TABLE OF CONTENTS

Section	Title	Page
12.0	RADIATION PROTECTION	12.1-1
12.1	ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA)	12.1-1
12.1.1	POLICY CONSIDERATIONS	12.1-1
12.1.2	DESIGN CONSIDERATIONS	12.1-2
12.1.2.1	Design Features	12.1-2
12.1.2.2	Utilization of Experience Gained from Operating Facilities to Ensure that Radiation Doses are ALARA	12.1-4
12.1.2.3	Design Guidance Given to Individual Designers	12.1-5
12.1.2.4	Design Review Procedures	12.1-6
12.1.2.5	Decommissioning Design Considerations	12.1-6
12.1.3	OPERATIONAL CONSIDERATIONS	12.1-7
12.1.3.1	ALARA Training Program	12.1-8
12.1.3.2	Radiation Zoning and Access Control	12.1-11
12.2	RADIATION SOURCES	12.2-1
12.2.1	CONTAINED SOURCES	12.2-1
12.2.1.1	Source Terms	12.2-1
12.2.1.2	Reactor Building	12.2-2
12.2.1.3	Auxiliary Building	12.2-7
12.2.1.4	Intermediate Building	12.2-8
12.2.1.5	Turbine Building	12.2-9
12.2.1.6	Radwaste Building	12.2-9
12.2.1.7	Offgas Building	12.2-10
12.2.1.8	Sources Resulting From Design Basis Accidents	12.2-11
12.2.2	AIRBORNE RADIOACTIVE MATERIAL SOURCES	12.2-11
12.2.2.1	Reactor Building (Outside of Drywell)	12.2-11
12.2.2.2	Radwaste Building	12.2-13
12.2.2.3	Turbine Building	12.2-13
12.2.2.4	Fuel Handling Area of the Intermediate Building	12.2-13
12.2.2.5	Other Buildings	12.2-14
12.2.3	REFERENCES FOR SECTION 12.2	12.2-14

TABLE OF CONTENTS (Continued)

Section	Title	Page
12.3	RADIATION PROTECTION DESIGN FEATURES	12.3-1
12.3.1	FACILITY DESIGN FEATURES	12.3-1
12.3.1.1	Equipment and Facility Design Features	12.3-1
12.3.1.2	Illustrative Examples of Plant Design Features to Minimize Occupational Doses	12.3-5
12.3.2	SHIELDING	12.3-8
12.3.2.1	Design Objectives	12.3-8
12.3.2.2	Design Description	12.3-8
12.3.3	VENTILATION	12.3-19
12.3.3.1	Design Bases	12.3-19
12.3.4	AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION	12.3-24
12.3.4.1	Area Radiation Monitoring	12.3-25
12.3.4.2	Airborne Radioactivity Monitoring	12.3-36
12.3.4.3	Detection of MPC (DAC) Levels of Airborne Radioactivity	12.3-54
12.3.4.4	System Setpoints	12.3-67
12.3.5	REFERENCES FOR SECTION 12.3	12.3-68
12.4	DOSE ASSESSMENT	12.4-1
12.4.1	ESTIMATES OF PERSONNEL OCCUPANCY REQUIREMENTS	12.4-1
12.4.2	ESTIMATES OF ANNUAL PERSON-REM DOSES	12.4-2
12.4.3	ESTIMATED INHALATION DOSES	12.4-3
12.4.4	ESTIMATED ANNUAL DOSE OUTSIDE THE NUCLEAR FACILITY AT THE BOUNDARY OF THE RESTRICTED AREA	12.4-5
12.4.4.1	Skyshine and Direct Dose from Turbines	12.4-5
12.4.4.2	Direct Doses From Stored Radwaste	12.4-9
12.4.4.3	Direct Doses From the External Surfaces of Buildings	12.4-9
12.4.4.4	Doses From Gaseous Radioactive Plume	12.4-9
12.4.5	REFERENCES FOR SECTION 12.4	12.4-10

TABLE OF CONTENTS (Continued)

Section	Title	Page
12.5	RADIATION PROTECTION PROGRAM	12.5-1
12.5.1	ORGANIZATION	12.5-1
12.5.2	EQUIPMENT, INSTRUMENTATION AND FACILITIES	12.5-2
12.5.2.1	Facilities	12.5-2
12.5.2.2	Access and Egress to Radiation Protection Controlled Areas	12.5-4
12.5.2.3	Health Physics Instrumentation	12.5-5
12.5.3	HEALTH PHYSICS INSTRUCTIONS	12.5-9
12.5.3.1	Radiation and Contamination Surveys	12.5-9
12.5.3.2	Procedures and Methods Ensuring ALARA	12.5-10
12.5.3.3	Controlling Access and Stay Time	12.5-13
12.5.3.4	Contamination Control	12.5-14
12.5.3.5	Training Programs	12.5-15
12.5.3.6	Personnel Dosimetry	12.5-17
12.5.3.7	Evaluation and Control of Potential Airborne Radioactivity	12.5-19
12.5.3.8	Radioactive Material Handling and Storage Methods	12.5-20
12.6	DESIGN REVIEW OF PLANT SHIELDING FOR SPACES/ SYSTEMS WHICH MAY BE USED IN POSTACCIDENT OPERATIONS OUTSIDE CONTAINMENT	12.6-1
12.6.1	INTRODUCTION	12.6-1
12.6.2	RADIOACTIVE SOURCE RELEASE	12.6-2
12.6.3	RADIOACTIVE SOURCE DISTRIBUTION	12.6-3
12.6.4	SYSTEMS CONTAINING RADIOACTIVE SOURCES	12.6-6
12.6.4.1	LPCS, HPCS, RHR, (LPCI mode), RCIC Systems	12.6-6
12.6.4.2	RHR (Shutdown Cooling Mode)	12.6-7
12.6.4.3	RHR (Suppression Pool Cooling Mode)	12.6-7
12.6.4.4	RWCU	12.6-7
12.6.4.5	Liquid Radwaste System	12.6-7
12.6.4.6	Sampling System	12.6-8
12.6.4.7	Offgas System	12.6-8
12.6.4.8	Annulus Exhaust Gas Treatment System (AEGTS)	12.6-8
12.6.4.9	Feedwater Leakage Control System	12.6-8
12.6.5	SHIELDING REVIEW	12.6-8a

TABLE OF CONTENTS (Continued)

Section	Title	Page
12.6.6	AREAS REQUIRING PERSONNEL ACCESS	12.6-9
12.6.7	POSTACCIDENT RADIATION ZONE DRAWINGS AND SUMMARY	12.6-11
12.6.7.1	Radiation Dose Rates as a Function of Time Following an Accident	12.6-11
12.6.8	REFERENCES FOR SECTION 12.6	12.6-11

LIST OF TABLES

Table	Title	Page
12.2-1	Basic Reactor Data	12.2-15
12.2-2	Core Boundary Neutron Fluxes	12.2-18
12.2-3	Gamma Ray Source Energy Spectra	12.2-20
12.2-4	Gamma Ray and Neutron Fluxes Outside the Vessel Wall	12.2-22
12.2-5	Radiation Shielding Source Terms	12.2-24
12.2-6	Material Composition of the TIP Detectors and Cables	12.2-28
12.2-7	Radiation Levels From the TIP Detector and Cables	12.2-29
12.2-8	Traversing Incore Probe Detector Decay Gamma Activities of Materials in the Detector	12.2-30
12.2-9	Decay Gamma Activities of Materials in the Cable	12.2-31
12.2-10	Typical Turbine Component N-16 Inventories	12.2-32
12.2-11	Parameters and Assumptions Used in Calculating Reactor Building Airborne Activity	12.2-34
12.2-12	Reactor Building Airborne Activity	12.2-35
12.2-13	Radwaste Building Airborne Activity	12.2-36
12.2-14	Turbine Building Airborne Activity	12.2-37
12.2-15	Fuel Handling Area Airborne Activity	12.2-38
12.3-1	(Deleted)	12.3-69
12.3-2	Radiation Shield Thicknesses	12.3-70
12.3.3	Comparison of Non-ESF Charcoal Filter Systems to Regulatory Guide 1.140 Criteria	12.3-73
12.3-4	Detectors Associated with Unit 1	12.3-77
12.3-5	(Deleted)	12.3-79

LIST OF TABLES (Continued)

Table	Title	Page
12.3-6	Detectors Associated with Common Areas	12.3-81
12.3-7	Detectors Associated with Local Channels	12.3-82
12.3-8	Detector Design Requirements	12.3-83
12.3-9	Readout Module Design Requirements	12.3-84
12.3-10	Airborne Radiation Monitor Subgroup Unit 1 and Common	12.3-85
12.3-11	Isokinetic Probes	12.3-87
12.3-12	Reactor Building Subcompartment Ventilation Data M11, M14 Systems	12.3-88
12.3-13	Radwaste Building Subcompartment Ventilation Data M31 System	12.3-89
12.3-14	Auxiliary Building Subcompartment Ventilation Data M38 System	12.3-91
12.3-15	Intermediate Building Subcompartment Ventilation Data M33 System	12.3-92
12.3-16	Fuel Handling Area Subcompartment Ventilation Data M40 System	12.3-93
12.3-17	Heater Bay Subcompartment Ventilation Data M41 System	12.3-94
12.3-18	Turbine Building Subcompartment Ventilation Data M35 System	12.3-95
12.3-19	Offgas Building Ventilation Exhaust Subcompartment Ventilation Data M36 System	12.3-96
12.4-1	Estimated Manpower Needs and Occupancy Requirements, Unit 1 and Unit 2	12.4-12
12.4-2	Person-rem Estimates for Normal Plant Operations, Anticipated Operational Occurrences and Routine Maintenance, Unit 1 and Unit 2	12.4-13
12.4-3	Boiling Water Reactors Percentages of Exposure by Job Function	12.4-14

LIST OF TABLES (Continued)

Table	Title	Page
12.4-4	Boiling Water Reactors Percentages of Exposure by Work Function	12.4-15
12.4-5	Yearly Operational Person-rem for Selected BWR Plants	12.4-16
12.4-6	Summary of Total Occupational Radiation Exposure Estimates by Task	12.4-17
12.4-7	Occupational Dose Estimates During Routine Operations and Surveillance	12.4-18
12.4-8	Occupational Dose Estimates During Routine Maintenance	12.4-20
12.4-9	Occupational Dose Estimates During Waste Processing	12.4-21
12.4-10	Occupational Dose Estimates During Refueling	12.4-22
12.4-11	Occupational Dose Estimates During Inservice Inspection	12.4-23
12.4-12	Occupational Dose Estimates During Special Maintenance	12.4-24
12.4-13	Personnel Radiation Doses From Airborne Activity	12.4-25
12.4-14	Safety/Relief Valve Dischargers Dose for Type 2 Event	12.4-26
12.4-15	Estimated Skyshine Doses	12.4-27
12.4-16	Work Force by Occupational Crafts	12.4-28
12.4-17	Summary of Direct Doses	12.4-29
12.4-18	Dose to Non-Plant Personnel	12.4-30
12.4-19	Direct Dose from the External Surfaces of Buildings	12.4-32
12.4-20	Dose from Gaseous Radioactive Plume	12.4-33
12.5-1	Laboratory Equipment	12.5-21

LIST OF TABLES (Continued)

Table	Title	Page
12.5-2	Portable Survey Instruments	12.5-22
12.5-3	Personnel Monitoring Instruments	12.5-23
12.5-4	Radiation Protection Equipment	12.5-24
12.6-1	Occupancy and Radiation Design Objectives	12.6-12
12.6-2	Initial Core Inventory	12.6-13
12.6-3	Initial Radioactive Source Terms (Gammas/cc-sec)	12.6-17
12.6-4	Dose Rates	12.6-18

12.0 RADIATION PROTECTION

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

12.1.1 POLICY CONSIDERATIONS

The management of the FirstEnergy Nuclear Operating Company (FENOC) recognizes its responsibility and authority to operate and maintain the Perry Nuclear Power Plant in a manner that provides for the safety of plant personnel and the public. In accordance with the regulations, the company will maintain the policy of keeping radiation exposure as low as reasonably achievable (ALARA).

It is the intent of the ALARA Program to demonstrate that reasonable measures have been taken to maintain the radiation exposure of plant personnel and members of the public as far below the regulatory limits as reasonably obtainable.

The Manager, Radiation Protection is responsible for the radiation health and safety of PNPP. To implement his responsibilities, the Manager, Radiation Protection ensures: the necessary supervisory and technical support is available to provide radiation protection program oversight for monitoring radiological work activities during plant operation, maintenance, and refueling; the planning of the radiological work activities is accomplished to minimize worker doses; and the necessary actions are taken to reduce radiation sources within the plant ensuring occupational radiation doses are maintained ALARA during all radiological work activities.

The Director, Site Engineering, has the responsibility to assure that all design work includes appropriate ALARA considerations for installation, operation and maintenance of each design package.

The General Plant Manager, and the Director, Performance Improvement Department have the complete responsibility for all onsite activities in connection with the safe and efficient operation and maintenance of the PNPP. They are responsible for managing the affairs of the plant to ensure reliable, efficient and safe operation.

The General Plant Manager, the Director, Site Engineering, and the Radiation Protection Manager work in concert to implement the Company's ALARA policy. However, it is the responsibility of all supervision to enforce the requirements for keeping radiation doses ALARA, and the responsibility of each individual to comply with these requirements.

12.1.2 DESIGN CONSIDERATIONS

12.1.2.1 Design Features

Ensuring that occupational radiation doses are ALARA was begun during the early design of PNPP. By familiarizing design engineers with ALARA concepts, and by providing review of the design by radiation protection personnel, the operation of the PNPP will result in personnel doses that are ALARA and will fulfill the intent of <Regulatory Guide 8.8>, <Regulatory Guide 8.10>, and <10 CFR 20>.

The following design criteria have been, and will continue to be, adhered to:

a. Access labyrinths are provided for rooms housing equipment that contain high radiation sources to preclude a direct radiation path from the equipment to accessible areas.

b. Piping penetrations, ducts and voids in radiation shield walls are located to preclude the possibility of streaming from a high to low radiation area or otherwise will be adequately shielded.

c. Shielding discontinuities caused by shield plugs, concrete hatch covers and shield doors to high radiation areas are provided with offsets to reduce radiation streaming.

d. Radioactive piping is routed through high radiation areas where practicable, or in shielded pipe chases in low radiation areas.

e. Sufficient work area and clearance space is provided around equipment to permit ease of servicing.

f. Instruments requiring in situ calibration are not normally located in high radiation areas.

g. Non-radioactive equipment which requires servicing is not normally located in proximity with potentially radioactive equipment.

h. Spread of contamination from radioactive spillage is minimized by providing a floor drain system which collects and routes the liquid to the liquid waste processing system for proper handling. Decontamination of an area is facilitated by use of materials and coatings which lend themselves to cleaning by standard methods.

i. Natural traps which could be potential pockets for corrosion product activity are minimized in pipe and ducts by avoiding sharp bends, rough finishes and cracks.

j. Shielding is provided for equipment which is anticipated to be normally radioactive. The dose levels are designed not to exceed <10 CFR 20> requirements under the worst operating conditions of the plant.

k. Temporary shielding, such as lead blankets, is available on the site in case it is ever needed.

l. Remote handling of radioactive materials is provided wherever it is needed and practicable.

m. Process piping that may contain radioactive fluids is routed and dimensioned on piping system drawings, thereby minimizing any field run radioactive piping.

n. Redundant components for the radwaste processing system have been supplied to increase flexibility in plant operations and decrease radiation doses during maintenance.

o. Most radioactive components can be flushed to decrease radiation levels and subsequent personnel doses from maintenance.

p. The radwaste handling system is designed to the extent practicable to be remotely operated and have remote-manual overrides in case a failure occurs.

12.1.2.2 Utilization of Experience Gained from Operating Facilities to Ensure that Radiation Doses are ALARA

An important aspect in the design of a nuclear facility is the feedback information obtained from plants currently operating. For this design feedback, information has been obtained directly from operating facilities and from governmental and industrial publications. The following areas of information represent the type of feedback useful in plant design:

a. Operational radiation levels.

b. Trends in radiation levels associated with years of operation based on plant type, plant size, power levels, and plant design.

c. Radiation zones as determined by occupancy requirements and actual radiation levels.

d. Location of components with respect to plant operability.

e. Reliability of components.

f. Adequacy of plant layout in terms of traffic patterns, and space allocation, such as around radioactive components requiring maintenance, inspections and pipe routing.

g. Number of plant employees associated with different tasks and the resulting person-rem doses.

Feedback information is used in the design of the plant to identify potential problem areas. Such problems are reviewed with regard to current design and, where applicable, modifications made to the design to eliminate or decrease the potential for such problems to arise.

