Offgas system control panel area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 288'6"
2RMS-RE137	Hot machine shop	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 261'
2RMS-RE138	Reactor feed pumps area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 250'
2RMS-RE139	Above suppression pool	N/A	10^{-0} - 10^{5} mR/hr	(2)	Reactor building suppression chamber, el 222'6"
2RMS-RE140	Inside new fuel storage vault	N/A	10^{-2} - 10^{3} mR/hr	TRM	Reactor building, el 337'2"
2RMS-RE141	Turbine building sample room	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 250'
2RMS-RE142	Near distillate condenser	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 279'
2RMS-RE143	CRD module area north	N/A	10^{-1} - 10^{4} mR/hr	(2)	Reactor building, el 261'
2RMS-RE144	CRD module area south	N/A	10^{-1} - 10^{4} mR/hr	(2)	Reactor building, el 261'
2RMS-RE145	Sample sink	N/A	10^{-1} - 10^{4} mR/hr	(2)	Reactor building, el 240'
2RMS-RE146	Spent resin cask capping room	N/A	10^{0} - 10^{5} mR/hr	(2)	Radwaste building, el 279'
2RMS-RE147	Extruder evaporator/turntable area	N/A	10^{1} - 10^{5} mR/hr	(2)	Radwaste building, el 291'
2RMS-RE148	Solid radwaste sample panel area	N/A	10^{0} - 10^{5} mR/hr	(2)	Radwaste building, el 279'
2RMS-RE2A	Recirculation pump instrument panel A	N/A	10^{-1} - 10^{4} mR/hr	(2)	Reactor building, el 215'

TABLE 12.3-1 (Cont'd.)

Equipment No.	Area Monitors: Description	Area Monitors: QA Category/ Seismic Category	Range	High Alarm Setpoint	Location
2RMS-RE2B	Recirculation pump instrument panel B	N/A	10^{-1} - 10^{4} mR/hr	(2)	Reactor building, el 215’
2RMS-RE3A	Turbine building, heater bay area	N/A	10^{1} - 10^{6} mR/hr	(2)	Turbine building, el 250'
2RMS-RE3B	Turbine building, heater bay area	N/A	10^{1} - 10^{6} mR/hr	(2)	Turbine building, el 250'
2RMS-RE3C	Turbine building, heater bay area	N/A	10^{1} - 10^{6} mR/hr	(2)	Turbine building, el 250'
2RMS-RE149	Reactor building, RWCU valve area	N/A	10^{1} - 10^{6} mR/hr	(2)	Reactor building, el 328'10"
2RMS-RE150	Turbine building, resin regeneration room	N/A	10^{1} - 10^{6} mR/hr	(2)	Turbine building, el 250'
2RMS-RE151	Turbine building, low-pressure turbine area	N/A	10^{1} - 10^{6} mR/hr	(2)	Turbine building, el 306'
2RMS-RE152	Radwaste building, floor drain sump area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 240'2"
2RMS-RE153	Radwaste building, LWS pumps area	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 240'2"
2RMS-RE154	Turbine building, condenser area monitor	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 250'
2RMS-RE190	Control building, relay and computer room - vital area monitor	N/A	10^{-2} - 10^{3} mR/hr	(2)	Control building, el 288'6"
2RMS-RE191	Turbine building, low level counting room - vital area monitor	N/A	10^{-2} - 10^{3} mR/hr	(2)	Turbine building, el 306'
2RMS-RE192	Turbine building, gaseous effluent monitor location - vital area monitor	N/A	10^{-1} - 10^{4} mR/hr	(2)	Turbine building, el 306'
2RMS-RE193	Main stack - gaseous effluent monitor location - vital area monitor	N/A	10^{-1} - 10^{4} mR/hr	(2)	Main stack, el 261'
2RMS-RE194	Radwaste building - low level radwaste storage area - general area monitor	N/A	10^{-1} - 10^{4} mR/hr	(2)	Radwaste building, el 245’

(1) Setpoints for these monitors are determined in accordance with the Site Emergency Plan.

(2) Normal operation plant area radiation monitor setpoints are such that personnel will be warned of increasing ambient radiation levels. Setpoints are based upon Technical Specification access control requirements areas with dose rates greater than 1,000 mrem/hr or the 10CFR20 dose rate criteria for unrestricted areas, radiation areas, and high radiation areas, whichever is applicable.

TABLE 12.3-2

AIRBORNE PROCESS AND EFFLUENT RADIATION MONITORS

Equipment No.	QA Category/ Seismic Category	Monitor Name and Location	Monitor Type	Detector Channels: Channel	Detector Channels: Limiting Isotopes	Detector Channels: Range (uCi/cc)	High Setpoint (Basis)	Functions (Covered in Text)
2CMS*RE10A (AG)	1/1	Drywell atmosphere A reactor building el 289'	Offline gas and particulate	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Leak detection Leak detection	Yes
2CMS*RE10B (BY)	1/1	Drywell atmosphere B reactor building el 289'	Offline gas and particulate	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Leak detection Leak detection	Yes
2GTS-RE105	N/A	Standby gas treatment main stack el 261'(1)	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Occupational dose limits	Yes
2HVC*RE18A (ZG)	1/1	Control room air intake A control building el 306'	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Occupational dose limits	Yes
2HVC*RE18B (ZY)	1/1	Control room air intake B control building el 306'	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Occupational dose limits	Yes
2HVC*RE18C (ZG)	1/1	Control room air intake C control building el 306'	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Occupational dose limits	Yes
2HVC*RE18D (ZY)	1/1	Control room air intake D control building el 306'	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Occupational dose limits	Yes
2HVR*RE14A (AG)	1/1	Above refueling floor, reactor building el 328'10"	Offline gas and particulate	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Occupational MPC limit	Yes
2HVR*RE14B (BY)	1/1	Above refueling floor, reactor building el 328'10"	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Offsite dose limits	Yes

TABLE 12.3-2 (Cont'd.)

Equipment No.	QA Category/ Seismic Category	Monitor Name and Location	Monitor Type	Detector Channels: Channel	Detector Channels: Limiting Isotopes	Detector Channels: Range (uCi/cc)	High Setpoint (Basis)	Functions (Covered in Text)
2HVR*RE32A (AG)	1/1	Below refueling floor, reactor building el 289'	Offline gas and particulate	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Lowest setpoint which will not result in spurious alarms	Yes
2HVR*RE32B (BY)	1/1	Below refueling floor, reactor building el 289'	Offline gas	Gas	KR-85 XE-133	10^{-7} - 10^{-1}	Offsite dose limits	Yes
2HVW-RE195	N/A	Radwaste building equipment exhaust before HEPA filter	Continuous airborne monitor	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Occupational MPC limits	Yes
2HVW-RE196	N/A	Radwaste tank vent before HEPA filter	Continuous airborne monitor	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Offsite dose limits	Yes
2HVW-RE197	N/A	Radwaste building vent before HEPA filter	Continuous airborne monitor	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Occupational MPC limits	Yes
2HVR-RE237	N/A	Reactor building el 175' RHR heat exchanger cubicle - north	Continuous airborne monitor	Gas Particulate	Kr-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Occupational MPC limits Occupational MPC limits	Yes
2HVR-RE238	N/A	Reactor building el 175' RHR heat exchanger cubicle - south	Continuous airborne monitor	Gas Particulate	Kr-85 Xe-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Occupational MPC limits Occupational MPC limits	Yes
2HVW-RE199	N/A	Radwaste equipment decon service area exhaust, turbine building el 261'	Continuous airborne monitor	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Occupational MPC limits	Yes

TABLE 12.3-2 (Cont'd.)

Equipment No.	QA Category/ Seismic Category	Monitor Name and Location	Monitor Type	Detector Channels: Channel	Detector Channels: Limiting Isotopes	Detector Channels: Range (uCi/cc)	High Setpoint (Basis)	Functions (Covered in Text)
2HVT-RE206	N/A	Turbine building vent (common duct)	Continuous airborne monitor	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Offsite dose limits Occupational MPC limits	Yes
2HVR-RE229	N/A	Reactor recirculation mode	Continuous airborne monitor	Gas Particulate	KR-85 XE-133 I-131	10^{-7} - 10^{-1} 10^{-11} - 10^{-5}	Occupational MPC limits Occupational MPC limits	Yes
2RMS-RE180A,B,C	N/A	Reactor/radwaste building vent monitor, turbine building el 306(2)	Offline gaseous	Gas(3)	KR-85 XE-133	10^{-6} - 10^{5}	Offsite dose limits	Yes
2RMS-RE170A,B,C	N/A	Main stack monitor, main stack el 261'(2)	Offline gaseous	Gas(3)	KR-85 XE-133	10^{-6} - 10^{5}	Offsite dose limits	Yes

(1) Monitors containment purge system exhaust.
(2) The low end of the ranges required by NUREG-0473 and the high end of the ranges required by RG 1.97 are given.
(3) KR-85 is used to establish calibration curve and is subsequently removed from analysis during normal operation to avoid erroneous identification.