12.1.2.3 Design Guidance Given to Individual Designers

The following design guidance has been given to individual designers of systems associated with radiation protection:

a. Follow guidance given in <Regulatory Guide 8.8> and <Regulatory Guide 8.10>.

b. Use maximum realistic source strengths in all calculations.

c. Consider all sources that significantly contribute to the dose at a particular point.

d. Place instrument readouts and items needing routine surveillance or attention in as low as practicable radiation zone.

e. Make the design sensitive to the expected procedures of plant personnel under normal operation and anticipated occurrences.

12.1.2.4 Design Review Procedures

Independent design reviews are performed by competent specialists to assess the implementation of the design criteria and further ensure that the final design is compatible with maintaining occupational radiation doses ALARA. All designs influencing radiological control in the plant are reviewed by competent professionals in the area of radiation protection.

As indicated in <Section 12.1.2.3>, personnel associated with design aspects influencing radiation protection have been given the basic ALARA principles. After the preliminary design and layout of the system, a shielding engineer analyzes the various radiation sources and specifies shielding as necessary to conform to the appropriate radiation zone requirements. Radiation protection personnel review the system in terms of the total plant operation and specify necessary changes to keep occupational radiation doses ALARA. Radiation protection personnel are part of the overall design team and report their findings directly to the design project management for the PNPP. These findings are considered in conjunction with design requirements from disciplines not directly associated with radiation control to determine what modifications, if any, should be made to promote ALARA radiation doses.

12.1.2.5 Decommissioning Design Considerations

The radiation protection aspects of the plant design are described in <Section 12.1.2> and <Section 12.3.1>. While these features assure that the plant can be operated and maintained with ALARA exposures, they will

also aid in the ALARA aspects of decommissioning. These include shielding, accessibility criteria, equipment separation, decontamination features, and system design considerations.

In addition, a detailed study was performed for the plant that designates the required removal paths for major equipment located within structures of the nuclear island and secondary plant. The report contains the following information.

a. A list of path numbers by building and floor elevation including minimum path size and floor loading (PSF - live load) for each designated path.

b. Identification of major equipment for each structure on a floor-by-floor basis including total equipment and component weights, as applicable.

c. Where removal requirements were defined, required route numbers are indicated for each piece of equipment. Where elevators are used in the removal scheme, elevations are indicated for loading and unloading points.

d. Plant layout drawings are also provided showing the major equipment and removal routes for each floor elevation.

12.1.3 OPERATIONAL CONSIDERATIONS

The site ALARA Committee is comprised of plant staff personnel representing the various disciplines, such as Radiation Protection, Operations, Maintenance, and Nuclear Engineering. The site ALARA Committee is responsible for the overall coordination of the Station ALARA Program and for advising site management in matters relating to ALARA in accordance with <Regulatory Guide 8.8>.

Responsibilities of the site ALARA Committee shall include:

a. Ensure that a program is established with a management philosophy to maintain occupational exposures ALARA, with specific goals and objectives for implementation,

b. Ensure 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;

c. Ensure that the measurement system results are reviewed on a periodic basis and that corrective actions are taken when attainment of the specific objectives appears to be jeopardized;

d. Ensure that the authority for providing procedures and practices by which the specific goals and objectives will be achieved is delegated; and

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

12.1.3.1 ALARA Training Program

The Radiation Worker Training (RWT) at PNPP will help implement the Company's ALARA policy in accordance with <10 CFR 20>. RWT will help

workers understand how radiation protection relates to their jobs and all workers will have frequent opportunities to discuss radiation safety with the Radiation Protection Section personnel when the need arises.

A Radiation Work Permit (RWP) will be initiated for work activities based on the radiation, removable contamination and/or airborne activity levels to be encountered and by the area of the plant where the work will take place. An RWP is also required for work which involves open radioactive systems or when required by Radiation Protection.

The Radiation Work Permit will help implement the Company's ALARA policy in accordance with <10 CFR 20> by defining the radiological hazards and requiring specific radiological precautions. The RWP also becomes a record of how various jobs were performed and the radiological problems associated with specific jobs. By reviewing expired RWP's, recommendations can be made to change procedures or equipment that will result in lower radiation exposures in the future.

Training and RWP requirements will help ensure that the Company's ALARA policy is fulfilled. Some techniques that may be used are:

a. Temporary shielding may be used. Temporary shielding will be used only if total exposure, which includes installation and removal of the shielding, will be effectively reduced.

b. Prior to performing maintenance work, consideration will be given to flushing and/or chemically decontaminating in order to reduce crud levels and personnel exposure.

c. Dry run training will be used for jobs with exceptional radiological problems to familiarize personnel with the work they must perform at the job site. These techniques will assist in

improving efficiency and minimize the amount of time spent in radiation areas. These efforts will be documented to improve future efforts.

d. As much as practicable, work will be performed outside radiation areas. This includes items such as reading instruction manuals or procedures, adjusting tools or jigs, repairing valve internals, and prefabricating components.

e. For long term repair jobs, consideration will be given to establishing remote observation stations to assist supervising personnel in monitoring work progress from a lower radiation area.

f. On some jobs, special tools or jigs will be used so that work will be performed more efficiently to reduce errors, thus minimizing the time spent in a radiation area. Special tools may be used to increase the distance from a radiation source to the worker, thereby reducing the exposure.

g. Entry and exit control points will be established in areas with low levels of radiation to limit the exposure of personnel donning protective equipment or generally preparing to work in such areas. The access control points will be designed to minimize the spread of contamination from the work areas.

h. Protective clothing and respiratory protection equipment will be selected to minimize the discomfort of workers and to minimize the Total Effective Dose Equivalent (TEDE) of the workers. Efficiency will increase and less time will be spent in radiation areas.

i. Personnel will be assigned self reading dosimeters to estimate exposure during a work assignment.

j. On jobs where general area radiation levels are high, radiation protection coverage may be required during the period of work.

k. On intricate jobs, especially those which involve high radiation levels, preplanning will include estimation of the person-rem needed to complete the job. At the completion of the work, a debriefing session may be held with the personnel that performed the work (when practical) in an effort to determine how the work could have been completed more efficiently and with less radiation exposure.

12.1.3.2 Radiation Zoning and Access Control

During normal full power operation, the design maximum whole body dose rates within the plant that might be received by operating personnel, contractors and authorized visitors will depend upon the following zone designations:

Zone	Designation	Dose Rate (mrem/hr)
I	Unlimited Occupancy	≤0.5
II	Normal Continuous Occupancy	≤2.5
III	Controlled, Limited Access (4hr/wk)	≤25
IV	Controlled, Limited Access (1hr/wk)	≤100
V	Controlled, Limited Access (<1 hr/wk)	>100

12.1.3.2.1 Zone I

Zone I areas can be occupied by station personnel or visitors on an unlimited time basis. Health hazards due to ionizing radiation in Zone I are absolutely minimal. Access control to Zone I areas is due to security considerations.

12.1.3.2.2 Zone II

Design allows Zone II areas to have a higher dose rate than Zone I areas but still allows normal, continuous occupancy. Areas where radiation levels exist such that a major portion of the body could receive in any one hour a dose of 0.5 mrem or more.

12.1.3.2.3 Zone III

Zone III areas are designed for a maximum dose rate of 25 mrem/hr and for limited access of less than or equal to 4 hr/wk. Areas in which there exists radiation at such levels that a major portion of the whole body could receive in one hour a dose in excess of 5 mrem at 30 cm from the radiation source or from any surface that the radiation penetrates, will be designated as Radiation Areas. Access to a Zone III, Zone IV or a Zone V area will require personnel monitoring and a Radiation Work Permit.

12.1.3.2.4 Zone IV

Zone IV areas are designed for limited access of less than or equal to 1 hr/wk and a maximum dose rate of 100 mrem in 1 hour at 30 cm from the radiation source or from any surface that the radiation penetrates. Zone IV areas will be designated as Radiation Areas as described in <Section 12.1.3.2.3> above.

12.1.3.2.5 Zone V

Zone V areas are designated as High Radiation Areas (greater than 100 mrem in 1 hour at 30 cm from the Radiation Source or from any surface that the radiation penetrates). Access shall

not occur unless absolutely necessary. High Radiation Areas with general area dose rates greater than 1,000 mrem/hr or Very High Radiation Area where radiation levels could result in an individual receiving an absorbed dose in excess of 500 RADs in 1 hour at 1 meter from a radiation source or from any surface that the radiation penetrates, will be controlled in accordance with Perry Technical Specification 5.7.

12.1.3.2.6 General Guidelines for Maintenance in High Radiation Areas

a. Work will be performed in accordance with Regulatory Guide 8.8, Information Relevant To Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable and in accordance with the applicable Radiation Protection Nuclear Operating Procedures.

(DELETED)

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12.2 RADIATION SOURCES

12.2.1 CONTAINED SOURCES

12.2.1.1 Source Terms

With the exception of the vessel and drywell shields, shielding designs are based on fission product and activation product sources consistent with <Section 11.1>. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience supports a lower annual average than the design average (Reference 1). It should be noted that activation products, principally Nitrogen-16, control 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 Nitrogen-16 include: the condenser offgas system downstream of the steam jet air ejector, liquid and solid radwaste equipment, portions of the reactor water cleanup system, and portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

For application, the design sources are grouped first by location and then by equipment type (e.g., reactor building, core sources). The following paragraphs represent the source data in various pieces of equipment throughout the plant. General locations of equipment are shown in the general plant arrangement drawings of <Section 1.2> and <Section 12.3>.

12.2.1.2 Reactor Building

12.2.1.2.1 Radiation Sources from the Reactor Core

The information in this section defines a reactor vessel model and the associated gamma and neutron radiation sources. This section is designed to provide the data required for calculations beyond the vessel. The data selected were not chosen for any given program, but were chosen to provide information for any of several shield program types. In addition to the source data, calculated radiation dose levels are provided at locations surrounding the vessel. This data is given as a potential check point for calculations by shield designers.

12.2.1.2.1.1 Physical Data

<Table 12.2-1> presents the physical data required to form the model in <Figure 12.2-1>. This model was selected to contain as few separate regions as possible to adequately portray the reactor. <Table 12.2-1> provides nominal dimensions and material volume fractions for each boundary and region in the reactor model. To describe the reactor core, <Table 12.2-1> provides thermal power, power density, core dimensions, core average material volume fractions, and reactor power distributions. The reactor power distributions are given for both radial and axial distributions. This data contains uncertainties in the volume regions near the edge of the core. The level of uncertainties for these regions is estimated at 20 percent.

12.2.1.2.1.2 Core Boundary Neutron Fluxes

<Table 12.2-2> presents peak axial neutron multigroup fluxes at the core equivalent radius. The core equivalent radius is a hypothetical boundary enclosing an area equal to the area of the fuel bundles and the coolant space between them. The peak axial flux occurs adjacent to the

portion of the core with the greatest power. As shown by the data in <Table 12.2-1>, this point is below the core mid-plane. Since this data is calculated with a core equivalent radius, the flux represents a mean flux in the azimuthal angle around the core. While the flux within any given energy group is not known within a factor of 2, the total calculated core boundary flux is estimated to be within ±50 percent.

12.2.1.2.1.3 Gamma Ray Source Energy Spectra

Reactor core gamma ray source energy spectra are as follows:

a. Core Spectrum

<Table 12.2-3> presents average gamma ray energy spectra per watt of reactor power in both core and non-core regions. In <Table 12.2-3>, Item A, the energy spectrum in the core is presented. The energy spectrum in the core represents the average gamma ray energy released by energy group per watt of core thermal power. The energy spectrum in MeV per second per watt can be used with the total core power and power distributions to obtain the source in any part of the core.

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

b. Post-Operation Gamma Ray Energy Spectrum

<Table 12.2-3>, Item B, gives a gamma ray energy spectrum in MeV per second per watt in spent fuel as a function of time after operation. The data was prepared from tables of fission product

decay gamma fitted to integral measurements for operation times of 10^8 seconds or approximately 3.2 years. To obtain shutdown sources in the core, the gamma ray energy spectrum is combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectrum and the thermal power that the element contained during operation.

c. Non-Core Gamma Ray Energy Spectra

In <Table 12.2-3>, Item C, the gamma ray energy spectra in the cylindrical regions of the reactor from the core through the vessel are given. The energy spectra are given in terms of MeV per sec-watt-cm^3 at the inside surface and outside surfaces of the region. This energy spectrum multiplied by the core thermal power is the gamma ray source. The point on the inside surface of the region is the maximum point within the region. In the radial direction, the variation in source intensity may be approximated by an exponential fit to the data on the inside and outside surfaces of the region. The axial variation in a region can be estimated by using the core axial variation. The uncertainty in the gamma ray energy spectra is due primarily to the uncertainty in the neutron flux in these regions. The uncertainty in the neutron flux is estimated to vary from approximately ±50 percent at the core boundary to a factor of ±3 at the outside of the vessel. The calculations were carried out with voids beyond the vessel. The presence of shield materials beyond the vessel will cause an increase in the gamma source on the outside of the vessel.

12.2.1.2.1.4 Gamma Ray and Neutron Fluxes Outside Vessel

<Table 12.2-4> presents the maximum axial neutron and gamma ray fluxes outside the vessel. The maximum axial flux occurs on the vessel opposite the elevation of the core with the maximum power level. This

elevation can be located using the data from <Table 12.2-1>. The fluxes at this elevation are based on a mean radius core and do not show azimuthal angle variations. The calculational model for these fluxes assumed no shield materials beyond the vessel wall. The presence of shield materials will significantly alter the neutron fluxes in the lower end of the neutron energy spectrum. The gamma ray calculations include gamma ray sources from all of the cylindrical regions between the center of the core and the edge of the vessel. While the uncertainties in a given energy group flux may be a factor of ±3, the uncertainties in the total integral flux are estimated to be within a factor of 2.

12.2.1.2.2 Radioactive Sources in the Reactor Water, Steam and Offgas

The radioactive sources in the reactor water, steam and offgas are discussed in <Section 11.1>. This material provides the concentrations during normal operation of the radioisotopes in the reactor vessel or leaving the reactor vessel.

12.2.1.2.3 Reactor Water Cleanup System

The radioactive sources in the reactor water cleanup system 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 non-regenerative heat exchangers, pumps, valves, filter demineralizers, and the backwash receiving tank. The system is described in <Section 5.4.8>. The accumulated sources in the filter demineralizers, backwash receiving tanks and heat exchangers are given in <Table 12.2-5>.

The source is present in the filters and receiving tank during all modes of operation. Therefore, backwashing capability is provided to remove the residual activity for effective radwaste handling.

12.2.1.2.4 Main Steam System

All radioactive materials in the main steam system result from radioactive sources carried over from the reactor during plant operation. In most of the components carrying live steam, the source is dominated by N-16. In components where the N-16 has decayed, the other activities carried by the steam become significant. During plant shutdown, there is a residual activity resulting from prior plant operations.

12.2.1.2.5 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in <Table 12.2-3> in terms of MeV per second per watt. The design calculation is carried out for a mean element for an appropriate time after shutdown.

12.2.1.2.6 Other Radioactive Sources

Additional radioactive sources in the reactor building are:

a. Traversing Incore Probe (TIP)

The TIP system gamma detector and its drive cable become radiation sources following activation by neutrons in the reactor. The level of radiation sources depend upon the material composition of the components, the irradiation history and the decay time. The material composition of the gamma TIP is shown in <Table 12.2-6>. The radiation levels from the detector and the cable are shown in <Table 12.2-7> for a range of decay times after the TIP is retracted from the reactor vessel.

b. Reactor Startup Sources

The reactor startup sources were shipped to the site in a special cask designed for shielding. The sources were transferred under water while in the cask and were then loaded into the reactor while remaining under water. The sources will be removed after the initial operating cycle and will be stored in the fuel storage pools for later handling and removal. The removal of sources will be handled in accordance with approved procedures and specified shielding requirements.

12.2.1.3 Auxiliary Building

12.2.1.3.1 Radioactive Sources in the RHR, ADHR and RCIC Systems

The basic sources in the safeguard systems are the result of the radioactive materials in the reactor water or steam being transported to the system. The design basis sources for this equipment assume the total activity is the concentration of reactor water or steam decayed for the appropriate time interval times the total volume of steam or water in the equipment.

The gamma source strengths in the residual heat removal (RHR), alternate decay heat removal (ADHR) and reactor core isolation cooling (RCIC) systems were evaluated for the shutdown cooling mode of operation.

The sources in the RHR and RCIC systems are the Nitrogen-16 activities in the volumes of reactor steam contained in these systems. These sources are provided in <Table 12.2-5>.

The design gamma source strengths from fission products in the engineered safeguard systems following shutdown is typified by the source strength and fission product inventories for the system given in <Table 12.2-5>. In the shutdown mode the RHR system recirculates

reactor coolant to remove decay heat. The system is operated from approximately 2-4 hours after shutdown until the end of the refueling period. The source in the RHR system is the activity in the volume of reactor water contained in the system. This includes the increase of activity as a result of depressurization.