TABLE 12.3-3

PERSONNEL DOSES FOR REQUIRED OCCUPANCY TIMES IN VITAL AREAS

Vital Area	Task Performed	Occupancy Time	Dose (rem)[7]	Notes
Main control room and relay and computer room	Execute the safe shutdown of the plant	Continuous for 30 days	1.74+0	30-day average dose rate = 4.0 mRem/hr
Health physics/counting room Unit 1	Perform routine health physics functions	8 hr	3.32-1	Dose based upon continuous occupancy for an 8-hr workday at the time of maximum dose rate
Radwaste sample room/Unit 1 chemistry lab (PASS)[1,2]	a. Obtain and perform general isotopic and Boron analysis of dilute reactor coolant sample[3]	1 hr, 5 min	2.35+0 2.90+0	Whole body Extremity
	b. Obtain and perform isotopic analysis of containment atmosphere sample[3]	52 min	2.59+0 5.09+0	Whole body Extremity
	c. Determine level of dissolved gases (e.g., H_2) in reactor coolant	1 hr, 45 min	8.74-1 8.74-1	Whole body Extremity
	d. Obtain and perform chloride analysis of undiluted reactor coolant sample[3]	1 hr, 18 min	5.68+0 2.94+1	Whole body Extremity
Main stack off-line monitor	a. Manual sampling[5]	53 min	3.82+0 3.90+0	Whole body Extremity Dose includes dose received for one round trip between the OSC and the monitor location

TABLE 12.3-3 (Cont'd.)

Vital Area	Task Performed	Occupancy Time	Dose (rem)[7]	Notes
Radwaste control room	a. Turn off reactor building equipment and floor drain pumps b. Service ERF computer system	12 min 22 min	8.34-1 1.13+0	Dose includes dose received for one round trip between the OSC and the radwaste control room
A round trip between the OSC and the control room emergency zone	For information only	6 min	2.68-1	
Technical support center	Per NUREG-0696	Continuous for 30 days in accordance with the Site Emergency Plan procedures	4.30+0[6]	Whole body

(1) Maximum dose rate for each subtask was used to develop the maximum dose for the task.
(2) See Section 1.10, Item II.B.3, for specific information on the post-accident sampling system and Table II.B.3-l for a breakdown of the tasks and required occupancy times.
(3) Dose includes exposure received for one round trip from the OSC, to the radwaste sample room, to the Unit 1 chem lab, and back to the OSC.
(4) Deleted.
(5) Dose includes exposure received for one round trip from the OSC, to the main stack, to the Unit 1 counting room and back to the OSC.
(6) "Dose (rem)" values will increase by 22.4 percent due to EPU.
(7) Doses do not reflect credit for iodine removal in the suppression pool, as applied per SRP Section 6.5.5, Revision 0.

TABLE 12.3-4

DOSE RATE (REM/HR) AT LOCATION:*[(1)]

Time Post-LOCA (Hr)	A	B	C	D	E	F	G	H	I
1	9.91+1	9.93+1	1.00+2	1.00+2	1.00+2	1.01+2	1.21+3	1.21+3	1.21+3
3	2.76+0	3.05+0	4.17+0	4.54+0	4.54+0	5.24+0	5.59+0	4.87+0	4.96+0
6	2.94+0	3.30+0	4.30+0	4.80+0	4.80+0	5.67+0	6.08+0	5.63+0	5.68+0
9	3.05+0	3.41+0	4.48+0	4.99+0	4.99+0	5.94+0	6.38+0	5.94+0	5.99+0
12	2.15+0	2.53+0	3.32+0	3.87+0	3.87+0	4.85+0	5.16+0	5.08+0	5.05+0
18	2.03+0	2.43+0	3.17+0	3.73+0	3.73+0	4.74+0	5.05+0	5.06+0	5.04+0
24	1.96+0	2.36+0	3.02+0	3.60+0	3.60+0	4.60+0	4.90+0	5.02+0	4.96+0
50	1.57+0	1.94+0	2.44+0	2.94+0	2.94+0	3.86+0	4.14+0	4.35+0	4.28+0
100	1.54+0	1.91+0	2.39+0	2.91+0	2.91+0	3.85+0	4.13+0	4.37+0	4.28+0
200	1.50+0	1.87+0	2.32+0	2.84+0	2.84+0	3.76+0	4.04+0	4.29+0	4.21+0
400	9.25-1	1.15+0	1.44+0	1.75+0	1.75+0	2.32+0	2.50+0	2.66+0	2.61+0
550	5.39-1	6.66-1	8.28-1	1.01+0	1.01+0	1.34+0	1.45+0	1.54+0	1.50+0
720	2.52-1	3.14-1	3.89-1	4.76-1	4.76-1	6.32-1	6.82-1	7.27-1	7.12-1
	J	K	L	M	N	O	P	Q	R
1	1.21+3	1.21+3	1.21+3	1.21+3	1.21+3	1.21+3	1.43+2	2.45+2	2.45+2
3	4.59+0	3.82+0	2.63+0	2.48+0	2.48+0	4.44+0	5.05+0	5.28+0	4.95+0
6	5.21+0	4.27+0	2.80+0	2.68+0	2.68+0	5.09+0	5.85+0	6.08+0	5.86+0
9	5.49+0	4.47+0	2.89+0	2.76+0	2.76+0	5.37+0	6.16+0	6.39+0	6.18+0
12	4.55+0	3.48+0	1.84+0	1.78+0	1.78+0	4.48+0	5.20+0	4.79+0	4.76+0
18	4.51+0	3.43+0	1.75+0	1.70+0	1.70+0	4.46+0	5.20+0	4.77+0	4.76+0
24	4.45+0	3.36+0	1.68+0	1.65+0	1.65+0	4.41+0	5.14+0	4.70+0	4.69+0
50	3.82+1	2.85+0	1.34+0	1.33+0	1.33+0	3.81+0	4.47+0	4.08+0	4.08+0
100	3.82+0	2.85+0	1.33+0	1.33+0	1.33+0	3.82+0	4.49+0	4.10+0	4.10+0
200	3.65+0	2.78+0	1.29+0	1.29+0	1.29+0	3.65+0	4.42+0	4.02+0	4.02+0
400	2.33+0	1.73+0	7.96-1	7.96-1	7.96-1	2.33+0	2.74+0	2.50+0	2.50+0
550	1.34+0	9.94-1	4.58-1	4.58-1	4.58-1	1.34+0	1.58+0	1.45+0	1.45+0
720	6.32-1	4.72-1	2.17-1	2.17-1	2.17-1	6.32-1	7.45-1	6.82-1	6.82-1

* Refer to Figure 12.3-69.

(1) Dose rates do not reflect iodine removal by suppression pool scrubbing as applied per SRP Section 6.5.5, Rev. 0, mixing in secondary containment during the drawdown period or increase in the drawdown period from 6 min to 60 min.

TABLE 12.3-4 (Cont'd.)

Time Post-LOCA (Hr)	S	T	U	V	W	X	Y	Z	AA
1	2.45+2	2.44+2	2.44+2	2.44+2	2.42+2	2.42+2	1.24+2	1.24+2	1.22+2
3	4.95+0	4.72+0	4.03+0	3.15+0	2.45+0	2.31+0	3.95+0	2.76+0	2.63+0
6	5.86+0	5.59+0	4.72+0	3.71+0	2.84+0	2.66+0	4.37+0	2.90+0	2.80+0
9	6.18+0	5.88+0	4.93+0	3.82+0	2.90+0	2.70+0	4.58+0	3.00+0	2.89+0
12	4.76+0	4.45+0	3.47+0	2.30+0	1.33+0	1.12+0	3.70+0	2.06+0	1.84+0
18	4.76+0	4.45+0	3.44+0	2.27+0	1.28+0	1.07+0	3.59+0	1.91+0	1.75+0
24	4.69+0	4.39+0	3.39+0	2.23+0	1.23+0	1.02+0	3.47+0	1.79+0	1.68+0
50	4.08+0	3.80+0	2.89+0	1.91+0	1.02+0	8.53-1	3.35+0	1.84+0	1.34+0
100	4.10+0	3.81+0	2.87+0	1.89+0	9.96-1	8.04-1	2.86+0	1.34+0	1.33+0
200	4.02+0	3.74+0	2.82+0	1.86+0	9.70-1	7.79-1	2.79+0	1.29+0	1.29+0
400	2.50+0	2.32+0	1.75+0	1.15+0	6.01-1	4.83-1	1.73+0	7.97-1	7.96-1
550	1.45+0	1.34+0	1.01+0	6.66-1	3.46-1	2.78-1	9.97-1	4.60-1	4.58-1
720	6.82-1	6.32-1	4.76-1	3.14-1	1.64-1	1.31-1	4.72-1	2.17-1	2.17-1

TABLE 12.3-4 (Cont'd.)

Time Post-LOCA (Hr)	AB	AC	AD	AE	AF	AG	AH	AI	AJ
1	1.24+2	1.24+2	1.24+2	1.26+2	1.43+2	1.26+2	1.26+2	1.26+2	1.25+2
3	3.95+0	3.95+0	4.72+0	5.75+0	5.97+0	5.81+0	5.59+0	5.81+0	5.24+0
6	4.37+0	4.37+0	5.31+0	6.28+0	6.57+0	6.35+0	6.08+0	6.35+0	5.96+0
9	4.58+0	4.58+0	5.60+0	6.59+0	7.17+0	6.66+0	6.38+0	6.66+0	6.28+0
12	3.70+0	3.70+0	4.76+0	5.37+0	5.58+0	5.45+0	5.16+0	5.45+0	5.48+0
18	3.59+0	3.59+0	4.66+0	5.26+0	5.48+0	5.34+0	5.05+0	5.34+0	5.41+0
24	3.47+0	3.47+0	4.55+0	5.13+0	5.34+0	5.22+0	4.90+0	5.22+0	5.28+0
50	3.35+0	3.35+0	4.32+0	4.32+0	4.52+0	4.41+0	4.14+0	4.41+0	4.97+0
100	2.86+0	2.86+0	3.83+0	4.30+0	4.52+0	4.39+0	4.13+0	4.39+0	4.51+0
200	2.79+0	2.79+0	3.65+0	4.21+0	4.42+0	4.30+0	4.04+0	4.30+0	4.42+0
400	1.73+0	1.73+0	2.33+0	2.61+0	2.74+0	2.66+0	2.50+0	2.66+0	2.74+0
550	9.97-1	9.97-1	1.34+0	1.50+0	1.58+0	1.54+0	1.45+0	1.54+0	1.58+0
720	4.72-1	4.72-1	6.32-1	7.12-1	7.45-1	7.27-1	6.82-1	7.27-1	7.45-1
	AK	AL	AM	AN	AO	AP	AQ	AR	AS
1	1.26+2	1.26+2	1.00+2	9.99+1	9.99+1	9.75+2	9.91+1	9.94+1	9.91+1
3	5.81+0	5.59+0	5.00+0	4.54+0	4.54+0	3.40+0	4.17+0	3.40+0	2.76+0
6	6.35+0	6.08+0	5.35+0	4.80+0	4.80+0	3.57+0	4.30+0	3.70+0	2.94+0
9	6.66+0	6.38+0	5.60+0	4.99+0	4.99+0	3.89+0	4.48+0	3.89+0	3.05+0
12	5.45+0	5.16+0	4.51+0	3.87+0	3.87+0	3.01+0	3.32+0	3.01+0	2.15+0
18	5.34+0	5.05+0	4.38+0	3.73+0	3.73+0	2.94+0	3.17+0	2.94+0	2.03+0
24	5.22+0	4.90+0	4.25+0	3.60+0	3.60+0	2.86+0	3.02+0	2.86+0	1.96+0
50	4.41+0	4.14+0	3.54+0	2.94+0	2.94+0	2.39+0	2.44+0	2.39+0	1.57+0
100	4.39+0	4.13+0	3.52+0	2.91+0	2.91+0	2.37+0	2.39+0	2.37+0	1.54+0
200	4.30+0	4.04+0	3.44+0	2.84+0	2.84+0	2.32+0	2.32+0	2.32+0	1.50+0
400	2.66+0	2.50+0	2.13+0	1.75+0	1.75+0	1.44+0	1.44+0	1.44+0	9.24-1
550	1.54+0	1.45+0	1.23+0	1.01+0	1.01+0	8.26-1	8.28-1	8.26-1	5.39-1
720	7.27-1	6.82-1	5.81-1	4.76-1	4.76-1	3.89-1	3.89-1	3.89-1	2.52-1

TABLE 12.3-4 (Cont'd.)

Time Post-LOCA (Hr)	Health Physics Counting Room	Radwaste Sample Room	Turbine Building Online Isotopic Monitor	Main Stack Online Isotopic Monitor	Radwaste Control Room	Unit 1 Chemistry Laboratory
1	3.34+0	1.91+1	1.25+2	6.02+0	4.76+1	6.95+0
3	4.06-2	4.81-1	4.44+0	6.47+0	1.72+0	8.93-2
6	3.93-2	4.38-1	4.90+0	8.84+0	1.69+0	8.65-2
9	3.91-2	4.13-1	5.13+0	7.56+0	1.66+0	8.59-2
12	8.93-3	1.73-1	3.70+0	6.70+0	1.00+0	2.15-2
18	6.28-3	1.25-1	3.59+0	7.01+0	8.61-1	1.51-2
24	4.77-3	9.67-2	3.47+0	5.87+0	7.22-1	1.14-2
50	1.15-3	3.69-2	3.35+0	6.25+0	5.27-1	2.96-3
100	6.79-4	2.38-2	2.86+0	6.42+0	4.58-1	1.71-3
200	1.29-4	1.60-2	2.79+0	6.10+0	4.25-1	4.88-4
400	5.92-5	1.01-2	1.73+0	3.28+0	2.65-1	2.65-4
550	2.37-5	5.90-3	9.97-1	1.96+0	1.54-1	1.34-4
720	8.74-6	2.83-3	4.72-1	1.35+0	7.32-2	5.97-5

12.4 DOSE ASSESSMENT

Radiation exposures in the plant are primarily from components and equipment containing radioactive fluids, and to a lesser extent from the presence of airborne radionuclides. In-plant radiation exposures during normal operation, refueling, and anticipated operational occurrences are discussed in Section 12.4.2. Radiation exposures at onsite locations outside the plant are discussed in Section 12.4.3.

12.4.1 Design Criteria

The criteria for doses to plant personnel during normal operation and anticipated operational occurrences, including refueling, are based on the requirements discussed in 10CFR20. Zone I plant areas during normal operation, refueling, and anticipated occupational occurrences are shown on Figures 12.3-34 through 12.3-66.

Radiation exposures to operating personnel are within 10CFR20 limits. Radiation protection design features (Section 12.3) and the radiation protection program (Section 12.5) assure that the occupational radiation exposures (ORE) to operating personnel during normal operation, refueling, and anticipated operational occurrences are ALARA.

12.4.2 Exposures Within the Plant

12.4.2.1 Man-Rem Evaluation

The occupational radiation dose assessment for Unit 2 was performed using the guidelines of RG 8.19[1]. The bases for the annual man-rem estimates are operating data from Unit 1, which are modified to account for differences and improvements in Unit 2. The projected radiation dose rates throughout the plant facilities are based on assumed radiation conditions after 5 yr of plant operation and expected radiation dose rates. Table 12.4-12 presents operational data from several BWRs[2] and shows the average annual man-rem per unit over several operating years to be 948 man-rem per year. These data indicate that, in recent years, OREs have been much larger than the radiation exposures reported for operating BWR plants in the mid-1970s. The primary reason for the increase in radiation exposure has been the increase in manpower necessary to support the expanding number of special maintenance activities.

Table 12.4-13 shows the distribution of annual OREs suggested in RG 8.19[1] work functions for all BWRs over several years. The erage values indicate that operating BWR plants have approximately 76 percent annual occupational exposure attributed to routine (40 percent) and special (36 percent) maintenance. In recent years, plant modifications attributed to feedwater sparger repairs, inspection, repair and replacement of recirculation piping, Three Mile Island (TMI) lessons-learned modifications,

and increased snubber and pipe hanger inspections have contributed to the growing amount of OREs associated with special maintenance work functions. Design features described in Sections 12.1 and 12.3 for the Unit 2 BWR 5 plant should minimize the special maintenance work experienced at earlier-designed operating BWR plants.

Design improvements for Unit 2 that are expected to reduce the OREs include the following:

a. Incorporation of flush connections on the control rod drive (CRD) scram discharge volume (SDV) header permits condensate flushing of piping to minimize corrosion product holdup in a high personnel access area.

b. Use of filtered condensate water for CRD hydraulic fluid and the reactor recirculation pump seal purge provides a clean water source that should extend pump seal life.

c. Installation of permanent hoisting systems and access platforms for the recirculation pumps, main steam isolation valves (MSIVs), and safety relief valves (SRVs) minimizes maintenance time in the drywell.

d. Improved refueling platform makes fuel handling activities more efficient and reduces time spent on the platform.

e. A multistud tensioner reduces the amount of man-hours necessary to handle the reactor vessel head studs.

f. A new handling tool and platform for the removal of CRDs from beneath the reactor vessel reduces crew size and time spent in the high radiation area.

g. Improved fuel design minimizes the buildup of radiation levels near reactor coolant systems and reduces the amount of fuel assembly sipping activities.

h. Improved piping material for the recirculation system eliminates the special maintenance which was required on older BWR recirculation piping due to stress corrosion cracking.

i. ISI access is improved by remote equipment development and access doors for reactor vessel and nozzle weld inspection.

j. Decontamination methods are provided to decontaminate the walls of the reactor cavity pit and internals pool to minimize contribution from this source.

k. Use of separate shielded cubicles for locating redundant and highly-radioactive components minimizes radiation exposures during maintenance activities.

l. Compared to hydraulic-operated snubbers available during the original design of the plant, use of mechanical snubbers should reduce the frequency of necessary inspection.

m. Installation of a CRD flush tank removes highly-radioactive corrosion and fission products from the CRD internals prior to rebuilding.

n. Use of sealed hydraulic snubbers (improved design) that are less susceptible to failure than mechanical snubbers.

The ORE for Unit 2 is determined for each of the RG 8.19[1] work function categories by identifying specific tasks within each of the seven work function categories and determining the time and manpower requirements for those tasks. This information is used with the expected dose rates in the areas where work is performed to determine radiation exposure from each activity. Tables 12.4-5 through 12.4-11 provide the estimates of occupational exposures based on the identification of specific tasks within each of the seven work function categories: routine operations and surveillance, nonroutine operation and surveillance, routine maintenance, radwaste processing, refueling, ISI, and special maintenance. Table 12.4-4 summarizes the occupational dose estimates for the seven work functions. A comparison between Tables 12.4-4 and 12.4-12 shows that Unit 2 occupational exposure is consistent with the data for operating plants for the period 1974-1979 (before the TMI accident). The higher occupational exposures for the period 1980-1982 are not expected at Unit 2 because the plant modifications that caused the increases have been incorporated into the original design of Unit 2.