The system includes three RHR pumps and two heat exchangers. The highest radiation levels during reactor shutdown occur at the heat exchangers during the cooldown period. At other times or in other modes of operation, except hot standby, the sources are considerably decreased.

The ADHR system may be operated as early as 24 hours after shutdown. The source in the ADHR system is the activity in the volume of reactor water contained in the system. The design basis source term for shielding is based on plate out on the heat exchanger surfaces and the associated ADHR and P11 piping. The system includes one pump and one heat exchanger.

Source strengths of equivalent concentration in downstream piping are conservative for use in layout and shielding design of pipe chases.

12.2.1.4 Intermediate Building

12.2.1.4.1 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in <Table 12.2-3> in terms of MeV per second per watt.

12.2.1.4.2 Fuel Pool Cooling and Cleanup System

Sources in the fuel pool cooling and cleanup (FPCC) system are a result of transfer of radioactive isotopes from the reactor coolant into the spent fuel pool during refueling operations.

The FPCC system removes fission products and activated corrosion products as well as decay heat from the spent fuel pool. This system consists of surge tanks, pumps, heat exchangers, and precoat filter demineralizers. The activities in the FPCC components are given in <Table 12.2-5>.

12.2.1.5 Turbine Building

12.2.1.5.1 Turbine Building Sources

The radioactive sources in the turbine building are the result of the carryover of radioisotopes in the steam. These isotopes are distributed throughout the equipment in the turbine building. During power operation of the equipment, the most significant radioisotope source is N-16. Both the concentration and the energy per disintegration of N-16 contribute to the importance of this isotope during operation. Fission product sources are not important except in equipment where the steam or condensate passage time is sufficiently long to permit the N-16 to decay or where the physical processes preferentially separate the fission product isotopes.

Turbine component N-16 inventories are listed in <Table 12.2-10>.

12.2.1.5.2 Turbine Power Complex Sources

The radioactive sources in the turbine power complex are primarily fission products and activated corrosion products carried over from the steam. These sources are contained in condensate lines, filters and demineralizers and associated tanks. Activities in the condensate system are listed in <Table 12.2-5>.

12.2.1.6 Radwaste Building

The source of activity which enters the radwaste system is the activity in the reactor coolant including activation and fission products. The sources used for shielding radwaste system components are listed in <Table 12.2-5>.

In general, the maximum activity possible, even though remote, has been used in most pieces of equipment. For example, in the liquid radwaste

system reactor coolant has been used in each tank of the collector subsystems because of the interconnections of these tanks. In normal practice, undiluted reactor water would not be present in the subsystem.

12.2.1.7 Offgas Building

Radioactive sources in the offgas treatment system result from the decay of the noble gases and radioisotopes carried with the noncondensible gases. Radioactive sources entering the system have been defined at the design basis of an annual average release of 100,000 μCi/sec after a 30 minute holdup. An expected noble gas delay time of one minute was used from the vessel nozzle to the exit of the steam jet air ejector. It was assumed that the solid fission product isotopes were washed out in the main condenser. The N-16 activity at this point is primarily the result of the driving steam to the air ejector. Partitioning of other coolant activities is assumed to be like the N-16 with the distribution in the main condenser equivalent to the mass flows of the steam.

The radioactive sources in the gas treatment system are a function of the sources entering the system, the operational mode of the equipment piece and the residence time of the gas in the equipment. As a guide to the sources (other than N-16) in the gas treatment system, the licensing topical report NEDO-10734 "A General Justification for Classification of Effluent Treatment System Equipment as Group D" was used. Since this report was prepared as a conservative justification for a more conventional classification of the equipment, the conservative sources used in the report represent source levels which would be reached only after long term operation with design basis release rates.

The sources contained in the components of the offgas system are listed in <Table 12.2-5>.

12.2.1.8 Sources Resulting From Design Basis Accidents

The radiation sources from design basis accidents are discussed in <Chapter 15.0>.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

The airborne radioactivity sources that contribute to the plant effluent releases through the radioactive waste management system and the plant ventilation system are described in <Section 11.3>. The primary sources of airborne radionuclides are leakage of reactor coolant and main steam. Contributions to the airborne radionuclide concentrations in plant buildings due to leakage from the radioactive waste management system are small compared to the contribution from primary coolant.

12.2.2.1 Reactor Building (Outside of Drywell)

The main sources of activity leakage to the reactor building atmosphere during normal operation are conservatively estimated to be 2,000 lbs/hr of steam leakage from the safety/relief valves and 34,000 lb/month of steam from the RCIC turbine exhaust. The path for the resultant activity to reach the containment atmosphere is through the suppression pool. Other sources of leakage can be identified in the RWCU equipment vaults but these are normally inaccessible areas. The halogen and noble gas concentrations in the steam are taken from <Section 11.1>. The reactor building free volume is assumed to be 1.165×10^6 ft^3. <Table 12.2-11> lists other parameters used in the analysis.

Various scenarios can be assumed for the operation of an intermittent purge system including varying the frequency and the duration of purge. During full power plant operation, the purge system will be used to control potential airborne radioactive materials based on actual plant radiation monitor readings and radiation protection surveys. The need for purge will be a function of required accessibility, extent of fuel

cladding failure and leakage of reactor coolant and main steam. Analyses have been performed based on a typical scenario to determine the maximum expected peak airborne concentrations in the reactor building during normal power operation. The specific case evaluated assumes a 12 hour buildup period followed by a 4 hour purge. The results are presented in <Table 12.2-12>.

During refueling it is anticipated that the only major contribution to airborne activity in the reactor building will be from radioiodides. The fuel pool cooling and cleanup system is designed to clean and purify the water in the spent fuel pool and the upper fuel pools in the containment. The iodine activity in the pools will be reduced by passing the water through a 1,000 gpm filter demineralizer. The resultant airborne concentrations of iodine in the reactor building are expected to be less than 2 percent of the equilibrium values during normal operation. For the purposes of calculating operating exposures in <Section 12.4>, a value of 2 percent of the normal operation thyroid dose rate is assumed.

Transfer of the steam dryer and steam separator during refueling are performed with both the dryer and the separator partially out of water. This creates a potential source of airborne contamination. Administrative controls will be established to minimize exposure to airborne radioactive materials while performing the transfer.

Another source of potential airborne contamination in the reactor building is the activity release through relief valve discharge to the suppression pool. These are classified as Type 1 and Type 2 events. Type 1 events are of minor consequences because of the relatively short duration of the blowdown (≤15 seconds). Type 2 events are of more concern because they involve isolation and depressurization of the system. The expected frequency of the Type 2 events is 2.5 times per year. (Reference 2) provides the source terms used to determine the

containment airborne concentrations following a Type 2 event and the methodology used to determine operational doses following the event. <Section 12.4.3> presents the anticipated operator exposures per event.

12.2.2.2 Radwaste Building

Leakage to the radwaste building is assumed to be 2,000 gallons per day at 10 percent of the primary coolant iodine activity. The airborne noble gas activity in the radwaste building is negligible. A partition factor of .001 is assumed for iodine. The radwaste building free volume is 1.1×10^6 ft^3 and the purge rate is 30,000 cfm.

<Table 12.2-13> presents the calculated airborne concentrations in the radwaste building.

12.2.2.3 Turbine Building

Leakage to the turbine building atmosphere is assumed to be 1,700 lb/hr of steam at main steam activity. A partition factor of 1 is assumed for both noble gases and halogens. The turbine building free volume is assumed to be 3.2×10^6 ft^3 and the purge rate is 1.8×10^5 cfm.

<Table 12.2-14> presents the calculated airborne concentrations in the turbine building.

12.2.2.4 Fuel Handling Area of the Intermediate Building

Leakage to the fuel handling area atmosphere is based on evaporation from the spent fuel pool. The evaporation rate is calculated to be 320 lb/hr assuming the building is at 90°F and 50 percent relative humidity, and the pool is at 120°F. The equilibrium I-131 concentration in the pool was conservatively taken at 1×10^{-6} μCi/cc based on information given in (Reference 3). The building free volume is assumed to be 1.5×10^6 ft^3 and the purge rate is 30,000 cfm.

<Table 12.2-15> presents the calculated airborne concentration for I-131 and proportional values for I-133 and I-135.

12.2.2.5 Other Buildings

Other plant buildings are expected to have negligible noble gas and iodine airborne activity concentrations.

12.2.3 REFERENCES FOR SECTION 12.2

1. Smith, J. M., "Noble Gas Experience in Boiling Water Reactors," Paper No. A-54, presented at Noble Gases Symposium, Las Vegas, Nevada, September 24, 1974.

2. General Electric Co., "Mark III Containment Dose Reduction Study," 22A5718 Rev. 2, Jan. 29, 1980.

3. Johnson, A. B., "Behavior of Spent Nuclear Fuel in Water Pool Storage," BNWL-2256, Batelle, Pacific Northwest Laboratories, September 1977.

TABLE 12.2-1

BASIC REACTOR DATA

(Data used in the Original Design of the Nuclear Island Shields)

A.	Reactor thermal power, MW	3,579
B.	Average power density, watts/cm^3	54.07
C.	Physical dimensions[1]	

		Radii (in)
1.	Core equivalent	92.58
2.	Inside shroud	99.90
3.	Outside shroud	101.90
4.	Inside vessel (nominal)	119.0
5.	Outside vessel (nominal)	125.0
6.	Outside vessel (reinforced - nominal)	125.75
7.	Shroud head inside	192.0
8.	Vessel top head inside	119.0
9.	Vessel bottom head inside	130.19

		Distance (in)
10.	Outside of vessel bottom head	-7.75
11.	Inside of vessel bottom head	-0.25
12.	Vessel bottom head tangent	129.94
13.	Bottom of core support plate	202.56
14.	Top of core support plate	204.56
15.	Bottom of active fuel	213.50
16.	Top of reinforced vessel wall	210.00
17.	Top of active fuel	363.5
18.	Bottom of top guide	371.31

TABLE 12.2-1 (Continued)

		Distance (in)
19.	Top of fuel channel	377.87
20.	Shroud head tangent	424.23
21.	Inside of shroud head	452.27
22.	Outside of shroud head	454.27
23.	Normal vessel water level	566.6
24.	Top of steam dryer	720.63
25.	Vessel top head tangent	727.0
26.	Inside of vessel top head	846.0
27.	Outside of vessel top head	849.0

D. Material Densities, gm/cm^3

Region[2]	Coolant	UO_2	Zircaloy	340 Stainless
A	0.740	0.0	0.0	0.178
B	0.338	0.0	0.0	4.349
C	0.318	2.3	0.978	0.056
C-1	0.597	0.0	0.166	1.697
C-2	0.234	0.0	1.099	0.255
D	0.240	0.0	1.004	1.209
E	0.390	0.0	0.0	0.0
F	0.669	0.0	0.0	0.200
G	0.036	0.0	0.0	0.0
H	0.740	0.0	0.0	0.0
I	0.740	0.0	0.0	0.26

TABLE 12.2-1 (Continued)

E. Typical Core Power Distributions

Radial Power Distribution		Axial Power Distribution (Typical end-of-life)	
Percent of Equivalent Radius	Relative Power	Distance[3] (in)	Relative Power
0	1.2	-75	0.343
20	1.2	-68	0.755
35	1.19	-60	1.055
50	1.17	-48	1.190
60	1.15	-36	1.200
70	1.12	-24	1.190
80	1.05	-12	1.170
85	0.995	0	1.155
90	0.778	12	1.140
92.5	0.590	24	1.105
95.0	0.430	36	1.055
97.0	0.375	48	0.945
98.0	0.395	60	0.715
99.0	0.432	68	0.462
100.0	0.518	75	0.212

NOTES:

(1) Relative locations of dimensions are shown in <Figure 12.2-1>.
(2) Region locations are shown in <Figure 12.2-1>.
(3) Distances are measured from the mid-plane of the core.

TABLE 12.2-2

CORE BOUNDARY NEUTRON FLUXES

(Data used in the Original Design of the Containment Shields)

Energy Bounds		Neutron (neutrons/cm^2 - sec)
16.5	MeV	3.9 E+10
10.00	MeV	5.5 E+11
6.065	MeV	2.0 E+12
3.679	MeV	3.8 E+12
2.231	MeV	4.4 E+12
1.353	MeV	3.9 E+12
820.8	KeV	3.8 E+12
497.9	KeV	2.6 E+12
302.0	KeV	2.3 E+12
183.2	KeV	3.2 E+12
67.38	KeV	2.2 E+12
24.79	KeV	2.2 E+12
9.119	KeV	2.0 E+12
3.355	KeV	2.0 E+12
1.234	KeV	1.9 E+12
454.0	eV	2.0 E+12
167.0	eV	1.9 E+12
61.44	eV	1.8 E+12
22.60	eV	8.8 E+11
13.71	eV	8.8 E+11
8.315	eV	8.2 E+11
5.043	eV	8.4 E+11
3.059	eV	8.3 E+11

TABLE 12.2-2 (Continued)

Energy Bounds	Neutron (neutrons/cm^2 - sec)
1.855 eV	8.2 E+11
1.125 eV	8.8 E+11
0.616 eV	3.2 E+13
0.00 eV	

TABLE 12.2-3

GAMMA RAY SOURCE ENERGY SPECTRA

(Data used in the Original Design of the Containment Shields)

A. Gamma ray sources in the core during operation

Energy Bounds (MeV)	Gamma Ray Source (MeV/sec-watt)
16.5	8.0 E+8
8.0	7.3 E+9
6.0	5.9 E+10
4.0	5.8 E+10
3.0	5.2 E+10
2.6	6.7 E+10
2.2	7.2 E+10
1.8	8.3 E+10
1.4	9.1 E+10
1.0	7.5 E+10
0.75	6.8 E+10
0.5	6.0 E+10
0.25	9.8 E+10
0.003	

B. Post-operation gamma sources in core[1] (MeV/sec - watt)

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

TABLE 12.2-3 (Continued)

C. Gamma ray sources in non-core regions during operation (MeV/cm^3-sec-watt)

Energy Bounds (MeV)	Region H Inside	Region H Outside	Shroud Inside	Shroud Outside
16.5	2.8 E-1	3.6 E-2	2.5 E+2	4.1 E+1
8.0	2.5	3.0 E-1	8.2 E+2	1.3 E+2
6.0	4.8 E-3	6.2 E-4	2.4 E+2	3.9 E+1
4.0	2.3 E-2	4.1 E-3	1.1 E+2	1.9 E+1
3.0	1.0 E-3	1.4 E-4	4.3 E+1	1.3 E+1
2.6	2.3 E+2	4.9 E+1	2.4 E+1	4.5
2.2	5.4 E-3	1.0 E-3	2.7 E+1	5.4
1.8	6.3 E-5	8.1 E-6	7.1 E+1	1.3 E+1
1.4	2.6 E-3	5.0 E-4	3.3 E+1	6.5
1.0	7.5 E-3	1.4 E-3	3.9 E+1	9.9
0.75	4.6 E-4	5.8 E-5	2.9 E+1	4.7
0.5	-	-	1.2 E+2	1.9 E+1
0.25	-	-	9.3 E+1	1.5 E+1
0.003				

Energy Bounds (MeV)	Region I (Jet Pumps) Inside	Region I (Jet Pumps) Outside	Vessel Inside	Vessel Outside
16.5	1.4	2.2 E-2	2.2 E-1	2.1 E-4
8.0	4.6	6.9 E-2	2.1	1.6 E-3
6.0	1.3	2.0 E-2	5.6 E-1	1.6 E-3
4.0	6.3 E-1	9.3 E-3	2.8 E-1	2.0 E-3
3.0	2.6 E-1	3.7 E-3	1.0 E-1	1.2 E-3
2.6	6.1	4.7 E-2	4.7 E-2	1.1 E-3
2.2	1.8 E-1	2.4 E-3	5.3 E-2	1.4 E-3
1.8	4.2 E-1	6.0 E-3	1.8 E-1	1.1 E-3
1.4	2.1 E-1	2.9 E-3	7.5 E-2	9.3 E-4
1.0	3.2 E-1	3.6 E-3	9.1 E-2	5.5 E-3
0.75	1.6 E-1	2.4 E-3	6.4 E-2	4.2 E-5
0.50	6.6 E-1	9.9 E-3	2.5 E-1	2.0 E-4
0.25	5.1 E-1	7.6 E-3	1.9 E-2	1.8 E-4
0.003				

NOTE:

(1) Operating history of 3.2 years.

TABLE 12.2-4

GAMMA RAY AND NEUTRON FLUXES OUTSIDE THE VESSEL WALL

(Data used in the Original Design of the Containment Shields)

A. Neutron fluxes

Energy Bounds		Neutrons/cm^2-sec.
16.5	MeV	5.8 E+6
10.00	MeV	2.9 E+7
6.065	MeV	2.2 E+7
3.679	MeV	4.5 E+7
2.231	MeV	7.5 E+7
1.353	MeV	1.1 E+8
820.8	KeV	1.6 E+8
497.9	KeV	1.5 E+8
302.0	KeV	9.1 E+7
183.2	KeV	1.1 E+8
67.38	KeV	1.2 E+7
24.79	KeV	6.7 E+7
9.119	KeV	1.4 E+7
3.355	KeV	8.6 E+6
1.234	KeV	6.4 E+6
454.0	eV	2.9 E+6
167.0	eV	4.2 E+6
61.44	eV	3.9 E+6
22.60	eV	1.9 E+6
13.71	eV	2.0 E+6
8.315	eV	1.8 E+6
5.043	eV	1.6 E+6
3.059	eV	1.5 E+6

TABLE 12.2-4 (Continued)

A. Neutron fluxes (Continued)

Energy Bounds	Neutrons/cm^2-sec.
1.855 eV	1.4 E+6
1.125 eV	7.9 E+5
0.616 eV	6.0 E+5
0.000 eV	

B. Gamma ray energy fluxes

Energy Bounds (MeV)	MeV/cm^2-sec^2
16.5	1.0 E+9
8.0	3.4 E+9
6.0	3.3 E+9
4.0	1.7 E+9
3.0	7.0 E+8
2.6	1.0 E+9
2.2	6.9 E+8
1.8	6.1 E+8
1.4	5.3 E+8
1.0	3.2 E+8
0.75	4.2 E+8
0.50	4.0 E+8
0.25	1.5 E+8
0.003	

TABLE 12.2-5

RADIATION SHIELDING SOURCE TERMS

Equipment Identification		Source Volume (cc)	Shielding Sources (γ/cc-sec) 0.2 MeV	0.6 MeV	1.0 MeV	1.6 MeV	2.4 MeV	3.4 MeV	5.0 MeV	6.1 MeV	7.1 MeV
A. AUXILIARY BUILDING											
E12B001	RHR hx shutdown mode	8.4+6	1.4+4	1.2+4	1.1+4	8.2+3	1.0+3	4.9+1	2.2-1		
E12C002	RHR pump	1.4+5	1.4+4	1.2+4	1.1+4	8.2+3	1.0+3	4.9+1	2.2-1		
E12C001	LPCS pump	1.4+5	1.4+4	1.2+4	1.1+4	8.2+3	1.0+3	4.9+1	2.2-1		
E51C002	RCIC pump turbine	2.8+5	3.4+3	1.6+3	1.2+3	2.1+3	2.3+2	1.2+2	5.3+1	4.6+4	3.3+3
G33C001	RWCU pump	1.1+4	1.4+5	6.6+4	6.3+4	7.9+4	8.8+3	3.4+3	1.5+3	1.2+5	8.5+3
G40B0005	ADHR Heat Exchanger	9.2+6	5.0+1	4.1+3	1.1+4	1.5+5					
B. REACTOR BUILDING											
G33B002	RWCU hx	2.5+5	1.4+5	6.6+4	5.3+4	7.9+4	8.8+3	3.4+3	1.5+3	2.2+4	1.6+3
G36A003	RWCU F/D bkwsh. rec. tk	1.8+6	5.3+6	2.6+6	1.4+6	1.1+6	6.7+4	1.3+3	1.7+1		
G36C001	RWCU F/D holding pump	1.1+4	1.4+5	6.6+4	5.3+4	7.9+4	8.8+3	3.4+3	1.5+3	2.2+4	1.6+3
G36D001	RWCU F/D	1.4+6	1.9+7	9.0+6	4.7+6	4.0+6	2.4+5	7.4+3	1.6+3	2.2+4	1.6+3
G50C012	RWCU bkwsh. trans. pump	1.4+4	5.3+6	2.6+6	1.4+6	1.1+6	6.7+4	1.3+3	1.7+1		
C. INTERMEDIATE BUILDING											
G41A002	Fuel pool surge tk.	2.8+7	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41A003	Fuel trans. tube drn. tk.	6.1+6	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41B001	Fuel pool HX	1.9+6	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41C001	Fuel pool F/D holding pump	7.0+3	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41C003	Fuel pool circ. pump	7.7+4	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41C004	Cask pool drn. pump	1.3+4	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41C005	Fuel trans. tube drn. pump	7.0+3	1.2+4	4.2+3	1.7+1	3.0+0	3.0+0	1.8-1			
G41D001	Fuel pool F/D	9.5+6	4.5+5	3.3+5	1.9+5	1.5+5	6.6+3				
G50A022	Fuel pool F/D bkwsh. rec. tk.	2.6+7	4.5+5	3.3+5	1.9+5	1.5+5	6.6+3				
G50C027	Fuel pool F/D bkwsh. trans. pump	2.8+4	4.5+5	3.3+5	1.9+5	1.5+5	6.6+3				

TABLE 12.2-5 (Continued)

Equipment Identification		Source Volume (cc)	Shielding Sources (γ/cc-sec) 0.2 MeV	0.6 MeV	1.0 MeV	1.6 MeV	2.4 MeV	3.4 MeV	5.0 MeV	6.1 MeV	7.1 MeV
D. RADWASTE BUILDING											
G50A001	Liquid waste coll. tk.	1.3+8	4.4+4	2.9+4	3.3+4	2.4+4	2.9+ 3	2.2+2	2.7+0		
G50A002	Liquid waste sample tk.	1.3+8	4.4+4	2.9+4	3.3+4	2.4+4	2.9+ 3	2.2+2	2.7+0		
G50A003	Floor drns. coll. tk.	1.3+8	4.4+4	2.9+4	3.3+4	2.4+4	2.9+ 3	2.2+2	2.7+0		
G50A004	Floor drns. sample tk.	1.3+8	4.4+4	2.9+4	3.3+4	2.4+4	2.9+ 3	2.2+2	2.7+0		
G50A005	Chemical waste tk.	8.8+7	5.5+4	4.9+4	2.4+4	2.0+4	7.5+ 2	4.5+1	6.7+1		
G50A006	Concentrated waste tk.	1.9+7	2.3+6	2.0+6	1.0+6	8.4+3	3.1+ 1	1.9+3	2.8+1		
G50A007	Chem. waste dist. tk.	7.5+7	5.6+0	5.0+0	2.5+0	2.1+0					
G50A009	Spent resin tk.	3.3+7	3.9+6	1.9+6	2.2+6	9.3+5	4.1+ 4	9.6+2	6.6+0		
G50A011	Cnds. F/D settling tk.	3.6+7	4.4+6	4.3+6	1.9+6	1.7+6	6.6+ 4	4.1+3	6.1+1		
G50A013	RWCU settling tk.	5.9+6	1.9+7	9.0+6	4.7+6	4.0+6	2.4+ 5	7.9+3	1.6+3		
G50A014	Waste sludge settling tk.	8.0+7									
G50A024	Waste coll. filtrate tk.	1.5+6	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50A025	Floor drains, filtrate tk.	1.5+6	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50Z001	Waste evap. condenser	5.7+7	2.3+6	2.0+6	1.0+6	8.4+5	3.1+4	1.9+3	2.8+1		
G50C001	Waste collector trans. pump	1.2+4	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C002	Waste sample pump	7.3+3	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C003	Floor drns. coll. pump	1.2+4	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C004	Floor drns. sample pump	7.0+3	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C005	Chemical waste pump	7.0+5	5.5+4	4.9+4	2.4+4	2.0+4	7.5+2	4.5+1	6.7+1		
G50C006	Chemical waste dist. pump	1.2+4	5.6+0	5.0+0	2.5+0	2.1+0					
G50C008	Spent resin pump	1.2+4	8.2+6	3.5+6	2.6+6	1.6+6	9.9+4	2.4+3	1.6+1		
G50C010	Cond. sludge disch. mix pump	1.2+4	1.0+6	9.6+5	4.3+5	3.9+5	1.5+4	9.3+2	1.4+1		
G50C011	Cond. sludge decant pump	1.2+4	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C013	RWCU sludge disc. mix pump	1.2+4	7.3+6	3.5+6	1.8+6	1.5+6	9.3+4	3.1+3	6.1+2		
G50C014	RWCU sludge decant pump	2.9+3	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C015	Waste sludge disc. mix pump	1.2+4	7.3+6	3.5+6	1.8+6	1.5+6	9.3+4	3.1+3	6.1+2		

TABLE 12.2-5 (Continued)

Equipment Identification		Source Volume (cc)	Shielding Sources (γ/cc-sec) 0.2 MeV	0.6 MeV	1.0 MeV	1.6 MeV	2.4 MeV	3.4 MeV	5.0 MeV	6.1 MeV	7.1 MeV
D. RADWASTE BUILDING (Continued)											
G50C016	Waste sludge decant pump	1.2+4	4.4+4	2.4+4	3.3+4	2.4+4	2.9+3	2.2+2	2.7+0		
G50C017	Waste coll. filtrate pump	4.7+3	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3	2.2+2			
C50C018	Floor drns. filtrate pump	4.7+3	4.4+4	2.9+4	3.3+4	2.4+4	2.9+3				
G50C026	Conc. waste trans. pump	1.2+4	2.3+6	2.0+6	1.0+6	8.4+5	3.1+4	1.9+3	2.8+1		
G50D001	Waste collector filter	7.1+5	2.0+7	8.4+6	6.4+6	3.4+6	2.4+5	5.7+3	3.9+1		
G50D002	Floor drains filter	7.1+5	2.0+7	8.4+6	6.4+6	3.9+6	3.9+6	5.7+3	3.9+1		
G50D003	Waste demin.	2.3+6	2.0+7	8.4+6	6.4+6	3.9+6	3.9+6	5.7+3	3.9+1		
G50D004	Floor drns. demin.	2.3+6	2.0+7	8.4+6	6.4+6	3.9+6	3.9+6	5.7+3	3.9+1		
E. TURBINE POWER COMPLEX											
N23D001	Condensate filter	5.0+5	4.4+6	4.3+6	1.9+6	1.7+6	6.6+4	4.1+3	6.1+1		
N24A001	Conds. demin. cation regen. tk.	7.4+6	3.4+5	3.0+5	1.5+5	1.3+5	4.6+3	2.8+2	4.1+3		
N24A002	Conds. demin. anion regen. tk.	3.7+6	3.4+5	3.0+5	1.5+5	1.3+5	4.6+3	2.8+2	4.1+3		
N24A003	Conds. demin. mix & hold tk.	7.4+6	7.6+1	3.3+2	5.2+2	1.5+2	1.3+1	3.5+0	5.4+2		
N24A004	Conds. demin. bkwsh. rec. tk.	1.9+7	3.4+5	3.0+5	1.5+5	1.3+5	4.6+3	2.8+2	4.1+3		
N24A005	Conds. demin. regen. chem. rec. tk.	4.5+7	5.5+4	4.9+4	2.4+4	2.0+4	7.5+2	4.5+1	6.7+1		
N24C001	Waste transfer pump	7.3+3	5.5+4	4.9+4	2.4+4	2.0+4	7.5+2	4.5+1	6.7+1		
N24D001	Condensate demin.	7.4+6	3.4+5	3.0+5	1.5+5	1.3+5	4.6+3	2.8+2	4.1+0		
G50A010	Cnds. F/D bkwsh. rec. tk.	2.0+7	1.1+5	1.1+5	4.7+4	4.3+4	1.6+3	1.0+2	4.1+0		
G50C009	Cond. bkwsh. trans. pump	2.8+4	1.1+5	1.1+5	4.7+4	4.3+4	1.6+3	1.0+2	1.5+0		
F. TURBINE BUILDING											
N64B001	Offgas preheater	5.1+5	1.7+5	6.3+4	4.0+4	3.2+4	1.5+4	5.4+3	2.8+2	2.4+5	
N64B002	Offgas condenser	9.6+5	3.8+6	1.9+6	1.1+6	8.7+5	4.5+5	9.9+4	5.8+4	1.4+6	
N64D005	Offgas catalytic recombiner	2.4+6	4.1+4	1.6+4	9.8+3	7.8+3	3.8+3	1.3+3	8.2+1	5.3+4	

TABLE 12.2-5 (Continued)

Equipment Identification		Source Volume (cc)	Shielding Sources (γ/cc-sec) 0.2 MeV	0.6 MeV	1.0 MeV	1.6 MeV	2.4 MeV	3.4 MeV	5.0 MeV	6.1 MeV	7.1 MeV
G. OFFGAS BUILDING											
N64B010	Offgas Cooler Condenser	1.2E+5	7.8+6	4.0+6	2.0+6	3.0+6	2.2+6	1.9+5	1.3+5		
N64D011	Offgas Prefilter	2.7E+5	9.40E+5	1.06E+6	1.09E+6	1.19E+6	5.78E+5	1.19E+5			
N64D012	Offgas Charcoal Absorber	7.2E+6	4.03E+6	1.39E+6	1.64E+5	7.7E+5	1.26E+6	5.50E+4			
N64D016	Offgas After-filter	2.7E+5	6.40E+4	9.43E+3	9.90E-2	2.15E+3	9.55E+3	3.26E+0			
N64D003	Gas Dryer	7.5E+5	1.71E+6	7.28E+5	1.49E+5	5.46E+5	4.43E+5	3.03E+4			

NOTE:

(1) Source model geometry for all calculations is cylindrical in shape.

TABLE 12.2-6

MATERIAL COMPOSITION OF THE TIP DETECTORS AND CABLES

A. Detector Region

Material	Quantity
304L stainless (Co ≤ 0.014%)	4.92gm
Titanium	0.662gm
Alumina	0.885gm
Nickel-iron alloy	0.248gm
Copper	0.021gm

B. Cable Region

Material	Quantity
304L stainless (Co ≤ 0.014%)	0.43gm/in.
AISI 1070 steel	2.16gm/in.
Magnesium oxide (Insulation)	0.0798gm/in.

TABLE 12.2-7

RADIATION LEVELS FROM THE TIP DETECTOR AND CABLES[1]

Decay Time Days	Dose Rate, R/hr Detector[2]	Dose Rate, R/hr Cable[3]
0.0014	5.6	54.
0.0035	4.7	53.
0.021	3.7	47.
0.042	3.2	41.
0.083	2.4	32.
0.17	1.4	18.
0.50	0.17	2.2
1.00	0.013	0.10
2.00	0.0038	0.018

NOTES:

(1) Based on three years of simulated use consisting of one hour of detector exposure in core semi-monthly.
(2) At one meter.
(3) At one meter from the midpoint of the 12 foot length of irradiated cable adjacent to the detector.

TABLE 12.2-8

TRAVERSING INCORE PROBE DETECTOR

DECAY GAMMA ACTIVITIES OF MATERIALS IN THE DETECTOR[1]

Decay Time = 0 Seconds

Activation Time = 10^2 Seconds

Activated (Isotope)	Activity (μCi)
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

NOTE:

[1] Excluding U-235.

TABLE 12.2-9

DECAY GAMMA ACTIVITIES OF MATERIALS IN THE CABLE

Decay Time = 0 Seconds

Activation Time = 10^2 Seconds

Activated Isotope	Activity (μCi/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

TABLE 12.2-10

TYPICAL TURBINE COMPONENT N-16 INVENTORIES

	System/Components	Inventory (Curies)
1.	Main steam line and header system	263
2.	High pressure turbine	6.4
3.	Low pressure turbines (6 flow machine)	9.8
4.	Moisture separator shell-side steam	53
5.	Moisture separator shell-side liquid	41
6.	Moisture separator drain system	56
7.	First stage reheat system[1]	33
8.	Second stage reheat system[1]	32
9.	First stage reheat drain system[2]	1.4
10.	Second stage reheat drain system[2]	1.1
11.	Crossover pipe system	59
12.	Crossaround pipe system	17
13.	Feedwater heater and extraction system	
	First Stage[3]	26
	Second Stage[3]	23
	Third Stage[3]	27
	Fourth Stage[3]	15
	Fifth Stage[4]	0.6
	Sixth Stage[5]	42
	Condenser[6]	286
14.	Hotwell[7]	18
15.	SJAE first stage system[8]	0.6
16.	Recombiner system	0.4
17.	Separate steam system[9]	0.9
18.	Feedwater turbine system[9]	8.8
	Total	1,021

TABLE 12.2-10 (Continued)

NOTES:

(1) Includes inventory in liquid and steam in reheat tubes and in steam supply line.

(2) Includes total inventory beyond reheater outlet.

(3) Includes total inventory beyond extraction point. Distribution of this will depend on equipment arrangement and sizing.

(4) Excludes moisture separator drain system activity listed in Item 6.

(5) Excludes first and second stage reheat drain system activities listed in Items 7, 8.

(6) Excludes residual activity returned from feedwater turbine.