12.4.2.2 Estimates of Inhalation Thyroid Doses

Inhalation doses during full-power operations will be negligible in every area except the reactor, turbine, and radwaste building areas. Potential airborne activities for these areas are given in Section 12.2.2. These concentrations are based upon data given in NUREG-0016 and EPRI-495. The inhalation thyroid doses that result are given in Table 12.4-2.

Thyroid dose rates in Table 12.4-2 are calculated according to:

$$D_t = \sum_i \quad (B.R.)\ (A_i)\ (C)\ (K_i)$$

Where:

D_t = thyroid dose rate (mRem/hr)

K_i = thyroid dose conversion factor (from TID 14844, March 1962) (Rem/Ci)

B.R. = breathing rate, m^3/sec

A_i = building airborne concentration, Table 12.2-15 of the ith isotope (uCi/cm^3)

C = conversion factor = (10^3 mRem/Rem) (3.6 x 10^3 sec/hr) (10^6 cm^3/m^3) (1 Ci/10^6 uCi)

12.4.3 Exposures at Locations Outside the Plant Structures

Radiation exposures at locations outside the plant arise from:

1. Direct and air-scatter (skyshine) contributions due to the presence of N-16 in the plant buildings.

2. Release of gaseous effluents from the plant.

3. Contributions due to the operation of Unit 1 and the James A. FitzPatrick plant.

Estimated doses at the restricted area boundary (RAB) directly north of Unit 2 are summarized in Table 12.4-1 and are based on 300 hr/yr expected occupancy and EPU conditions. These estimates meet the dose guidelines of 10CFR20 and 40CFR190.

12.4.3.1 N-16 Dose Contributions

Dose contributions due to N-16 are evaluated for both direct and skyshine components. Skyshine doses are due to air scattering of the high-energy gammas emitted by decaying N-16 present in reactor steam in the steam lines, turbines, and moisture separators. The layout of the turbine building walls and floors used in the dose evaluation is given on Figures 12.3-13 through 12.3-18. Data regarding the source term and the major shields providing skyshine shielding are as follows.

Source Term

The N-16 activity at the RPV nozzle is 50 uCi/gm. The N-16 activity in the turbine building equipment is 50 uCi/gm decayed for the transit time from the RPV to the entrance of each specific component.

Major Shielding

The shield walls surrounding the turbine operating floor are sufficiently thick to reduce direct radiation levels. Air-scattered radiation is reduced by enclosure of the most intense sources (i.e., the moisture separator reheaters and cross-around piping). This enclosure includes a 20-in thick,

horizontally-oriented concrete canopy. This concrete slab extends from 20 ft on either side of the turbine axis to the turbine building's outside walls. Additional reduction of radiation available for scattering in air over the building is provided by 3-in thick vertically oriented steel skirts that run the length of the turbine train at a distance of 20 ft from the axis. The outer wall on the north side of the building has 4 ft of concrete to reduce direct radiation levels on the lake.

The dose rate from the Unit 2 turbine building N-16 direct and air scatter contributions at the nearest point of the RAB is provided in Table 12.4-1.

12.4.3.2 Dose Contributions Due to the Loading of Solid Waste Liners

Dose contributions due to the loading of solid waste liners prior to shipment at Unit 2 are estimated using the expected solid waste activities and loading durations for which the liners are unshielded. For conservatism, the waste activities are not decayed for the solidification process or time spent in storage prior to loading the liners for shipment.

12.4.3.3 Dose Contributions from Unit 1 and the J. A. FitzPatrick Plant

Dose contributions due to the operation of Unit 1 and the J. A. FitzPatrick plant are estimated based upon readings from selected TLDs located near the shoreline of Lake Ontario, directly north of Unit 2.

12.4.3.4 Exposures Due to Airborne Activity

Dose rates resulting from airborne activity at the RAB, based on 300 hr/yr expected occupancy, are provided in Table 12.4-1.

12.4.4 Estimate of Doses to Construction Workers

The estimated doses to construction workers at Unit 2 due to the operation of Unit 1 and the J. A. FitzPatrick plant are shown in Table 12.4-3. This table presents the estimated average number of construction workers from the beginning through the end of construction and the estimated yearly doses to the construction workers at Unit 2 due to radiation from Unit 1 and J. A. FitzPatrick operations. The doses are estimated based upon readings from selected TLDs on the construction site and that an individual was in that specific location 43 hr per week, 50 wk per year (except in 1975 when 16 wk was assumed, and in 1987 when 26 wk was assumed). The estimates are, therefore, very conservative with respect to the doses that the construction workers actually received.

12.4.5 References

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.19, Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates, Revision 1, June 1979.

2. U.S. Nuclear Regulatory Commission, NUREG-0713, Occupational Radiation Exposure at Commercial Nuclear Power Reactor 1982, Volume 4, December 1983.

TABLE 12.4-1

ESTIMATED DOSES AT THE RESTRICTED AREA BOUNDARY

(mRem/yr)

Radiation Source	Dose Rate (mRem/yr)*
Unit 2	
N-16	7.47
Loading of solid waste liners	7.92
Airborne	1.839
Unit 1 and James A. FitzPatrick plant	4.04
TOTAL	21.27

* Estimated levels at the shoreline directly north of Unit 2.

TABLE 12.4-2

ESTIMATES OF INHALATION THYROID DOSE RATES
IN MAJOR BUILDINGS
(mRem/hr)

Building	Expected Dose Rate - Normal Power Operation	Expected Dose Rate - Reactor Shutdown
Turbine		
1. Main condenser area	3.9+00	2.5-01
2. Miscellaneous areas*	5.4-01	1.9-01
Radwaste		
1. Liquid waste handling areas		
a. Waste collector areas	5.9-01	1.8-01
b. Floor drain collector areas	2.7-01	8.1-02
2. Solid waste handling areas	8.1-02	2.1-02
Reactor Building	6.4-01	2.7-01

* Miscellaneous areas include the turbine operating floor, the feedwater pump area, the steam jet air ejector area, and the mechanical vacuum pump area.

TABLE 12.4-3

ESTIMATED ANNUAL DOSE TO UNIT 2 CONSTRUCTION WORKERS
DUE TO THE OPERATION OF UNIT 1 AND
THE JAMES A. FITZPATRICK PLANT

Year	Number of Workers	Estimated Annual Dose (manRem)
1975	160 (4 mo)	20*
1976	600	235*
1977	1600	712*
1978	1300	7
1979	1900	88
1980	710	29
1981	2000	43
1982	4200	88
1983	5500	110
1984	6475	142
1985	5650	46
1986	2573	4**
1987	773	3**,***

* Based on maximum possible exposure, high dose was due to radiation exposure from the waste storage tanks located in the northeast corner of Unit 1. The remaining are based on average exposures.

** Man-rem values are derived including quarterly variations of manpower and TLD doses.

*** For six months.

TABLE 12.4-4

ESTIMATED OCCUPATIONAL RADIATION DOSES FOR UNIT 2
BY WORK FUNCTIONS

Function	Annual Dose (man-Rem/yr)	Percentage of Total Dose
Routine operations and surveillance	104.0	18.0
Nonroutine operations and surveillance	32.0	6.0
Routine maintenance	182.0	32.0
Radwaste operations	60.0	11.0
Refueling	23.0	4.0
Inservice inspection	109.0	19.0
Special maintenance	55.0	10.0
Total	565.0	100.0

TABLE 12.4-5

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE OPERATIONS AND SURVEILLANCE

Activity	Avg Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Operations surveillance					
Reactor building	1.2	1.6	1	2/shift	4.3
Turbine building	15.0	1.7	1	2/shift	54.0
Chemistry surveillance	0.5	5.5	10	daily	10.0
Security surveillance	0.6	0.8	1	1/hr	4.4
Instrumentation and controls	0.4	6.0	10	1/day	8.8
Radiation protection surveillance	18.7	19.4	1	1/week	18.9
Fire protection surveillance	0.6	14.5	1	daily	3.2
				Total	104.0

TABLE 12.4-6

OCCUPATIONAL DOSE ESTIMATES DURING NONROUTINE OPERATIONS AND SURVEILLANCE

Activity	Avg. Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Equipment operations					
RWCU system	1.0	6.0	2	1/yr	0.01
Condensate system	2.2	8.0	2	1/day	13.00
RHS system	0.2	2.0	2	1/month	0.01
RCIC system	0.2	0.5	2	1/month	0.01
SLS system	1.0	3.0	2	1/month	0.07
Instrument calibration					
Instrumentation and controls	0.8	6.0	10	1/day	18.00
Radiation monitors					
Linearity checks	2.5	50.0	1	2/yr	0.25
Calibration	2.5	225.0	2	1/1.5 yr	0.75
				Total	32.0