(7) Excludes residual activity returned from feedwater turbine.

(8) Includes inventory in steam supply system.

(9) Includes total inventory beyond inlet at steam supply line.

TABLE 12.2-11

PARAMETERS AND ASSUMPTIONS USED IN CALCULATING REACTOR BUILDING AIRBORNE ACTIVITY

Initial iodine partition factor in suppression pool	.00086
Iodine halving time in suppression pool	1,000 hours
Initial noble gas partition factor in suppression pool	0.5
Noble gas halving time in suppression pool	96 hours
Suppression pool cleanup demineralizer decontamination factor for iodine	100

TABLE 12.2-12

REACTOR BUILDING AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
Kr-85m	2.6-7
Kr-85	5.7-9
Kr-87	2.6-7
Kr-88	5.7-7
Kr-89	7.0-8
Xe-131m	3.5-9
Xe-133m	4.4-8
Xe-133	1.7-6
Xe-135m	7.0-8
Xe-135	1.6-6
Xe-137	9.8-8
Xe-138	2.2-7
I-131	1.1-9
I-132	4.8-10
I-133	3.8-9
I-134	2.6-10
I-135	2.0-9

TABLE 12.2-13

RADWASTE BUILDING AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
I-131	1.4-11
I-132	1.5-10
I-133	1.0-10
I-134	2.2-10
I-135	1.5-10

TABLE 12.2-14

TURBINE BUILDING AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
Kr-83m	3.9-9
Kr-85m	7.6-9
Kr-85	2.8-11
Kr-87	2.2-8
Kr-88	2.4-8
Kr-89	3.4-8
Xe-131m	1.9-11
Xe-133m	3.5-10
Xe-133	1.1-8
Xe-135m	1.9-8
Xe-135	2.8-8
Xe-137	4.7-8
Xe-138	6.7-8
I-131	1.0-11
I-132	1.2-10
I-133	7.5-11
I-134	2.0-11
I-135	1.2-10

TABLE 12.2-15

FUEL HANDLING AREA AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
I-131	3.2-12
I-133	1.4-11
I-135	1.1-11

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

The PNPP has been designed to attain as low as is reasonably achievable radiation doses to plant personnel as well as personnel located around the facility. The guidance of <Regulatory Guide 8.8> has been used in designing the facility to result in radiation doses that are only a small fraction of the limits given in <10 CFR 20>. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.)

12.3.1.1 Equipment and Facility Design Features

The plant restricted area is established in accordance with <10 CFR 20> and is separated into five controlled access zones to aid in the design of radiation protection features and plant operation. An access zone designation is assigned to each area of the plant for each of two operating conditions: normal operation and shutdown/refueling. <Figure 1.2-1> lists the Perry equipment names and numbers.

<Figure 12.3-1>, <Figure 12.3-2>, <Figure 12.3-3>, <Figure 12.3-4>, <Figure 12.3-5>, <Figure 12.3-6>, <Figure 12.3-7>, <Figure 12.3-8>, <Figure 12.3-9>, <Figure 12.3-10>, and <Figure 12.3-11> show the plant layout. These maps furnish design guidance for normal operating and shutdown plant dose rates. They provide the basis for decision making for locating and designing shielding and equipment in accordance with ALARA design principles. Design reviews include a review of actual in-plant conditions to verify that the intended design and installation

is in keeping with ALARA principles and the design guidance provided in these figures. <Figure 12.3-1>, <Figure 12.3-2>, <Figure 12.3-3>, <Figure 12.3-4>, <Figure 12.3-5>, <Figure 12.3-6>, <Figure 12.3-7>, <Figure 12.3-8>, <Figure 12.3-9>, <Figure 12.3-10>, and <Figure 12.3-11> include:

a. Locations of the sources described in <Section 12.2> and <Chapter 11>.

b. Shield wall thicknesses.

c. Design radiation zones for normal operation and refueling.

d. Personnel and equipment decontamination areas.

e. Locations of health physics facilities.

f. Control panels for radioactive waste treatment equipment.

g. Onsite laboratories for analysis of chemical and radioactive samples.

h. Counting room.

These figures also illustrate locations of airborne radioactivity and area monitors.

The counting room is located so that the background radiation levels will be low enough to allow for continuous occupancy and to provide an accurate analytical environment under normal operating conditions and anticipated operational occurrences. The counting room is sized to provide adequate space for the required instrumentation. See <Section 12.5> for a discussion of instrumentation in the counting room.

Nonradioactive equipment that may require maintenance is located, when possible, in either Zone I or Zone II. Adjacent areas containing potentially radioactive systems are designed to maintain a radiation level less than the Zone IV maximum (100 mrem/hr) during required maintenance.

Equipment located in Zone IV or Zone V is designed to minimize required maintenance and to be operated remotely. Shield wall penetrations for remote operating devices, electrical equipment, pipes, and ventilation ducts are designed and located at positions that prevent a direct line of sight to any significant source, thereby minimizing radiation streaming.

The primary defense against corrosion product buildup and associated neutron activation in the reactor vessel followed by crud transport is to minimize the input of impurities (i.e., iron, cobalt) in the feedwater. The Perry Plant design includes both full flow condensate filters and deep bed demineralizers. This design provides maximum removal of both suspended and dissolved impurities. In addition, an extensive condenser sampling and analysis system is provided to ensure prompt detection of small condenser leaks. Condensate demineralizers are provided to measure water quality in the bed effluent as described in <Section 10.4.6.2>.

The following design considerations have been given to reduce radiocobalt production and crud buildup in normally radioactive systems:

a. System materials are specified for low corrosion and erosion rates and for low neutron activation source characteristics. Hardfacing materials which have high cobalt content, such as Stellite, are used only where substitute materials cannot satisfy performance requirements.

b. Packless valves (bellows seals or live-loaded valve packings for vent and drain valves) are specified for systems which normally handle radioactive fluids. Where packed valves are specified, they are provided with positive backseats, lantern ring leakoffs to the liquid waste management system and special close tolerance graphoil packing in lieu of conventional packing.

c. System design considers decontamination of components. Isolation, vent and drain valves are provided in suitable locations to facilitate local decontamination of system components.

d. Piping systems are of all welded construction with minimum use of flanged and socket weld connections.

Design practices will allow, whenever practicable, the separation of radioactive piping from nonradioactive piping, electrical equipment and personnel passageways.

The following examples illustrate specific design features that aid in minimizing exposure levels:

a. Components containing radioactive materials will be separated, when practicable, to reduce radiation doses associated with maintenance.

b. Cubicles are sized to provide adequate space around components for anticipated maintenance operations and for ease of entry and exit.

c. The service mode of operation for the waste filters proceeds automatically after operator initiation from the radwaste building control room. Backwashing and precoating are done from a local control panel outside the filter cubicles.

d. Both regenerable and nonregenerable demineralizers have provisions for remote removal of radioactive contents (spent resins or regenerative solutions) to the waste management systems.

e. Activated carbon adsorber media can be removed from the filter plenums by a portable vacuum removal system. Adsorber media can be removed without entering the filter plenum.

f. Particulate filters can be bagged during removal and sealed as the filters are removed from the plenums. Filters are covered with plastic during the entire change.

g. Tanks containing potentially radioactive fluids are vented and can be drained to the waste management systems. Tanks containing fluids at atmospheric pressure are designed to withstand a pressure equivalent to a full tank of water. Static heads will be somewhat less due to overflow lines near the top of tanks.

h. Evaporators have provisions for removal of noncondensibles to the waste management systems. Flush and rinse features permit decontamination before maintenance.

12.3.1.2 Illustrative Examples of Plant Design Features to Minimize Occupational Doses

Plant design features represent a comprehensive effort to achieve minimization of radiation exposure. These features include:

a. Radiation shielding of individual items of equipment.

b. Accumulation of associated items of nonsafety-related equipment within contiguous areas of plant structures.

c. Shielded chases for pipe runs between equipment cells and elsewhere about the plant.

d. Other structural design relative to minimizing radiation exposure to operating and maintenance personnel.

Individual shielding means that the person approaching the location of a radioactive component is shielded from direct and most scattered radiation from both the item of equipment he is approaching (until he enters the equipment area) and all other items of radioactive equipment in the path to, and in the near vicinity of, the equipment being approached.

A semi-automated solid radwaste packaging and handling system has been included in the PNPP design to minimize the radiation doses associated with this routine operation. This system is discussed thoroughly in <Section 11.4>. As mentioned in <Section 12.4>, the anticipated person-rem dose from waste processing for PNPP is only a small fraction of that being experienced in the industry at the time of submittal of the original FSAR.

The Mark III containment design includes a water filled suppression pool that provides the following functions:

a. A heat sink for safety/relief valve (SRV) discharges,

b. A heat sink for hot standby operation,

c. A means to condense steam released to the drywell during a LOCA, and

d. A continuing long term source of water for the emergency core cooling system.

The surface of the suppression pool is open to the containment so that some fractions of radionuclides discharged to the suppression pool from safety/relief valve operation and other sources can evolve into the containment atmosphere. Previous studies of the radiological consequences have concluded that the expected exposures of operation personnel are within the limits of <10 CFR 20>. (Radiological

assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.) However, there is a need to achieve exposures as low as reasonably achievable (ALARA) during normal operation and following normally expected transients.

There are occasional plant upset events that result in steam release from the SRVs. One such event is a complete depressurization of the reactor to the suppression pool following a power isolation transient. During such a transient there may be sufficient increase in radioactivity within the containment to require egress of all personnel. In an effort to reduce these operator exposures at the PNPP, the following change was made during the design process:

a. Addition of a suppression pool cleanup system

   This system uses a mixed bed non-regenerative demineralizer (1,000 gpm). The main benefit of the system is to significantly reduce operator exposure to radioiodides which would evolve relatively slowly from the pool after the transient. In addition, the system will improve plant availability by allowing earlier operator re-entry without the use of respiratory equipment following a power isolation event.

12.3.2 SHIELDING

12.3.2.1 Design Objectives

The design objectives of the plant radiation shielding are:

1. To ensure that during normal operation, including anticipated operational occurrences, the radiation dose to plant personnel and authorized site visitors is as low as reasonably achievable and within the limits set forth in <10 CFR 20>.

2. To provide the necessary protection for plant operating personnel following a reactor accident to maintain habitability of the control room as specified in <10 CFR 50.67>.

3. To limit offsite exposures to the general public to meet the dose requirements of <10 CFR 50.67> for postulated accident conditions and to maintain exposures as low as reasonably achievable, a small fraction of the <10 CFR 20> limits during normal operation.

4. To protect certain components from excessive radiation damage or activation.

12.3.2.2 Design Description

12.3.2.2.1 Plant Shielding Description

Detailed layout drawings showing all plant structures are shown in <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>,

<Figure 1.2-11>, <Figure 1.2-12>, <Figure 1.2-13>, <Figure 1.2-14>, <Figure 1.2-15>, <Figure 1.2-16>, and <Figure 1.2-17>. A general description of the major shielding in the buildings housing radioactive process equipment and fluids is outlined as follows:

a. Reactor building complex

The reactor building complex shielding includes the biological shield wall, drywell shield walls and the shield building wall.

The purpose of the biological shield is to minimize gamma heating in the drywell shield wall, to provide access to the drywell during shutdown and to reduce activation of drywell equipment and materials. The design dose rate used in sizing this shield is to maintain a radiation level in the drywell below 100 rem/hr at full power operation.

The drywell shield wall maintains the major area outside the drywell at Zone II level except for some individual cubicles housing radioactive process equipment and piping, such as cubicles for the reactor water cleanup system and the chase for the main steamline pipes. The shielding for these is sized to maintain a Zone II level outside of each respective cubicle.

Areas in containment routinely visited during power operation include the following systems: SLC(C41), RWCU(G33), CRD(C11), and TIP(C51). The expected occupancy requirements to these areas and the average personnel exposures is provided in <Table 12.4-7>. Their respective locations are shown on <Figure 12.3-2>, <Figure 12.3-3>, <Figure 12.3-4>, and <Figure 12.3-6>. Routine surveillance and control functions are all accomplished from Zone II areas where the design basis gamma dose rate is

≤2.5 mrem/hr. The design neutron dose rate in these areas is negligible due to shielding provided by the five foot thick concrete drywell shield wall.

There are several major penetrations through the drywell shield wall. These as well as all other plant penetrations are located and/or designed to preclude the possibility of streaming from high to low radiation areas, or otherwise will be adequately shielded. Details of the personnel access lock and shield door and the equipment hatch are shown on <Figure 3.8-31> and <Figure 3.8-32>, respectively. <Figure 12.3-2> provides their orientation in the plant layout.

The shield building completely surrounds the steel containment vessel and ensures that levels outside the building are less than 0.5 mrem/hr (Zone I) during normal plant operation. In addition, the building serves to attenuate radiation to plant personnel and the general public in the event of an accident.

b. Turbine room and heater bay

The major source of radiation in the turbine room and heater bay are the N-16 gammas. Shielding is provided around the radioactive equipment in the following systems to ensure that the dose levels are consistent with the access requirements:

1. Condensate and feedwater

2. Condensate filtration and demineralizer

3. High pressure heater drains and vents

4. Turbine

5. Steam seal

6. Condenser and auxiliaries

7. Offgas

Areas within the shield enclosures will normally be restricted access.

c. Turbine skyshine

The 0^{16} (n, p) N^{16} reaction is of interest in the boiling water reactor because of the coolant activation induced by the high energy end of the fission neutron spectrum. The N^{16} present in the steam of a direct cycle boiling water reactor is carried with the steam into the turbine, moisture separators and associated equipment of the secondary cycle. Although the N^{16} decays with a half life of 7.35 seconds, the gamma emission can present a radiation dose problem to the site boundary as a result of the high energy gamma scatter from structures and atmosphere. Relative to this, turbine building gamma ray air scattering (skyshine) analyses with the Morse (Reference 1) Monte-Carlo code and the G-3 (Reference 2) code were made to evaluate the site boundary dose contribution from the N^{16} radiation. The Morse Code results, as presented in (Reference 3), were normalized in the G-3 code. The G-3 point kernel procedure which was then used to perform the site boundary dose rate calculations is based on application of the Klein Nishina scattering formula to the uncollided flux in a predetermined scattering grid. Normal scattering in air was approximated by use of water buildup factors on the scattered leg from each point in the scattered grid. This approach as well as previous work cited in literature has shown this method yields

results in very close agreement with Monte-Carlo (for example, above cited Morse Code) air scattering calculations and with measurements.

The turbine component N-16 inventories are presented in <Table 12.2-4> and the layout of the turbine building walls and floors is presented in the general arrangement drawings, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, <Figure 1.2-11>, and <Figure 1.2-12>. The major shields are:

1. A three foot thick outer turbine building wall extending above the moisture separator reheaters.

2. A steel shield plate at the ends of the high and low pressure turbines with a labyrinth entry into the turbine area.

3. A 6-inch steel shield plate inboard of the moisture separator reheaters with labyrinth entry into the moisture separator reheater area.

4. A one foot thick concrete floor slab extending between the turbine and the 6-inch steel shield plate.

For distances beyond 300 feet, a single lumped source was assumed in the turbine building; for distances less than 300 feet, all major sources were considered separately. A curve of the dose rate (mrem/hr) versus distance from the turbine building is presented in <Figure 12.3-12>.

d. Offgas building, auxiliary building, radwaste building, intermediate building, and fuel handling building.

The shielding provided in these buildings is designed to maintain the dose levels consistent with the plant zone designations given in <Section 12.3.1>. As far as practicable, separation of radioactive equipment is provided in order that maintenance and repairs may be accomplished without the necessity of shutdown and decontamination of equipment in adjacent cubicles.

Permanent radiation shielding is provided around all areas of the inclined fuel transfer system (IFTS) to preclude the possibility of any direct radiation streaming to accessible areas of the plant. <Figure 1.2-4>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, and <Figure 9.1-19> provide general arrangement drawings showing the plant areas through which the IFTS tubes (1F42-D001 and 2F42-D001) pass. <Figure 12.3-21> provides details of the shielding configuration and resultant maximum design dose rates during fuel transfer.

There are three Maintenance Access Areas. All three areas are posted with a placard stating that potentially lethal radiation fields are possible during fuel transfer.

Access to these three maintenance access areas, where physical contact with the inclined fuel transfer tube (IFTT) may occur, is controlled through a key activated system.

Information withheld in accordance with 10 CFR 2.390

A key is used to unlock the access door to the valve room area. Further, locking devices controlled by two independent locks, or other stricter control devices, are installed on the IFTS floor plugs providing access to the transfer tube areas, and the IFTS valve room shield doors. The key activated system is fail safe when power is lost to

IFTS system. The IFTS is designed such that access to the valve room area is not possible when the system is in operation and it is not possible to operate the system if any of the three areas are unsecured. Access is to be strictly controlled when the IFTS is in operation and during movement of irradiated components. Two of the areas have concrete plugs with lock bars across the plug to prevent access. The third area is the Valve Room Maintenance Access Area.