TABLE 12.4-7

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE MAINTENANCE

Activity	Avg. Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Minor reactor building repairs	1.0	20	2	1/week	2.1
Ventilation and air conditioning	1.0	20	1	1/week	1.0
Control rod drive	75.0	7.5	21	1/1.5 yr	7.9
Recirculation pumps	100.0	20	5	2/1.5 yr	13.0
Recirculation system valves	100.0	2	2	1/yr	0.4
Reactor water cleanup pump	90.0	25	4	1/1.5 yr	6.0
Reactor water cleanup valves	90.0	8	2	1/yr	1.4
Condensate system	5.0	200	7	1/yr	7.0
Residual heat removal system	90.0	25	8	1/yr	18.0
Safety relief valves	100.0	60	5	1/1.5 yr	20.0
Main steam isolation valves	100.0	100	6	1/yr	60.0
Snubbers	20.0	100	5	1/yr	10.0
Minor turbine building repairs	0.5	8	1	1/day	1.5
Turbine overhaul	1.0	240	10	1/1.5 yr	1.6
Drain coolers	12.0	40	2	1/yr	0.96
Steam jet air ejectors	1.0	40	2	1/yr	0.08
Offgas system	20.0	40	2	6/yr	9.6
Miscellaneous radwaste pump repairs	20.0	40	2	6/yr	9.6
Miscellaneous radwaste valve repairs	20.0	40	2	6/yr	9.6
Filter and demineralizer	20.0	30	3	1/yr	1.8
Standby liquid control system	1.0	60	2	1/1.5 yr	0.08
Radiation monitors	1.0	12	2	1/month	0.29
				Total	182.0

TABLE 12.4-8

OCCUPATIONAL DOSE ESTIMATES DURING WASTE PROCESSING
(RADWASTE OPERATION)

Activity	Avg. Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Operation of liquid radwaste system	0.6	6.0	3	1/shift	12.0
Operation of solid radwaste system	0.7	6.0	3	1/shift	14.0
DAW compacting	3.5	6.0	2	2/day	31.0
Radwaste shipments	1.0	6.0	2	1/week	0.62
DAW shipments	5.0	16.0	2	1/month	1.9
				Total	60.0

TABLE 12.4-9

OCCUPATIONAL DOSE ESTIMATES DURING REFUELING

Activity	Avg. Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Reactor disassembly	12.0	75.0	10	1/1.5 year	6.0
Reactor assembly	12.0	150.0	10	1/1.5 year	12.0
Fuel unload	2.0	200.0	8	1/1.5 year	2.1
Fuel load	2.0	180.0	8	1/1.5 year	1.9
Fuel preparation	2.0	100.0	7	1/1.5 year	0.93
				Total	23.0

TABLE 12.4-10

OCCUPATIONAL DOSE ESTIMATES DURING IN-SERVICE INSPECTION

Activity	Avg. Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Reactor building primary containment	100.0	116	6	1/1.5 year	70.0
Reactor building secondary containment	25.0	465	4	1/1.5 year	31.0
Turbine and miscellaneous buildings	10.0	190	6	1/1.5 year	7.6
				Total	109.0

TABLE 12.4-11

OCCUPATIONAL DOSE ESTIMATES DURING SPECIAL MAINTENANCE

Activity	Avg. Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rem/yr)
Offgas system charcoal overhaul	100.0	100	2	1/20 year	1.0
Special maintenance reactor water cleanup system	90.0	100	4	1/10 year	3.6
Special maintenance spent fuel cooling	300.0	100	5	1/10 year	15.0
Miscellaneous RCIC repairs	9.0	140	4	1/10 year	0.5
CRD disposal	1.0	400	10	3/year	12.0
Radwaste flatbed filter maintenance	400.0	8	3	2/year	19.0
Sparger replacement	15.0	75	14	*	
Feedwater heater repair	1.0	80	10	1/10 year	0.08
Recirculation pump overhaul	30.0	225	5	1/10 year	3.4
				Total	55.0

* Should not be necessary.

TABLE 12.4-12

OPERATIONAL MAN-REM PER YEAR FOR SELECTED BWR PLANTS

	1974	1975	1976	1977	1978	1979	1980	1981	1982
Dresden 1, 2, 3	1,662	3,423	1,680	1,693	1,529	1,800	2,105	2,802	2,923
Monticello	349	1,353	263	1,000	375	157	531	1,004	993
Nine Mile Point	824	681	428	1,383	314	1,497	591	1,592	1,264
Peach Bottom 2 and 3	-	228	840	2,036	1,317	1,388	2,302	2,506	1,977
Quad Cities 1 and 2	482	1,618	1,651	1,031	1,618	2,158	4,838	3,146	3,757
Vermont Yankee	216	153	411	258	339	1,170	1,338	731	205
Pilgrim 1	-	798	2,648	3,142	1,327	1,015	3,626	1,836	1,539
Millstone Point 1	1,430	2,022	1,194	392	1,239	1,793	2,158	1,496	929
Oyster Creek	984	1,140	1,078	1,614	1,279	467	1,733	917	865
Brunswick 1 and 2	-	-	-	-	1,004	2,602	3,870	2,638	3,792
Browns Ferry 1, 2, and 3	-	-	-	-	1,792	1,667	1,825	2,380	2,220
FitzPatrick	-	-	-	1,080	909	859	2,040	1,425	1,190
Average Man-Rem/Unit	594	878	784	974	686	872	1,419	1,183	1,140

NOTES:

1. Overall average of 948 man-rem/yr per unit.
2. The information in this table is based on data in NUREG-0713, Volume 4.

TABLE 12.4-13

DISTRIBUTION OF ANNUAL MAN-REM BY
WORK FUNCTIONS BASED ON OPERATING BWR DATA

	Percentage					
Work Function	1978	1979	1980	1981	1982	Avg
Reactor operations and surveillance	12.3	13.4	7.6	7.5	9.1	10.0
Routine maintenance	43.2	39.3	42.8	42.2	33.7	40.2
Waste processing	5.8	4.3	3.1	11.0	6.2	6.1
Refueling	2.0	4.4	5.2	2.5	2.7	3.4
In-service inspection	2.6	7.3	3.3	3.7	4.3	4.2
Special maintenance	34.1	31.2	38.1	33.1	44.0	36.1

NOTE: The information in this table is extracted from data found in the following documents:

1. NUREG-0594, Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1978, November 1979.

2. NUREG-0713, Volume 1, Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1979, March 1981.

3. NUREG-0713, Volume 2, Occupational Radiation Exposure of Commercial Nuclear Power Reactors, 1980, December 1981.

4. NUREG-0713, Volume 3, Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1981, November 1982.

5. NUREG-0713, Volume 4, Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1982, December 1983.

12.5 RADIATION PROTECTION PROGRAM

12.5.1 Organization

12.5.1.1 Program Objectives

The Radiation Protection Program is designed to provide for the protection of all permanent and temporary personnel and all visitors from radiation and radioactive materials in a manner consistent with federal and state regulations during all phases of plant operation.

12.5.1.2 Personnel Experience and Qualifications

The Radiation Protection Program is developed and administered by the General Supervisor Radiation Protection. The General Supervisor Radiation Protection is responsible for the handling and monitoring of radioactive materials including source and by-product material. New fuel receipt, inspection, and initial storage into the fuel vault will be performed by maintenance and radiation protection personnel under the direction of the Manager Maintenance and the General Supervisor Radiation Protection. The responsibility for directing fuel movement will rest with reactor engineering personnel and radiologic monitoring with radiation protection personnel. The physical movement of fuel within the reactor and spent fuel pool will be performed by qualified personnel. The SM is responsible for all fuel movements within the Station. The experience and qualifications of all the above responsible individuals are given in Sections 13.1.2 and 13.1.3.

12.5.2 Equipment, Instrumentation, and Facilities

This section describes the radiation protection facilities provided at Unit 2 for measuring and maintaining personnel radiation exposure ALARA.

12.5.2.1 Location of Equipment, Instrumentation, and Facilities

12.5.2.1.1 Radiation Protection Office

The Radiation Protection Office, which is the center for directing the radiation protection technical staff and RWP coordination, is located in the access control building at el 261 ft. The office is equipped with desks and tables for work space and file cabinets and bookshelves for current survey data and reference material.

12.5.2.1.2 Counting Room

The counting room where radioactive samples are analyzed for radioisotopic content and activity level is located in the access control building linkway at el 261 ft. Counting room equipment is discussed in Section 12.5.2.2.1.

12.5.2.1.3 Chemistry and Radiochemistry Laboratory

The chemistry and radiochemistry laboratories where radioactive samples are chemically analyzed and/or prepared for radiochemical analysis are located in the Unit 1 turbine building at el 261 ft, and in the Unit 2 decontamination area. The laboratories are equipped with fume hoods with filters, an emergency shower, an eyewash, and miscellaneous chemistry laboratory equipment for chemical analysis. The laboratories support routine chemistry and radiochemistry analysis.

12.5.2.1.4 Personnel Decontamination Facility

A personnel decontamination facility is located in the turbine building at el 306 ft, and two additional facilities, one for women and one for men, are located in the auxiliary service building south at el 261 ft. These facilities are equipped with dressing areas, sinks, and showers.

12.5.2.1.5 Locker Rooms and Toilet Facilities

Locker rooms and toilet facilities for men and women are located in the maintenance building.

12.5.2.1.6 This section deleted.

12.5.2.1.7 Instrument Storage

The instrument storage room used for storage of portable survey instrumentation, respiratory equipment, and radiation protection supplies is located in the access control building linkway at el 261 ft.

12.5.2.1.8 Instrument Calibration and Repair Facility

Radiation-measuring instrumentation is normally calibrated and repaired at the instrument calibration facility located in the auxiliary service building south at el 261 ft.

12.5.2.1.9 Respiratory Equipment Repair, Assembly, and Testing

Respiratory equipment repair, assembly, and testing facilities are located at Unit 1. Respiratory equipment storage is discussed in Section 12.5.2.1.7.

12.5.2.1.10 Personnel Monitoring Stations

All personnel exiting the radiologically-controlled area are required to monitor/frisk themselves for contamination as directed by Station procedures. Monitor/frisk stations are provided as necessary.

In addition to local monitoring, a portal monitor is installed at the exit from the protected/restricted area to alarm in the case of contamination on person or clothing.

12.5.2.2 Radiation Protection Instrumentation

The following paragraphs describe the radiation protection instrumentation. Should new instrumentation become available or regulatory measuring and reporting requirements change, the new instrumentation may be purchased in lieu of that listed.

12.5.2.2.1 Counting Room Instrumentation

Radiation-measuring instrumentation located in the counting room is used by radiation protection personnel to analyze samples for radioactivity and/or radioisotopic content. Typical samples include contamination survey media, airborne survey media, and process samples.

Counting room instrumentation is listed in Table 12.5-1. The criteria for selection of these instruments are reliability, efficiency, and sensitivity to count and/or analyze sample media. Each instrument is checked and calibrated at routine intervals with standard radioactive sources traceable to a NIST source. Calibration of an instrument is also performed after repair and prior to use if the period since the last calibration is greater than the regular interval. Personnel qualified in radiation protection procedures perform instrument calibrations and check counting efficiency, background count rates, and high voltage settings according to Station procedures.

12.5.2.2.2 Portable Survey Instrumentation

Portable survey equipment allows Station personnel to perform surveys for direct radiation, airborne radioactivity, and surface contamination. Portable survey instrumentation is listed in Table 12.5-2.

The criteria for selection of portable survey equipment are based on accuracy, dependability, and simplicity of operation, calibration, and maintenance. The instrument can be easily serviced and can cover the entire spectrum of radiation measurements expected during normal or accident conditions.

Portable survey instruments used for establishing beta and gamma dose rates are calibrated at a frequency specified in Station procedures, with deviations not exceeding annually allowed based on documented instrument reliability. Instruments are also calibrated by personnel qualified in radiation protection procedures after repair and before use when the period since the last calibration is greater than the regular interval. All portable survey instruments are source checked prior to use to verify proper operation according to Station procedures.

12.5.2.2.3 Personnel Monitoring Instrumentation

Monitoring of personnel is accomplished by the use of TLDs, direct-reading dosimeters, and neutron TLD badges. Personnel entering the radiologically-controlled area (RCA) are issued TLDs and direct-reading dosimeters in accordance with 10CFR20 and Station procedures.

The TLD readings are normally used as the official record. The TLD has thermoluminescent chips and suitable filters to differentiate between penetrating and nonpenetrating radiations.

TLDs are normally processed quarterly or in accordance with Station procedures. If a TLD should be lost or damaged, an individual's exposure is estimated using appropriate methods and documented.

Direct-reading dosimeters are worn by plant personnel for a day-to-day or job-to-job estimate of personnel exposure. Direct-reading dosimeters are provided in various ranges and are calibrated at a frequency specified in Station procedures or when damage is suspected.

Table 12.5-3 lists personnel monitoring instrumentation with various ranges and sensitivities.

12.5.2.2.4 Radiation Protection Equipment

Portable air samplers are used to collect samples for determination of airborne radioactive material concentrations. Samples may be analyzed for radioactive particulate, radioiodine, and airborne gaseous activities. Portable air samplers are calibrated at a frequency specified in Station procedures.

To monitor a specific work area or field location, portable air samplers are used. Continuous air monitors with visual and audible alarms may also be used.

The ARMS is installed in areas where it is desirable to have continuous radiation level information. These monitors provide radiation level indication locally and in the control room. If radiation levels reach a preset level, an audible and visual alarm is provided locally and in the control room.

In areas where it is desirable to monitor dose rates and there is no area monitor in close proximity, a portable alarming dose rate instrument is used. If the radiation dose rate is high, the instrument provides a local audible and visual alarm. Table 12.5-4 lists other radiation protection instrumentation.

Respiratory protection equipment is available in the instrument storage room (Section 12.5.2.1.9).

12.5.3 Procedures

This section describes certain methods embodied in procedures and/or instructions to ensure that occupational radiation exposure is maintained ALARA. Strict adherence to plant radiation protection procedures and instructions ensures that the occupational radiation exposures are within the limits of 10CFR20 and ALARA concepts.

Procedures for personnel radiation protection are prepared consistent with the requirements of 10CFR20 and are approved, maintained, and adhered to for all operations involving personnel radiation exposure.

12.5.3.1 Radiation and Contamination Surveys

Radiation protection personnel conduct a routine radiation and contamination surveillance program of all areas within the Station. The frequency of the radiation surveys is dependent upon the specific location, frequency of use, and potential for change in radiological conditions. Radiation survey techniques are specified in the radiation protection procedures. The surveys include external radiation measurements, removable contamination tests and air sampling, as applicable. External radiation measurements may be performed for alpha, beta, gamma, and/or neutron radiation. Removable contamination tests are performed to establish beta-gamma contamination levels and may be processed for specific types of radiation (alpha, beta, gamma) or specific radionuclide (via gamma spectroscopy). Air samples are taken to establish airborne concentrations of particulates, noble gases, and/or radioiodine, and specific radionuclide information can be obtained. In addition to the above stated routine radiation and contamination surveys, specific surveys are performed before, during, and after each operational and maintenance function involving significant potential exposure of personnel to radiation and/or radioactive materials.

Radiation and contamination survey data are recorded and stored as permanent Station records. Current survey data are available in the Radiation Protection Office and pertinent data (radiation level, contamination level, and respiratory protection requirements) are posted as aids in keeping personnel exposures ALARA.

Airborne radioiodine concentration in the plant during an accident condition is determined by using an air sampler with a charcoal cartridge (silver zeolite if noble gas interference is suspected) which is analyzed in the counting room facilities.

Air sampling procedures are included in the technician training program.

12.5.3.2 Procedures and Methods to Maintain Exposures ALARA

ALARA considerations are incorporated into various types of plant procedures and instructions which cover activities involving exposure. Examples of procedures and methods used to maintain radiation exposure ALARA in the various operational categories are discussed in this section and the ALARA policy considerations are detailed in Section 12.1.

12.5.3.2.1 Refueling

Procedures and methods used to maintain radiation exposure ALARA during refueling outages include the following:

1. A RWP is used to provide positive radiological control over work in progress.

2. Training is conducted to familiarize workers with procedures and equipment to be used.

   a. Temporary shielding is used, when practicable, to reduce radiation exposure.

   b. Areas are conspicuously posted and maintained.

   c. Reactor status is monitored and alarms provided by the ARMS.

   d. Special tools are used where necessary and practical to minimize exposure.

3. Before removing the vessel head, the primary system is degassed and sampled to minimize expected airborne activity when the head is removed.

4. During movement of irradiated fuel assemblies, the active fuel water coverage is maintained at a minimum of 7 ft 7 1/2 in (approximately).

5. The refueling cavity water is filtered to reduce the activity in the water and to lower exposure rates.

6. Radiation levels in work areas are monitored and precautions taken as necessary, consistent with ALARA.

7. Filtered or exhaust ventilation is operated as appropriate to minimize airborne radioactive material.

12.5.3.2.2 In-service Inspection

Procedures and methods used during ISIs to maintain radiation exposure ALARA are as follows:

1. A RWP is used to provide positive radiological control over work in progress.

2. Training is conducted to familiarize workers with procedures, equipment, radiation and contamination levels, and protective clothing requirements appropriate to a particular job.

3. Insulation is designed, where practical, for ease of removal and replacement where removal is required for repetitive inspections.

4. Equipment is calibrated and checked for proper operation prior to entry into radiation area.

5. Temporary shielding is used, where practicable, to reduce radiation exposure.

6. Filtered or exhaust ventilation is provided, as necessary, to minimize airborne radioactive material.

12.5.3.2.3 Radwaste Handling

Procedures and methods used to maintain radiation exposure ALARA during radwaste handling include the following:

1. Handling of radwaste by personnel is minimized by plant design and local procedures.

    a. Training of personnel.

    b. Required RWP.

2. The volume of radwaste that requires handling has been reduced by plant design and operating practices.

3. Special tools are used when practical to minimize time spent close to the radioactive source.

4. An overhead crane is provided to move drums or liners as required.

5. Adequate storage has been provided to minimize multiple handling of drums.

6. The labeling of drums or liners is completed prior to filling.

7. Filtered or exhaust ventilation is provided to minimize the airborne radioactive material from waste handling operations.

12.5.3.2.4 Spent Fuel Handling, Loading, and Shipping

Procedures and methods used during spent fuel handling, loading, and shipping to maintain radiation exposure ALARA include the following:

1. A RWP is used to provide positive radiological controls over work in progress.

2. Training is conducted to familiarize workers with procedures and equipment required to complete assignments.

3. Movement of irradiated fuel assemblies and loading into shipping casks is accomplished with the active fuel maintained under at least 7 ft 7 1/2 in (approximately) of water.

4. Fuel handling cranes and extension tools are used to handle shipping and storage casks, fuel assemblies, and inserts.

5. The spent fuel pool water is filtered to reduce the radiation exposure of personnel in the area.

6. The spent fuel pool water is cooled and surface air ventilation is provided to minimize the airborne radioactive material.