The valve room is controlled by a locked door. If access is attempted during IFTS operation, an interlock secures the power to the IFTS. The Valve Access Room's panic button is incorporated into a crash bar on the door. Pushing the crash bar will de-energize the system and alarm in the Control Room, IFTS INOP ACCESS RMS OCCUPIED. When the IFTS is energized, the door to the Valve Access Room is electrically locked, a rotating beacon inside the Valve Access Room is energized, and a warning light external to the room is illuminated. In addition, if entry to the Valve Access Room is attempted, a loud alarm will sound locally.

The charcoal adsorber system, located in the offgas building, is comprised of two parallel trains with four vessels in each train. The shielding for the system is accomplished by providing refrigerated vaults <Figure 1.2-4>. The first vessel in each train is isolated in a separate compartment and the remaining two compartments have three vessels each.

The first vessel in a train is separately shielded since over 80 percent of the total activity in the system is associated with this vessel.

e. Control room

The control room shielding is designed to maintain the dose requirement of 5 rem whole body or its equivalent to any organ for the duration of the accident as specified in <10 CFR 50, Appendix A>, Criterion 19 (for the design basis LOCA analysis the licensing basis limit is 5 rem TEDE). The analysis of the operator dose is presented in <Section 15.6.5>.

The control room emergency filter system used under accident conditions is described in <Section 6.4>.

12.3.2.2.2 General Design Criteria

The following design criteria are applied to the plant shielding to minimize personnel exposures and to fulfill the intent of <Regulatory Guide 8.8>. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.)

a. Access labyrinths are provided for rooms housing equipment that contains high radiation sources to preclude a direct radiation path from the equipment to accessible areas.

b. Piping penetrations, ducts and voids in radiation shield walls are located to preclude the possibility of streaming from a high to low radiation area, or otherwise will be adequately shielded.

c. Shielding discontinuities caused by shield plugs, concrete hatch covers and shield doors to high radiation areas are provided with offsets to reduce radiation streaming.

d. Radioactive piping is routed through high radiation areas where practicable, or in shielded pipe chases in low radiation areas.

e. Sufficient work area and clearance space is provided around equipment to permit ease of servicing.

f. Instruments requiring in situ calibration will not normally be located in high radiation areas.

g. Non-radioactive equipment which requires servicing will not normally be located in proximity with potentially radioactive equipment.

h. Spread of contamination from radioactive spillage is minimized by providing a floor drain system which collects and routes the liquid to the liquid waste processing system for proper handling.

   Decontamination of an area is facilitated by use of materials and coatings which lend themselves to standard cleaning methods.

i. Natural traps which could be potential pockets of corrosion product activity are minimized in pipe and ducts by avoiding sharp bends, rough finishes, cracks, etc.

j. Shielding is provided for all equipment which is anticipated to be normally radioactive. The dose levels are designed not to exceed <10 CFR 20> requirements under the worst operating conditions of the plant.

k. Temporary shielding such as lead blankets, will be available on the site in the event it is ever needed.

l. Remote handling of radioactive materials is provided wherever it is needed and practicable.

m. The guidance of <Regulatory Guide 1.69> has been used with regard to design of concrete radiation shields.

12.3.2.2.3 Calculation Methods

Shielding calculations are performed using the following computer codes:

a. SDC (Reference 4)

This code is used for calculations involving relatively simple source configurations such as cylinders, spheres, slabs, disks, or rod clusters.

b. QAD6G (Reference 5)

This code is used for performing shielding calculations with complex geometries.

c. G-3B (Reference 2)

This code is used to calculate scattering dose rates.

d. ANISIN (Reference 6)

This code is used to calculate the reactor neutron and gamma flux spectra through shield walls.

e. Microshield (Reference 7)

This code is used to calculate shielding for modifications which involve relatively simple source configurations such as cylinders, spheres, truncated cones, disks, and slabs.

f. Microskyshine (Reference 8)

This code is used to calculate shielding for modifications which require complex calculations to account for air scatter.

g. Monte Carlo N-Particle Version 5 (Reference 11)

This code is used to calculate dose rates and is widely used and accepted by the industry to perform radiological analyses.

The source terms used in these computer codes are described in <Section 12.2.1>. In general, the maximum activity possible, even though remote, has been used in most pieces of equipment.

Beyond a conservative choice of source strengths, several simplifying assumptions have been used in the shielding analysis. Normally energy degradation (softening) of the radiation spectrum by the shielding is not considered. Thus, the calculated dose averaged energy is higher than would actually be the case with the given unshielded source.

Effects of structural steel (e.g., rebar) are neglected in shield walls. Instead, a uniform density of ordinary concrete of 2.35 grams/cc is assumed.

Design dose rates are generally calculated on the centerline of the piece of equipment and 6 inches from the outside of the shield wall. In the case of a floor slab, the dose rate is calculated 2 feet above the

floor on the equipment centerline. (Design dose rates, after October 4, 1993, will be calculated on the centerline of the piece of equipment and 30 cm from the radiation source or from any surface that the radiation penetrates with the exception of dose rates that equal or exceed 500 RAD/hr. Dose rates that equal or exceed 500 RAD/hr will be calculated on the centerline of the piece of equipment and 100 cm from

the radiation source or from any surface that the radiation penetrates.) Further attenuation as the receiver moves farther from the shield is neglected.

The resultant design radiation shielding thicknesses provided to maintain Zone II levels are given in <Table 12.3-2> for the radioactive components.

12.3.3 VENTILATION

12.3.3.1 Design Bases

The plant ventilation systems are designed to accomplish the following:

a. Maintain the required ambient air temperature to prevent extreme thermal environmental conditions for operating personnel and equipment.

b. Protect the operating personnel against possible airborne radioactive contamination in areas where this may occur.

c. Ensure that maximum airborne radioactivity levels for normal and emergency operations, including anticipated operational occurrences, are within the limits of <10 CFR 20, Appendix B>, for areas within plant structures and the restricted areas on the plant site where construction workers and visitors are permitted. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.) The maximum levels correspond to design bases reactor coolant inventory.

d. Provide a suitable environment for continuous personnel occupancy in the control room under normal and postaccident conditions in accordance with <10 CFR 50, Appendix A>, Criterion 19.

Those aspects of the plant design that transfer airborne radioactivity into the effluent control systems from equipment cubicles, corridors and operating areas normally occupied by operating personnel are described in <Section 11.3>.

The guidelines used to meet the intended design objectives for the plant ventilation systems are as follows:

a. Air movement patterns are generally from areas of lesser radioactive contamination to areas of progressively greater radioactive contamination prior to final exhaust.

b. Slightly negative pressures are maintained in specific areas such as the annulus between the containment and shield building to prevent uncontrolled release of contamination. The control room is maintained at slightly positive pressure during normal operation to prevent infiltration of potential contaminants.

c. Valves and equipment are as leak tight as practicable to prevent leakage of radioactive fluids and subsequent airborne contamination.

d. Individual air supplies are provided for each building to keep potentially contaminated air flows separate from noncontaminated air.

e. In general, potentially radioactive air is exhausted through filter trains consisting of roughing, HEPA and charcoal filters to reduce onsite and offsite radiation levels. Filtered and monitored exhausts are provided in all buildings that could potentially

contain radioactive airborne contamination, with the exception of the turbine building, which does not have a filtered exhaust. In the turbine building, air exhausted from the heater bay and turbine operating floor is monitored for radioactivity and then directly discharged to the atmosphere through the turbine building/heater bay vent.

f. Roughing, HEPA and charcoal filters are used for filtration of the recirculated air of the control room and associated offices during accident and other abnormal conditions.

g. Radiation exposures will be kept as low as practicable while servicing ventilation equipment by the following provisions (incorporated in the plant design):

1. The ventilation equipment is not located in normally high radioactive areas.

2. Suitable access doors and service aisles are provided to permit ease in servicing and maintenance.

3. The roughing and HEPA filters in the ESF filter trains and the HEPA filters in the non-ESF filter trains are serviced on the downstream side of the filter to minimize personnel exposure.

4. The activated charcoal adsorber is bulk loaded into the permanently installed, seal welded and gasketless adsorber section with the exception of the Intermediate Building Sub-Exhaust plenum which utilizes tray type adsorber cells. Spent charcoal adsorber material is vacuumed from the bottom or top of the plenums and loaded directly into drums for shipment offsite, with new charcoal adsorber material being

added at the top of the adsorber section. Thus, personnel are not directly exposed to potentially contaminated charcoal during the changing operation.

h. Control of airborne contaminants during maintenance operations is accomplished by the ventilation system, in accordance with the requirements of <10 CFR 20> as follows:

1. Equipment redundancy is provided where practicable and idle equipment is isolated by dampers so that these components can be serviced without disrupting the operation of the system.

2. Ventilation is accomplished in potentially contaminated areas by supplying air to the clean areas (corridors) and drawing it into the rooms through doorways and/or wall openings so that the air flow is normally from clean to contaminated areas. This normally precludes any air flow reversal or air flow pattern disruption when doors of access hatches are opened for maintenance operations.

i. Ventilation ducts enter potentially radioactive areas at least 7 feet above the floor where practical. This minimizes direct exposure to personnel from radiation streaming. Penetration criteria are discussed in <Section 12.3.2>.

j. Ventilation equipment and ductwork are located in low exposure areas where practical. Therefore, automatic dampers, manual dampers and fire dampers can be maintained with minimum exposure to radiation. In potentially high radiation areas, only manual balancing dampers on the inlet and outlet registers require attention. These were adjusted and set in position prior to plant operation and require little further maintenance or adjustment.

k. Interior surfaces of ducts are designed to minimize the buildup of dust. Shop made duct joints are welded and field joints are gasketed with bolted connections. Ductwork joints, therefore, do not have gaps where dust could settle.

   The air flow velocity in the ductwork is generally high enough to keep dust suspended in the air stream. For the type of dust expected (light particles from clean surfaces in most auxiliary building areas), an air flow velocity of 1,500 to 2,000 fpm is sufficient.

l. Access panels allow cleaning and inspection of ductwork. When ductwork needs cleaning, vacuum cleaning should usually be adequate. Where practical, ductwork is accessible for service and maintenance.

m. Exhaust air from vacuum cleaning or special ventilation can be ducted to an inlet connection in the building exhaust air system in most cases. In radioactive waste areas and the charcoal cleaning plenum areas, inlet connections are close to the expected cleaning equipment location. This minimizes extensive lengths of temporary ducting.

<Figure 12.3-13>, <Figure 12.3-14>, <Figure 12.3-15>, <Figure 12.3-16>, <Figure 12.3-17>, <Figure 12.3-18>, <Figure 12.3-19>, <Figure 12.3-20>, and <Figure 12.3-22> illustrate typical activated carbon adsorber plenum designs, showing filter mountings, access doors, aisle space, service galleries, and provisions for testing, isolation and decontamination.

The criteria established for changing of ESF air filters and adsorbers in the charcoal cleaning systems, and a discussion of compliance with <Regulatory Guide 1.52> is included in <Section 6.5.1>. Roughing and HEPA filters in non-ESF-activated carbon adsorber plenums are replaced when the pressure loss across the filter exceeds its design dirty value

(which is less than twice the initial clean value. The pressure loss is measured by permanently installed differential pressure indicators. Non-ESF-activated carbon adsorber beds are changed when laboratory tests of representative samples show that the adsorber fails to satisfy the testing requirements of Table 2 of <Regulatory Guide 1.140>. Design features of non-ESF filters are compared to the recommendations of <Regulatory Guide 1.140> in <Table 12.3-3>.

Design bases and methods of operation for the plant ventilation system are discussed in <Section 6.4>, <Section 6.5>, and <Section 9.4>. For many plant ventilation systems protection of operating personnel from airborne radioactivity is not a limiting design consideration. These ventilation systems maintain a temperature suitable for continuous equipment operation only.

The assumptions and analysis regarding the sources and amount of radioactivity that surround and leak into the control room (to adequately meet the radiation control requirements of <10 CFR 50, Appendix A>, Criterion 19) are discussed in <Section 15.6.5>.

Assumptions used in the analysis of the plant ventilation systems are given in <Table 11.3-8> and <Section 12.2.2>. Maximum expected airborne radioactivity levels in the plant structures and building free volumes are also discussed in <Section 12.2.2>.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Area monitoring instrumentation aids in minimizing personnel exposure to radiation. In addition, area monitors located in selected plant areas can provide useful information to the reactor operator following an incident, thereby enhancing the operators' ability to determine the nature and extent of the incident. Area monitors have a range of 0.1 to

10,000 mR/hr. This instrumentation also aids the operator in making correct decisions in directing personnel in the event of an incident involving high radiation. The system is used to monitor and demonstrate conformance with the guidelines of <Regulatory Guide 8.8> and <10 CFR 20>.

Airborne radiation monitoring instrumentation monitors for airborne radioactivity with sufficient sensitivity to provide information to the reactor operator to permit assessment of the radiological conditions within the plant. The system also aids in maintaining personnel exposure as low as reasonably achievable (ALARA). The system is used to monitor and demonstrate conformance with the guidelines of <10 CFR 20>, and <Regulatory Guide 8.8>.

To ensure system availability, the nonsafety radiation monitoring channels, which are non-movable, receive electrical power from an emergency diesel backed instrument bus. This supply is not available immediately following a LOCA.

Radiation monitoring provided in accordance with <Regulatory Guide 1.97> and as referenced to General Design Criterion 64 is discussed in <Section 7.6.1> and <Table 7.1-3>.

Portable instrumentation and laboratory analysis of manual samples are used to assist in the determination of a course of action for major plant incidents.

12.3.4.1 Area Radiation Monitoring

The objective of the area radiation monitoring system is to indicate and record gamma radiation levels in areas where radioactive material may be present, stored, handled, or inadvertently introduced. These monitors will alarm when administrative limits are close to being exceeded.

12.3.4.1.1 Design Bases

The area radiation monitoring system is designed to:

a. Provide plant personnel with a system that will indicate that the radiation levels are below those requiring special monitoring equipment.

b. Provide a system which can aid in minimizing personnel exposure to radiation and maintain occupational radiation exposure ALARA.

c. Provide instrumentation for the reactor operator to monitor selected plant area gamma radiation levels following an incident, thereby enhancing his ability to determine the nature and extent of the incident.

d. Augment and supplement other monitoring systems, such as the leak detection system and the airborne radiation monitoring system, in the detection of incidents involving release of radioactive material.

e. Provide alarms (alert and high radiation) to warn personnel when the gamma radiation level of selected areas increases substantially.

f. Provide a record of radiation levels as a function of time at key strategic areas within the plant.

g. Provide the reactor operator with alert, high radiation and circuit failure alarms for each channel.

h. Aid the operator in personnel deployment decisions following an incident involving high radiation.

i. Assist in the detection of unauthorized or inadvertent movement of radioactive material in the plant.

j. Warn of excessive gamma radiation levels in areas where special nuclear material is stored or handled.

k. Warn personnel of high radiation in areas prior to entry.

A criticality accident alarm system is not required based on the guidelines of <10 CFR 50.68(b)>. However, gamma sensitive area radiation monitors have been installed in the fuel handling and storage areas.

12.3.4.1.2 System Description

The area radiation monitoring system provides continuous detection, measurement and indication of the ambient gamma radiation level through the use of gamma sensitive detectors located in selected areas of the plant. This system supplements radiological protection for plant personnel, helps to minimize personnel exposure to radiation, and aids the reactor operator by providing instrumentation which may be used for monitoring radiation levels throughout the plant during normal operation and following an incident. Data derived from these monitors is used in demonstrating compliance with ANSI N 13.2, <Regulatory Guide 8.2> and <10 CFR 20>. The Nuclear Regulatory Commission's Safety Evaluation Report (SER) which accompanied Amendment 62 to the Perry Nuclear Power Plant (PNPP) Operating License (Technical Specifications) approved the de-classifying of three radiation monitors (1D21N080, D21N320, and D21N330) as criticality monitors. The SER stated that these monitors would continue to be utilized as area radiation monitors to alert personnel if radiation levels exceeded the monitors' setpoints. The system consists of independent channels strategically located throughout the plant in areas where radioactive material may be present or inadvertently introduced, in areas where high radiation levels may

develop, or in areas where the operator may gain information regarding the nature and extent of an incident. Most of the instrument channels use centralized control from rack mounted ratemeters (readout modules) located in the control room. Other instrument channels are local self-contained units that have no readout in the control room.

Channels associated with Unit 1 <Table 12.3-4> have instrumentation readout modules located in the Unit 1 control room. Channels associated with the common areas <Table 12.3-6> of the plant have readout modules located on a common area radiation monitoring panel located in the Unit 1 control room. Channels associated with the waste processing and the radwaste building (except OD21N280) have remote alarms and meter indication in the radwaste building control room.