7. After a shipping cask is loaded with spent fuel and sealed, its exterior surfaces are decontaminated using a pressurized water washing device prior to shipment.

12.5.3.2.5 Normal Operation

Procedures and methods used during normal operation to maintain radiation exposure ALARA include the following:

1. By Station design and construction, significant radiation sources are minimized and shielded.

2. The RWP system is used to provide positive radiological control over work in progress.

3. Training and retraining in radiation protection is provided to persons assigned tasks in controlled areas.

4. Controlled areas are conspicuously posted and maintained in accordance with 10CFR20.

5. Special access control procedures are used for entry to areas where significant exposure might be received.

6. Protective clothing and equipment are provided.

7. An ARMS is installed and provides indication of radiation levels with local alarms, where appropriate.

8. The ventilation system is designed to minimize the spread of airborne contamination.

9. Shielding effectiveness is verified by an initial startup survey for gamma and neutron radiation.

12.5.3.2.6 Routine and Nonroutine Maintenance (Repair)

Examples of procedures and methods used during routine maintenance to maintain personnel radiation exposure ALARA are as follows:

1. Maintenance work involving systems that collect, store, contain, or transport radioactive materials must be covered by an approved RWP.

2. Training is provided, as required, to accomplish assigned tasks in controlled areas.

3. Maintenance procedures incorporate appropriate precautions and radiological considerations.

4. Equipment is moved to areas with lower radiation and contamination levels for maintenance when practical.

5. Special tools are used when practical.

6. Portable shielding is used when practical.

7. Periodic monitoring by radiation protection technicians, as radiological conditions warrant, is provided.

8. Post-job debriefings are utilized for high-exposure jobs to obtain input from personnel actually performing the work. This is utilized in revising procedures and instructions as appropriate for ALARA considerations.

12.5.3.2.7 Sampling

Procedures and methods of maintaining personnel radiation exposure ALARA during sampling include the following:

1. Sampling of radioactive systems may be performed from inside sample hoods or locally.

2. Procedures specify the protective clothing and sampling techniques to be used.

3. Radiation levels of the sample container and of the work area during sampling are monitored.

4. Personnel are trained in the proper handling of sample containers after samples are collected.

5. Personnel are trained in the proper storage and disposal of radioactive samples.

12.5.3.2.8 Calibration

Procedures and methods of maintaining radiation exposure of personnel ALARA during calibration include the following:

1. Detailed procedures are followed for each calibrator use.

2. The instrument calibrator is properly shielded.

3. An interlock is provided to prevent opening the calibrator door while the source is exposed.

4. Portable sources used to calibrate fixed instruments are transported and maintained in shielded containers.

5. The RWP system is used to provide positive radiological controls over calibration.

6. Where possible fixed instruments are situated in low radiation areas so that necessary test signals can be inserted with the instrument in place.

12.5.3.3 Control of Personnel Radiation Exposure

Occupational radiation exposures are kept ALARA by a combination of shielding, access control, contamination control, radiation protective equipment, and radiation protection policies. As plant operation progresses, if it is found that shielding in given locations is insufficient or that the level of contamination is unacceptable, measures will be taken to reduce radiation levels by providing additional shielding, decreasing the amount of contamination, or limiting personnel access.

12.5.3.3.1 Access Control

Access control can be accomplished by a locked barrier such as a door or gate or a posted guard. Access control within the fenced area of the site for radiological purposes is determined by the radiation level, contamination level, or the presence of radioactive materials. Areas are posted as required by 10CFR20. Posted areas include the following:

1. Radiologically-controlled area - radioactive materials.

2. Radiation areas.

3. High radiation areas.

4. Airborne radioactivity areas.

Access to the RCA is through an established control point. Minimum conditions for entry into this area include:

1. Personnel monitoring equipment when monitoring is required by 10CFR20 or Station procedures.

2. Training as required by 10CFR19.

3. No personnel with open wounds, except as specifically inspected and authorized by radiation protection personnel.

4. Other requirements as posted at the entrance or imposed by radiation protection personnel.

Positive control is exercised over entries into high radiation areas as specified by 10CFR20 or the Technical Specifications.

12.5.3.3.2 Contamination Control

Controlling the spread of removable contamination is accomplished by routine surveillance and the enforcement of contamination limits for personnel, equipment, and surface areas. Key control points and step-off-pads are surveyed for removable contamination by radiation protection personnel as scheduled by the General Supervisor Radiation Protection. In addition to the routine radiation and contamination survey (Section 12.5.3.1), special surveys are performed in areas whenever a change in contamination levels is likely.

Any areas found contaminated beyond the limits established in radiation protection procedures are isolated (with ropes, barriers, etc.), posted, and decontaminated as soon as practical. Areas where complete removal of surface contamination is not practical are designated as contaminated areas. The level of contamination within these areas is maintained as low as practical, and step-off-pad procedures are used to help contain the contamination.

Tools and equipment are surveyed by radiation protection personnel for removable and fixed contamination prior to release from the RCA. Contamination limits for unconditional release are established in radiation protection procedures. If tools or equipment do not meet these limits, they must be decontaminated and resurveyed. Some tools and equipment are for use only in the RCA. These are surveyed periodically by radiation protection personnel.

Personnel are responsible for surveying themselves for contamination in accordance with approved Station procedures. If contamination is found, the individual notifies radiation protection personnel and is decontaminated in accordance with approved plant procedures. As a safeguard, all personnel must

pass through portal radiation monitors before leaving the security building protected area. If contamination is found, a security guard notifies radiation protection personnel and the individual will be directed to the decontamination facilities.

12.5.3.3.3 Personnel Protective Equipment

Procedures have been established for use, inspection, cleaning, and maintenance of all personnel protective equipment. The use of protective clothing, such as coveralls, shoe covers, head covers, gloves, plastic suits, and safety-related items, is described and demonstrated as part of the Radiation Protection Training Program (Section 12.5.3.6). In addition to classroom training, individual instruction is provided on how to properly don and remove protective clothing during the training program on a mock-up facility.

12.5.3.3.4 Respiratory Protection

As part of the continuous respiratory protection program, continuous air monitors monitor building ventilation systems and low volume air samplers are strategically located and monitor appropriate building areas. Upon indication of increasing airborne radioactivity by the above methods, or if work is planned in a high contamination area, a high volume air sample is obtained.

The high volume air sampler is equipped with a particulate filter. An iodine cartridge with a certified retention efficiency is added when the location to be sampled is suspected of containing iodine. After completion of the air sample, the sampling media is analyzed for radionuclide concentration. The analysis results are evaluated against the derived air concentration (DAC) limits, set in 10CFR20 Appendix B, to obtain the percent of DAC present.

When airborne radioactivity is DAC detected at 30 percent of the limits set in 10CFR20 Appendix B, the area is isolated and posted as an Airborne Radioactivity Area, and access is controlled. A RWP which specifies the respiratory protection required is needed for entry. Air sampling and analysis is performed by radiation protection personnel to determine that the respiratory protection equipment is appropriate for working conditions.

Special emphasis is placed on the proper use of respiratory equipment in the Radiation Protection Training Program (Section 12.5.3.6). The necessity for respiratory protective equipment is carefully explained to personnel so that each individual will fully understand and consciously use respiratory equipment properly. Respiratory equipment is outlined in Table 12.5-4.

Routine medical examinations are given to specifically identify individuals who may have a medical problem with respect to the use of respiratory protective equipment. These individuals are

restricted from using various types of respiratory devices, as appropriate. In addition, a routine test is performed to assure proper selection of respiratory devices. Fit testing is accomplished by a challenge gas and takes into account special problems such as unusual facial features, head movements, and facial movements when talking. Radiation protection procedures prohibit anything that inhibits a good seal of face masks such as eyeglasses and facial hair.

12.5.3.3.5 Bioassay and Whole-Body Counting

Bioassay and/or whole-body counting of permanent Station personnel is performed on a schedule determined by the work environment of each individual. Personnel who have had extensive use of respiratory equipment in contaminated areas or who have had potential exposure to airborne radioactivity are monitored by the radiation protection staff and are given whole-body counts. Follow-up bioassays and/or whole-body counting are performed at threshold levels required by procedure or when deemed necessary by the General Supervisor Radiation Protection.

Bioassay and/or whole-body counting of a contractor or temporary employee is performed for personnel working in applicable areas prior to initial Station entry, final exit from the Station, and on a schedule determined by the individual's work environment. Instruction to workers and notifications and reports to individuals will be accomplished in accordance with 10CFR19.

12.5.3.3.6 Post-operation Review

The Supervisor Radiological Engineering, under the supervision of the General Supervisor Radiation Protection, is responsible for review of requested and completed RWPs and man-Rem reports to identify those activities that would benefit from an ALARA review. The Supervisor Radiological Engineering is responsible for reviewing specific activities and investigating a means for reducing occupational radiation exposure for that activity.

12.5.3.3.7 Personnel Dosimetry

Plant employees, contractors, and visitors wear a film badge, and/or a TLD, and/or a personal dosimeter when in the RCA when monitoring is required by 10CFR20 or Station procedures. Only those individuals who have successfully completed the Radiation Protection Training Program (Section 12.5.3.6) are authorized to enter these areas unescorted. Those individuals who do not successfully complete the Radiation Protection Training Program must, at all times, be escorted within the RCA. In any case, personnel will be given training commensurate with the radiological complexity involved.