Most channels are operated from the control room panels. Each of these channels consists of three basic components: detector, alarm indicator unit and a control room readout module. Channels for the personnel air lock, TIP drive area and the upper pool area have additional remote alarm indicator units. The control room channel is provided with a detector and readout module and has no alarm indicator unit.

Local instrument channels <Table 12.3-7> consist of the detector, and an enclosure containing a single channel ratemeter, alarm light and horn. The portable drywell area monitor is an enclosure with a single channel ratemeter. This monitor will actuate the drywell evacuation alarm in a high radiation condition.

The detectors are wall mounted gamma sensitive devices located in the specific area of concern. The alarm indicator unit is located nearby to provide plant personnel in the area with radiation dose rate level

indication, visual alarms and an audible alarm. Where necessary, remote warning units are provided in addition to the local alarm and indicator units.

Each channel has two warning functions at the local alarm indicator unit: an amber warning light corresponds to an alert radiation level and a red warning light, with an associated audible alarm, corresponds to a high radiation level. The only exception is the TIP drive area radiation monitoring channel which uses a flashing (stroboscopic) blue warning light to indicate a high radiation level in lieu of a red warning light. Each channel has visual alarm indication of alert, high radiation and channel failure on the readout module. In addition, all channels (except local channels) are recorded on multipoint recorders located in the control room.

Components of the area radiation monitoring system are classified as nonsafety-related. Because of the relative importance of this system regarding plant personnel safety and information available to the reactor operator, the components of this system are high quality, reliable, stable, and capable of operating in the expected environments at the installed location as provided through the augmented quality program.

During refueling operations in the containment, a postulated fuel bundle drop in the upper pool would cause gamma radiation dose levels in the drywell (at Elevation 655’-0”) to reach approximately 1 Rem/hr. Since personnel will be in the drywell for maintenance and test operations during the refueling period, a portable area radiation monitoring system is provided during this time <Table 12.3-7>.

The instrument channel consists of a detector, local alarm indicator unit, and a connection for interface with the drywell evacuation alarm. The alarm interface is utilized whenever irradiated fuel or irradiated reactor components are removed from the vessel and transferred to the

upper pools, and when irradiated fuel and irradiated reactor components are transferred from the upper pools and placed into the vessel. The channel components are portable, and are placed in the drywell prior to refueling. After refueling is completed, they are removed from the drywell. Electrical power, interconnecting cabling and connection boxes are permanently installed in the drywell. To install the portable components, personnel must place the detector on a wall bracket provided on the side of the primary reactor shield wall at Elevation 655’-0”, and place the alarm and indicator unit at Elevation 629’-0”. Connection boxes are mounted near the mounting brackets. Electrical power for each channel is provided by the local 120-volt ac supply.

Channels 1D21N030 and 1D21N080 have an additional alarm indicator unit remotely located and in full view just outside the respective containment personnel locks. This additional alarm provides plant personnel with containment radiation level status at the point of entry.

Channel 1D21N060 has a remote alarm indicator unit and an additional remote meter mounted on the TIP drive control panel in the Unit 1 computer room, in order to provide the TIP operator with continuous indication of radiation dose-rate in the TIP drive area of the containment.

Channels D21N250, D21N260, D21N270, and D21N290 have an additional remote meter and an alarm in the radwaste control room in order to provide the radwaste control room operator with radiation level information relative to certain areas of the radwaste building.

Channels D21N370 and D21N380 are locally mounted channels and have a remote meter and alarm in the radwaste control room in order to provide the radwaste control room operator with radiation level information relative to certain areas of the radwaste building.

Area radiation monitoring for the solid radwaste drumming and storage area (Elevation 623’ 6” of the radwaste building) is provided by using

two permanently mounted local channels. These channels indicate and alarm locally, and in the radwaste building control room, to warn personnel when gamma radiation levels exceed predetermined limits. Each channel consists of a wall mounted detector, local alarm indicator unit and a meter and alarm in the radwaste building control room. Each alarm indicator unit provides circuitry to activate an amber alert warning light, a red high radiation warning light and a single klaxon horn providing an alarm. The channels and horn are powered by local 120-volt ac supply.

A detailed description of area radiation monitoring equipment is as follows:

a. Detectors

The detector is a gamma sensitive device housed in a sealed container separated from the local alarm indicator unit. A halogen quenched Geiger-Mueller tube is coupled with a preamplifier to convert the incident gamma radiation into an electrical signal which is transmitted to the readout module. Design information for the detector is listed in <Table 12.3-8>. The detector assembly is wall mounted and strategically located so as to effectively survey the area for gamma radiation. The unit can be easily removed from its mounting for calibration or repair. The unit contains a radioactive check source which can be activated from the associated readout module to provide a verification of channel response. Radiation monitor detectors with Geiger-Mueller tubes use circuitry which, upon saturation of the detector, maintain a continuous upscale reading.

b. Local Alarm Indicator Unit

All channels, except the control room channel (1D21N400) and the local channels (1D21N340, D21N380, and

D21N370), are equipped with an alarm indicator unit located near the detector assembly.

The local alarm indicator unit consists of a wall mounted NEMA enclosure containing the following:

1. A readout meter.

2. A high radiation alarm light visible on a 180 degree horizontal azimuth from the wall and protected by a watertight red glass cover and a metal cage. The light receives 120-volt ac, 60-Hertz, electrical power from the readout module.

   Channel 1D21N060 utilizes a flashing (stroboscopic) blue high radiation alarm light visible on a 180 degree horizontal azimuth from the containment TIP drive warning gates and fence. The light receives 120-volt ac, 60-Hertz, electrical power from the readout module.

3. An alert alarm light visible on a 180 degree horizontal azimuth from the wall and protected by a watertight amber glass cover and a metal cage. The light receives 120-volt ac, 60-Hertz, electrical power from the readout module.

4. A klaxon horn providing an audible alarm in conjunction with the high radiation alarm. The horn receives 120-volt ac, 60-Hertz, electrical power from the readout module. The audible alarm is capable of being silenced at the associated readout module in the control room.

c. Remote Warning Unit

Channels 1D21N030, 1D21N060 and 1D21N080, have remote warning units in addition to alarm indicator units. The remote warning units are identical to the alarm indicator unit except that no audible alarm is required.

Channel 1D21N060 has an additional remote radiation dose-rate (mrem/hr) indicator on the TIP drive control panel located in the Unit 1 computer room.

A remote warning light (required for Channels 1D21N060 and 1D21N160 in addition to alarm indicator units) is provided. The light is an incandescent wall mounted, 120-volt ac, 60-Hertz, 75 watt device powered from the associated alarm indicator unit. The light is mounted on a NEMA 12 enclosure. The remote warning light (high radiation alarm) is visible on a 180 degree horizontal azimuth from the wall and is protected by a watertight red glass cover and a metal cage. The only exception is Channel 1D21N060 which has two remote warning lights which are flashing (stroboscopic) blue lights instead of red lights.

d. Readout Module

The readout module contains most of the electronic circuitry for system operation. The module consists of compact, solid state circuitry, a modular design which allows up to 3 modules to be arranged side-by-side in a rack mounted chassis located in the area radiation monitoring panel. The module contains alarm circuitry, a functional control switch and signal processing amplifiers for dose rate indication. Each module contains an independently fused regulated power supply. The modular compact design allows removal of the module from the chassis for replacement or repair. Circuit alignment can be accomplished while the system is energized.

The front panel of the readout module has the following features:

1. Three and one-half inch meter with meter range of 0.1 mR/hr to 10^4 mR/hr.

2. High level alarm lamp.

3. Alert alarm lamp.

4. Failure alarm lamp.

5. Switching capability with the following functions:

   (a) Function switch with off, operate and alarm positions.

   (b) Alarm trip test.

   (c) Check source actuate.

   (d) Horn silence.

   (e) Alarm acknowledge which changes module alert or high level alarm lamp from “flashing” to “on” state, and returns module alarm output contacts to normal.

Design information for the readout module is listed in <Table 12.3-9>.

e. Recorders

Multi-point strip chart recorders, located on the area radiation monitoring recorder panels, provide a permanent record of the

radiation levels at the selected locations throughout the plant. Each centralized radiation monitoring channel provides an input to one of the recorders <Table 12.3-4>.

f. Power Supply

The area radiation monitoring system channels utilizing control from the centralized control room panels receive electrical power from two sources. The 120-volt ac power to the readout module electronics is supplied from non-Class 1E ac instrument bus which is emergency diesel generator backed through a transfer switch (except during a LOCA). The second power source is used to supply 120-volt ac power for all horns and alarm lamps. This power is supplied from a miscellaneous 120-volt ac distribution panel and is not diesel backed.

g. System Setpoints

The alarm setpoints which are not controlled by other licensing requirements are adjustable and are set and revised as necessary based on operational experience gained throughout plant maturation. These setpoints will normally be placed at values consistent with other requirements of the radiation protection program, in order that significant changes in area radiation levels will be alarmed. Based on this philosophy, the resulting operationally meaningful alarms may then be used to direct further investigation or survey activities, as appropriate, in order to ascertain the cause of the change in radiation level.

h. Calibration

Area radiation monitors are to be calibrated on a routine basis and after any major maintenance work is performed on the detector or its associated ratemeter. Detector calibration is obtained by

exposure of the detector to a radioactive source with its activity or radiation exposure rate traceable to a National Institute of Standards and Technology (NIST). Frequency of routine calibration will be established in accordance with plant procedures.

12.3.4.2 Airborne Radioactivity Monitoring

One of the design objectives for plant ventilation systems is to minimize the accumulation of airborne radioactivity in areas within the plant by maintaining proper air movement patterns. This design objective aids in providing protection for plant operating personnel against airborne radioactive material exposure, and aids in maintaining personnel occupational radiation exposure ALARA. The airborne radiation monitoring system supplements the design objective of the plant ventilation systems by monitoring for airborne radioactive contamination in the gaseous, iodine or particulate form, in plant ventilation system exhaust paths and in the atmosphere of certain areas of the plant, in order to demonstrate compliance with <10 CFR 20, Appendix B>. The airborne radiation monitoring system also provides information to the reactor operator that permits an assessment of the radiological conditions to be encountered within the many areas of the plant.

The system complies with <Regulatory Guide 8.8> and <10 CFR 20> to ensure that personnel exposures are maintained ALARA. The system aids the reactor operator in determining the nature and extent of incidents involving the release of radioactivity by augmenting and supplementing the reactor coolant pressure boundary leak detection system and the area radiation monitoring system. Data are recorded to provide a permanent record of plant airborne radioactivity levels. The instrument channels of this system provide control room indication, alarm and control functions as required to limit the dispersal of radioactive material within ventilation systems.

Localized airborne activity monitoring is used for sampling selected work areas as determined by administrative radiological protection procedures to ensure compliance with the requirements of <10 CFR 20, Appendix B>.

12.3.4.2.1 Design Basis

The airborne radiation monitoring system is designed to:

a. Furnish quantitative information (based on representative sampling) to the reactor operator and to operations personnel on the level of airborne radioactivity in plant ventilation systems and selected areas of the plant.

b. Provide a system which can aid in minimizing personnel exposure to airborne radioactivity and maintain occupational radiation exposure ALARA.

c. Furnish information to substantiate radiation surveys as required by <10 CFR 20>, and provide supporting documentation of working environments.

d. Provide instrumentation for the reactor operator to monitor plant ventilation systems, and selected areas of the plant for level of radioactivity during and following an incident, thereby enhancing the ability to determine the nature and extent of the incident.

e. Supplement the leak detection system in detecting leakage from the reactor coolant pressure boundary.

f. Provide overall plant monitoring of airborne radioactivity and reasonable assurance that the ambient airborne radiation levels are below those requiring special monitoring equipment.

g. Supplement other monitoring systems, such as the area radiation monitoring system, in the detection of incidents involving release of radioactive material.

h. Aid in the protection of the plant personnel from exposure to airborne radioactive materials in excess of the levels allowed by <10 CFR 20, Appendix B>.

i. Provide the reactor operator with alarms for each channel (alert, high radiation or channel failure) and alarms for each subsystem (sample flow low).

j. Provide the operations staff with a hard copy record of radioactivity levels in the monitored systems.

k. Continuously monitor the plant ventilation systems for airborne radioactivity in order to permit an assessment to be made of the radiological hazards to be encountered within various regions of plant buildings, and to call attention to equipment malfunction or component failure resulting in the release of radioactivity.

l. Provide instrumentation for use as the basis for initiating actions related to the plant radiation emergency plan.

m. Provide instruments of sufficient range so as to monitor the radioactivity levels postulated for accident events.

12.3.4.2.2 System Description

The airborne radiation monitor typically consists of a particulate measuring channel, an iodine measuring channel and a gas measuring channel. These monitors provide supporting data for the surveillance of plant radioactive levels as recommended by ANSI N13.2 and <Regulatory Guide 8.2> and <Regulatory Guide 8.8> and documentation for

demonstrating compliance with the requirements of <10 CFR 20>. Instrumentation is provided to monitor the locations listed in <Table 12.3-10>.

A typical airborne radiation monitor subsystem is as follows:

A representative sample of air from a ventilation duct is drawn through a sample line to the airborne monitor unit by means of a Roots-type air blower. Sampling of the ducts is achieved by the use of an isokinetic sample probe placed in the air stream. The area of the probe tip(s) is sized so that the velocity of the sample at the probe tip(s) equals the velocity of the air at the design flow rates in the duct. The line is a 1-inch stainless steel pipe with a minimum number of bends and kept short to minimize loss of particulates due to gravity deposition. Sampling guidelines, as outlined in ANSI N13.1 (1969), are used where applicable. <Table 12.3-11> indicates channels which use isokinetic probes.

Sampling points on ventilation ducts are taken, whenever possible, a minimum of five duct diameters downstream from abrupt changes in flow direction or flow entry, and a minimum of three duct diameters upstream of abrupt changes in flow direction; points are chosen such that the ventilation flow is fully developed and mixing is complete.

Sample line flowrate is such that particulate losses are limited due to gravity settling or turbulent flow. Bends in sample lines are of large radius (approximately 10 times line diameter).

The sample passes through a particulate, iodine and gas channel in series. Each channel is independent. In the particulate channel, the sample air passes through a fixed or moving filter which collects particulates and is monitored by a beta scintillation detector, the output of which is preamplified and transmitted to a ratemeter located in the control room. The detector and filter are enclosed in a

4-Pi lead shield to reduce the background radiation effects. In the iodine channel, the sample passes through an activated charcoal cartridge which traps the radioactive iodine. A 4-Pi shielded gamma scintillation detector monitors the cartridge. The output signal is preamplified and transmitted to a ratemeter located in the control room. In the gas channel, the sample enters a 4-Pi shielded volume monitored by a beta sensitive scintillation detector. The output signal is preamplified and transmitted to a ratemeter in the control room.

The gas is exhausted back to the ventilation duct. Differential pressure switches across the filter and charcoal cartridges are provided to give a low flow alarm at the unit and in the control room. Flow regulation is used to maintain a constant flow through the filters of approximately 1-CFM. A flow indicator, flow fault alarms and log ratemeter indication and alarms are also provided on the unit enclosure.

General system design requirements are as follows:

a. Equipment located outside the control room is housed in NEMA Type 12 ventilated enclosures.

b. Setpoint adjustment devices are protected to prevent inadvertent operation.

c. Readout modules are accessible for test, alignment, changing setpoints, and calibration or inspection without interrupting power to the module.

d. The detectors used in the airborne radiation monitoring system are scintillation detectors. The entire detector assembly is built into a housing which serves as a shield against changes due to light photons or electrostatic or magnetic fields. The housing extends entirely over the base of the assembly, making a single unit. The detector assembly is covered by 4-Pi shielding. The

design is such that the detector and source geometry are reproducible. The beta sensitive scintillation detector employs a phosphor with low sensitivity to gamma radiation, and the gamma scintillation subsystem resolution does not exceed 10 percent at full width at half maximum of the 0.661 MeV photopeak.

e. The detector preamplifier is contained within the detector housing. Preamplifiers are drift-free, linear and ensure a high signal-to-noise ratio.

f. High and alert radiation level alarm trip setpoints are adjustable over the entire range at the readout module.

g. Equipment is designed to be capable of withstanding an integrated gamma dose of 10^4 rads.

h. For analog readout modules remote actuated check sources provided with the detector assembly are Cesium-137 for beta scintillation detectors, and Barium-133 for gamma scintillation detectors. For digital readout modules remote LED pulsers are provided with the detector assembly for both beta and gamma scintillation detectors.

The overall accuracy of the system is as follows:

The instrument error shall not exceed ±20 percent of reading over the upper 80 percent of its range, with the error defined as:

$$\% \text{ error} = \frac{R_T - R_R}{R_T} \times 100\%$$

where:

R_T = true quantity based on the signal leaving a pulse rate signal generator.