In addition to the above personal dosimetry devices, neutron-sensitive badges are issued to those individuals who may

be exposed to neutron radiation. The RWP will specify the need for neutron dosimetry.

TLDs are processed quarterly. The TLD serves as the official exposure record of personnel. Personal dosimeters are read daily by each individual who has entered the RCA to provide a daily estimate of exposure.

TLDs are processed more frequently when an individual's exposure status is in doubt. Radiation protection procedures address dosimetry collection, processing, recording, and maintenance of records.

Exposure data of all personnel is recorded on Form NRC-5, Current Occupational External Radiation Exposure, or the equivalent. Occupational exposures incurred by individuals prior to working at Nine Mile Point are summarized on Form

NRC-4, Occupational External Radiation Exposure History, or the equivalent. These records are maintained under the supervision of the General Supervisor Radiation Protection at the plant and are retained indefinitely, or until the Nuclear Regulatory Commission (NRC) staff authorizes their disposal.

12.5.3.4 Controlling Potential Airborne Radioactivity Concentrations

Confinement or containment of loose radioactive particles is the primary method of reducing airborne radioactive contamination. Specific methods such as temporary tents, filtered air streams, keeping objects wet, and other methods, have been carefully considered in operating, maintenance, and repair procedures. Radiation protection procedures have been developed for sampling air volumes before, during, and after work procedures to determine the airborne radioactivity levels so that appropriate actions can be taken. Criteria are established, based on DACs and whole body exposure rates, specifying the level of contamination mandating the use of protective clothing and respiratory equipment, as well as the type of equipment that must be used.

12.5.3.5 Radiation Work Permit

Authorization for performing a maintenance or operational function in the Station RCA requires a RWP for that function. The purpose of the RWP is to help control exposure of plant personnel to radiological hazards, thus maintaining occupational radiation exposure ALARA. The RWP provides the means to:

1. Control access to radiologically hazardous areas.

2. Restrict the spread of radioactive contamination.

3. Help prevent unexpected radiological hazards.

4. Assure that proper radiological procedures and precautions are followed.

5. Provide a history of how work was completed and exposure received by personnel which can be used for review in efforts to reduce exposures.

6. Integrate exposures of personnel to ensure no one exceeds the limits of 10CFR20.

RWPs are issued by qualified radiation protection personnel under the supervision of the General Supervisor Radiation Protection.

Adherence to RWP requirements ensures that work within posted areas is performed in a radiologically safe manner. Adherence to the permit is mandatory. Radiation protection procedures have been established for the issuance of RWPs, and the following information is contained on the RWP.

1. Location and type of job.

2. Exposure limits of personnel involved.

3. Radiological conditions in the work area (radiation, contamination, airborne radioactivity, etc.).

4. Dosimetry requirements.

5. Protective clothing and equipment requirements.

6. Radiation protection monitoring requirements.

7. Special instructions and ALARA considerations, such as warnings of significant contamination levels, activities which could change dose rates, ALARA measures from the Radiation Protection Technician, work hold points, directions for work sequence, work prerequisites, and/or requirements of an ALARA review.

When a RWP is issued, the job contact or supervisor of the working group will review the RWP requirements with the worker.

12.5.3.6 Radiation Protection Training Program

All plant personnel (permanent or temporary), visitors, and contractors are trained in the fundamentals of radiation protection and must qualify in radiation protection prior to entering the RCA unescorted. Permanent plant employees are required to attend a refresher training program annually. Those personnel whose job duties do not require entrance to the RCA are trained to recognize radiation and contamination signs and are made aware of the reasons for not entering such areas.

Plant personnel (permanent or temporary) and contractors who enter areas where respiratory equipment is required are trained and qualified in respiratory protection prior to entering the area. These personnel are also required to attend a retraining program annually.

The Radiation Protection Training Program is essentially a summary of radiation protection procedures and policies. The formal training sessions cover the following subject areas:

1. Review of the plant ALARA policy.

2. Review of radiation protection procedures.

3. Proper use of dosimetry devices including action on lost or damaged devices.

4. Radioactive waste volume reduction policy.

5. Area postings (Radiation, High Radiation, Airborne, Contamination, and Radioactive Materials).

6. Use of respiratory equipment and proper fitting techniques.

7. Use of RWPs.

8. Occupational radiation exposure limits established in 10CFR20, exposure guides set by plant administration, and instructions to female personnel on prenatal exposure as established by 10CFR20 and further discussed in RG 8.13.

9. Emergency Plan and Procedures.

10. Practical experience with individual critique on the proper donning and removal of protective clothing at a mock-up facility.

All personnel who attend the Radiation Protection Training Program must pass a written examination in order to be qualified in radiation protection. All examination questions used in the Radiation Protection Training Program shall be approved by the Manager Radiation Protection or his designee. All personnel shall understand how radiation protection relates to their job and shall have reasonable opportunities to discuss radiation protection safety with radiation protection personnel.

12.5.3.7 Radioactive Materials Safety Program

Radiation protection procedures are established for the safe handling, storage, and inventory of sealed and unsealed radionuclide sources having activities greater than the

quantities of radionuclides defined in 10CFR20 and 10CFR30. These procedures include:

1. Monitoring of shipments containing radioactive materials.
2. Inventory and monitoring for surface contamination of sources shall be done every 6 months.
3. Labeling of each source with the radiation symbol, stating the activity, isotope, and source identification number.
4. Storage in a locked area of each source that is not installed in an instrument or other piece of equipment.
5. Maintenance of records on the results of the inventories, leakage tests, use, location, condition, principal user, and the receipt and final disposition dates for all sources.

Section 3.7.4 of the TRM describes additional requirements for sealed sources.

Radioactive sources that are subject to the materials controls described herein are used or handled only by or under the direction of personnel qualified in radiation protection procedures.

12.5.3.8 Compliance with Regulatory Guides

Sections 12.5.1 and 13.1.3 pertain to compliance with RG 1.8.

An assessment of compliance with RG 8.2 is described in this section, 12.5.3.

Information pertaining to compliance with RG 8.4 and 8.14 is described in Sections 12.5.2.2.3 and 12.5.3.3.7.

As described in this section and Section 12.1, many of the recommendations set forth in RG 8.8 and 8.10 have been incorporated in policy and procedures.

Section 12.5.3 and Exhibit 12.1-2 provide an assessment of compliance with RG 8.7, 8.9 and 8.26.

Section 12.5.3.6 describes information pertinent to compliance with RG 8.27 and 8.29.

TABLE 12.5-1

COUNTING ROOM INSTRUMENTATION

Qty	Instrument	Efficiency (%)	Range	Remarks
2	Gamma spectrometer	15	0-2 MeV	Computer-based pulse height analysis with GeLi, HpGe, ReGe detectors
1	Gas flow proportional counter or scintillation counter	45 (35 for scintillation counter)	0-10^5 counts	Contamination levels on survey samples
3	Beta sample counter	10	0-10^5 counts	Contamination levels on survey sample

TABLE 12.5-2

PORTABLE RADIOLOGICAL SURVEY INSTRUMENTATION

Qty	Instrument	Range	Accuracy (%)	Remarks
15	Ion chamber survey meter	0-50 R/hr	±10	Dose rate meter
15	Ion chamber survey meter	0-5 R/hr	±10	Dose rate meter
5	GM survey meter	0-1,000 R/hr	±10	Telescoping probe
20	GM survey meter	0-50,000 cpm	±10	Pancake probe portable count instrument
5	Beta sample counter	0-100,000 counts	±10	Equivalent to BC-4
1	Neutron survey	0-5,000 mRem/hr	±10	BF_3 tube within cadium-loaded polyethylene sphere
2	Alpha survey meter	0-500,000 cpm	±10	Alpha count rate meter

TABLE 12.5-3

PERSONNEL MONITORING INSTRUMENTATION

Quantity*	Instrument	Range	Sensitivity	Remarks
As required	Film dosimetry badge	20-100,000 mR	10 mR	If Used
As required	TLD	10-200,000 mR	10 mR	
As required	Direct-reading dosimeter	0-500 mR	10 mR	Direct-reading ion chamber
25	Direct-reading dosimeter	0-1 R	50 mR	Direct-reading ion chamber
25	Direct-reading dosimeter	0-5 R	200 mR	Direct-reading ion chamber
10	Direct-reading dosimeter	0-50 R	2 R	Direct-reading ion chamber
1	Whole-body counter	0-several body burdens	1% of most nuclide body burdens	To determine internal exposure
10	Dosimeter chargers			
As required	Electronic dosimeter	0-999 R	10%	Solid state or GM dosimeter
As required	Electronic dosimeter reader	NA	NA	

* Quantity is minimum required.

TABLE 12.5-4

RADIATION PROTECTION EQUIPMENT

Quantity*	Instrument	Range	Accuracy	Remarks
6	Gamma area monitors	1-1,000 MR/hr	± 10	Present audio and visual alarm
3	Air sampler	1-2 cfm	± 10	Low volume
5	Air sampler	1-6 cfm	± 10	High volume
1	Whole-body counter	MCA**		Minimal detectable activity
As required	Half-face mechanical filter respirator	NA	NA	--
As required	Full-face mechanical filter respirator	NA	NA	--
As required	Full-face air line respirator	NA	NA	--
As required	Full-face self-contained breathing apparatus	NA	NA	--

*	Quantity is minimum required.
**	Gross gamma counting acceptable for passive whole body monitoring.