R_R = indicated quantity based on the linear recorder output signal.

Particulate filter media have a particulate collection efficiency of approximately 99 percent for aerosol particulates of 0.3 micron and larger.

Activated charcoal cartridges or filters provide a minimum of 95 percent efficiency for field iodine retention (elemental and organic).

Sensitivity Requirements:

a. Sensitivity of particulate channels equipped with fixed filter:

The approximate sensitivity is 2.7×10^{-11} μCi/cc for Cs^{137} and is understood as being the concentration of airborne radioactivity which will produce, after 8 hours sampling, a count rate twice the count rate caused by a background radiation of 1×10^{-10} μCi/cc Radon in equilibrium with daughters plus an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

b. Sensitivity of particulate channels equipped with moving filter:

The approximate sensitivity is 4.1×10^{-10} μCi/cc for Cs^{137} and is understood as being the concentration of airborne radioactivity which will produce, at equilibrium, a count rate twice the count rate caused by a background radiation of 1×10^{-10} μCi/cc Radon in equilibrium with daughters plus an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

c. Sensitivity of iodine channels:

The approximate sensitivity is 1.6×10^{-11} μCi/cc for I^{131} and is understood as being the concentration of I^{131} (elemental or methyl iodide) which will produce, after 8 hours sampling, a count rate (with analyzer on differential) equal to the count rate due to background radiation caused by an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

d. Sensitivity of gas channels:

The approximate sensitivity is 4.7×10^{-7} μCi/cc for Kr^{85} and is understood as being the concentration of airborne radioactivity which will produce a count rate equal to the count rate caused by a background radiation of an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

Detectors:

a. Particulate monitoring channel:

The detector is a photomultiplier tube coupled to a beta sensitive plastic scintillator.

b. Iodine monitoring channel:

The detector is a photomultiplier tube coupled to a gamma sensitive NaI (Tl) scintillator.

c. Gas monitoring channel:

The detector is a photomultiplier tube coupled to a beta sensitive plastic scintillator.

Readout Module:

a. The readout module contains most of the electronic circuitry for system operation. The module consists of compact, solid state circuitry and modular design which provides for modules to be arranged side by side in a standard 19-inch rack mounted chassis. The readout modules are located in the control room airborne radiation monitoring instrument panels, except for readout modules associated with the "movable" subsystems. These readout modules are located on the movable equipment enclosure.

b. The analog readout module has a time constant which is inversely proportional to the count rate with the probable statistical error "E" less than 15 percent:

$$E = 0.67 \sqrt{\frac{(cpm)}{2RC}}$$

where RC = instrument time constant, min.

c. Each readout module contains its own independent fused regulated power supply suitable for 120-volt ac, 60-Hertz power input.

d. Each readout module has provision for determining voltages essential for proper channel operation.

e. The readout modules have the following features (front panel):

1. Range: 10 to 10^6 cpm logarithmic
   10 to 10^7 cpm digital, for digital readout modules

2. Meter Size: 4-1/2"; ±2% full scale accuracy
   approximately 3" wide; ±1% full scale accuracy (±1 digit), for digital readout modules

3. Meter Scales: 10 to 10^6 cpm
High voltage 500 to 2,500 volts dc;
Calibration check point.
10 to 10^7 cpm, for digital readout modules
High Voltage 450 to 1,500 volts dc for digital readout modules

4. Alarm Lamps:

   (a) High level.

   (b) Alert.

   (c) Failure.

5. Switching capability with the following functions:

   (a) Function switch: OFF-Calibrate-High Voltage-Operate. (Does not apply to digital readout modules.)

   (b) Check source actuate.

   (c) Alarm trip test.

   (d) Alarm acknowledge.

6. Iodine channel readout modules have a single channel analyzer circuit to provide energy discrimination for selectively monitoring the 364 KeV I^{131} photopeak. These modules are used in conjunction with the iodine sampling subassemblies. Switching is provided on these modules for "Integral" or "Differential" mode. The differential and integral modes differ as follows: The differential mode selects the I^{131} photopeak; the integral mode provides for gross counting of all photopeaks present. The window and baseline of the analyzer is adjustable to allow differential measurement over the complete photopeak.

f. The readout module has the following features:

1. Independent regulated power supply. (No independent regulated power supply provided within digital readout modules)

2. High level alarm contacts (nonlatching).

3. Alert alarm contacts (nonlatching).

4. Failure alarm contacts (nonlatching).

5. Buffered output for computer (0 to 10 volts dc isolated, positive signal).

6. Output for recorder (0 to 10 mV dc).

7. Output for remote readout meter(s).

8. Fixed circuit failure alarm setpoint.

9. Alert level alarm setpoint adjustment (variable over full range).

10. High level alarm setpoint adjustment (variable over full range).

11. Test points (MCA and signal generator jacks). (No MCA jacks provided on digital readout modules)

g. Special features:

1. Both the high level and the alert level alarm lamps “flash” in an alarm condition. These lamps change from the “flashing” state to a “steady on” state when the module alarm acknowledge button is depressed.

2. Switching capability for alarm setpoint trip adjustment is provided such that, when operated, the meter will indicate the alarm setpoint and the alarm trip circuitry will actuate. Alarm point adjustment can also be made at this position.

3. Switching capability for check source operation is provided such that, when the pushbutton is depressed, a radioactive source is actuated at the detector assembly to provide a response check of the channel. For digital readout modules, on LED pulser is actuated at the detector's photomultiplier tube to generate a signal for the response check of the channel. In addition, the circuitry incorporates an alarm defeat provision such that when the check source is actuated, the alert and high level alarm will not actuate. The check source returns to the retracted position upon loss of power.

4. The failure alarm will actuate upon loss of detector voltage, loss of line voltage, loss of amplifier output signal, or loss of detector input signal.

5. The readout module alert level alarm circuit and high level alarm circuit will be manually reset when the initiating signal returns to normal. The failure alarm circuit will automatically reset when the initiating signal returns to normal.

6. Alarm outputs:

   (a) Alarm outputs have a DPDT contact for use with external control and annunciator circuitry.

   (b) Alarm relays are fail safe. The alarm relay remains energized until an alarm signal causes the relay to de-energize.

7. An accessible test point or connector is provided for input of a signal generator to the circuit for calibrating the alarm trip setpoint circuit. Capability is provided for disconnection of the detector input signal during pulse generator test. In addition, for the analog readout modules, a test point or connector is provided for scaler readout at a point after the discriminator circuit.

8. Each analog readout module has two separate alert and two separate high level setpoint adjustment circuits, either setpoint of which may be inserted into the alarm circuit. This provides two levels of alert and two levels of high alarm capability. Digital readout modules have one alert and one high level setpoint adjustment circuit.

9. Analog readout modules are provided with four connectors for termination of wiring on the module. One is provided for termination of electrical power supplied to the module, one provided for termination of field wiring (excluding high voltage), one for termination of high voltage, and one is used for termination of module output circuits (which includes alarm outputs, recorder and computer signals). Digital readout modules are provided with six connectors for termination of wiring on the module.

10. Power supply:

The airborne radiation monitoring channels utilizing control room readout modules receive electrical power from two sources. The 120-volt ac power to the readout module electronics is supplied from non-Class 1E ac instrument bus which is emergency diesel generator backed through a transfer switch (except during a LOCA). The second power source is used to supply 120-volt ac non-Class 1E power for local horns and alarm lamps. This power is fed from miscellaneous 120-volt ac distribution panels and is not diesel backed. Air sample pumps are fed from the 480-volt ac 3 phase bus.

11. Calibration:

Each channel is calibrated routinely by exposure to N.B.S. traceable sources for verification of initial calibration. Calibration of the monitors is also performed following any major required maintenance of the detectors.

<Table 12.3-10> contains a summary of plant airborne radioactivity monitoring. Airborne monitors used as effluent monitors are described in <Section 11.5>. A general description of each unit follows:

a. Reactor Building:

1. Drywell Atmospheric Radiation Monitor (1D17K670)

The purpose of the drywell atmospheric radiation monitor is to provide airborne radiological monitoring during periods of drywell entry as well as indicate drywell activity to the operator during reactor operation. This unit monitors for particulate, iodine and gaseous airborne radioactivity and utilizes a moving particulate filter. Alarms and level indication are provided in the control room and at the monitor which is located outside of the containment. A high radiation alarm on any of the three channels will actuate the drywell evacuation alarm system.

A high radiation alarm on the noble gas channel will close hydrogen purge isolation Valves M51-F090 and M51-F110. This monitor complies with <Regulatory Guide 1.45> for reactor coolant pressure boundary leak detection systems, whereas the channel components are qualified to function during and after a safe shutdown earthquake. Therefore, the monitor is classified as Seismic Category I.

The sample pump motor is powered from the diesel backed Bus F1D08 to ensure system availability.

Isolation valves are provided at all containment penetrations as required for drywell and containment isolation. The intake valves are motor operated ball valves and the discharge valves are solenoid operated. These valves automatically close on a LOCA isolation signal. Valve control switches and indication lights are located in the control room.

2. Containment Atmospheric Radiation Monitor (1D17K680)

The purpose of the containment atmospheric radiation monitor is to alert the operator and personnel entering the containment of the airborne activity levels in the containment. This unit monitors the containment recirculated air for particulates, iodine and gaseous activity <Figure 9.4-16> and is located outside the containment. Local and control room alarms and level indication are associated with this unit. High radiation alarm will actuate the containment and drywell evacuation alarm system. In the event of reactor building isolation, this unit will provide iodine, particulate and gaseous monitoring capability when it is feasible to re-open the containment isolation valves.

Isolation valves are provided at all containment penetrations as required for drywell and containment isolation. These valves automatically close on a LOCA isolation signal. Sample line intake valves are motor operated ball valves. Valve control switches and indication lights are located in the control room.

3. (Deleted)

4. Containment Vessel and Drywell Purge Exhaust Radiation Monitor (1D17K660)

This unit provides particulate, iodine and gaseous monitoring of the containment exhaust airstream <Figure 9.4-17>.

An isokinetic probe at a point outside the containment upstream of the exhaust filter trains is used to obtain the monitored air sample. Local and control room alarms and level indication are available. High radiation alarm will actuate the containment and drywell evacuation alarm.

b. Radwaste Building:

Radwaste Building Ventilation Exhaust Radiation Monitor (D17K720)

The unit provides monitoring of the exhaust airstream for iodine, particulate and gaseous activity at a point upstream of the building exhaust filter trains <Figure 9.4-7>.

A common plenum discharges to either of two filter/exhaust fan trains. One isokinetic probe is located in each suction duct to a filter/exhaust fan train. Each probe line terminates at a common sample delivery line to provide a sample flow of approximately 1-CFM to the monitor unit. Motor operated ball valves (one per line) are installed in each probe line and are interlocked to open when the corresponding exhaust fan train is operated.

Interlocks stop the radwaste building ventilation supply fans (M31C001A & B) upon a high alarm from the gaseous monitoring channel. Alarms and indications are provided locally, in the control room, and in the radwaste building control room.

c. Auxiliary Building:

Auxiliary Building Ventilation Exhaust Radiation Monitor (1D17K700)

This unit provides monitoring of the building exhaust airstream for iodine, particulate and gaseous activity. An isokinetic probe at a point upstream of the building exhaust filter trains <Figure 9.4-5> is used to obtain a 1-CFM air sample. Interlocks stop the auxiliary building ventilation supply fans (M38C001A,B) upon a high alarm from the gaseous monitoring channel. Alarms and indications are provided locally and in the control room. A high level warning light is provided on the local ventilation control panel.

d. Control Complex:

Control Room Airborne Radiation Monitor (D17K770)

This unit is used to monitor the control room atmosphere for particulates, iodine and gaseous radioactivity in order to maintain control room habitability as required by <10 CFR 50, Appendix A>, Criterion 19. An isokinetic probe is used to obtain a 1-CFM air sample from a point downstream of the common supply plenum <Figure 6.4-1>. High gaseous radioactivity initiates a signal to isolate the control room from the outside environment and places the control room ventilation system into the emergency recirculation mode. The system also monitors the recirculated air. Alarms and indication are provided locally and in the control room.

A postulated design basis accident (LOCA) could result in airborne activity entering the control room through the ventilation system. Since a LOCA signal will also place the control room ventilation system into the emergency recirculation mode, and isolate the control room from the environment; and considering the fact that a

LOCA signal itself incorporates sufficient redundancy, the airborne radiation monitor signal is considered a “diverse” signal, and does not require redundancy. Electrical isolation is utilized to disassociate safety class and nonsafety class circuits.

e. Intermediate Building:

Intermediate Building Ventilation Exhaust Radiation Monitor (D17K730)

This unit is used to provide monitoring for gaseous activity in the building exhaust airstream. An isokinetic probe is used at a point downstream of the building exhaust fan <Figure 9.4-18> to obtain a 1-CFM air sample. Iodine and particulate filters are available for sampling purposes and laboratory analysis. Interlocks are provided to stop the intermediate building ventilation supply fan (M33C001) upon a high alarm. Alarms and indications are provided locally and in the control room. A high radiation alarm will also energize a radiation trouble light on the local intermediate building ventilation control panel.

f. Fuel Handling Building:

Fuel Handling Area Ventilation Exhaust Radiation Monitor (D17K710)

This unit is used to provide monitoring for iodine, particulate and gaseous activity in the fuel handling area ventilation exhaust airstream. An isokinetic probe upstream of the filter exhaust plenums <Figure 9.4-4> is used to obtain a 1-CFM air sample. Interlocks are provided to stop the fuel handling building ventilation supply fans (M40C001A,B) upon a high alarm from the gaseous monitoring channel. Alarms and level indication are provided locally and in the control room. High radiation actuates the fuel handling area evacuation alarm system.

g. (Deleted)

h. Offgas Building:

Offgas Building Ventilation Exhaust Radiation Monitor (1D17K760)

This unit is used to provide monitoring of the building exhaust for iodine, particulate and gaseous activity. An isokinetic probe is used at a point upstream of the exhaust filter trains <Figure 9.4-10> to provide a 1-CFM air sample. Alarms and indications are provided locally and in the control room.

12.3.4.3 Detection of MPC (DAC) Levels of Airborne Radioactivity

The function of the airborne radioactivity monitoring system is to monitor the air within a particular enclosure, or the exhaust air from an enclosure, for airborne radioactivity, and to indicate to the operator the level of airborne radioactivity. In performing this function, the system will assist plant operating personnel in maintaining the essentials of personnel industrial hygiene and in maintaining airborne radioactive contamination levels ALARA. The adequacy of the system and the necessity for the particular location of airborne radioactivity monitoring units is based on the following analysis:

The particular radioisotopes considered to be representative of typical airborne activities associated with BWR operation are Cs^{137}, I^{131}, Xe^{135}, Xe^{133}, Kr^{87}, Kr^{85m}, I^{132}, I^{133}, and I^{135}. Of these isotopes, Cs^{137} can be considered to be representative of the particulate group, I^{131} as representative of the iodine (halogen) group and Kr^{85} can be considered to be representative of the noble gas group for the purpose of the calculation which follows. To determine the adequacy of the radiation

monitoring system, the dilution of the airborne radioactivity, as it is mixed with the building ventilation system, must be considered. One MCP (maximum permissible concentration) per <10 CFR 20, Appendix B> of the above mentioned radioisotopes was postulated separately for each subcompartment throughout the plant during normal operation, and the capability of detecting this radioactivity was determined. The term MPC was replaced with the term DAC (derived air concentration) when the revised <10 CFR 20> was implemented in 1993. The DAC values in the revised <10 CFR 20> differ from the MPC values from the older <10 CFR 20> for many isotopes. The DAC values that are equivalent to the older MPC values used in design basis calculations are provided in parentheses. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.) The following data were used in the analysis (<Table 12.3-12>, <Table 12.3-13>, <Table 12.3-14>, <Table 12.3-15>, <Table 12.3-16>, <Table 12.3-17>, <Table 12.3-18>, and <Table 12.3-19>, inclusive):

a. Assume a constant one MPC (or DAC equivalent) level of either Cs^{137}, I^{131} or Kr^{85} in any subcompartment during normal operation.

b. Exhaust flowrate from the subcompartment.

c. Air dilution factor relative to the airborne radiation monitor sampling point is:

$$D = \frac{F_i}{F_t}$$

where: F_i = subcompartment exhaust flowrate (CFM)

F_t = total flowrate at sample point (CFM)

d. Typical sensitivity of airborne radiation monitor as referenced in <Section 12.3.4.2.2>.

e. Airborne radioactivity concentration at radiation monitor sampling point.

$$C = C_i \ D$$

where: C_i = concentration of airborne radioactivity in subcompartment

D = air dilution factor

As a result of the analysis, the following conclusions were reached:

a. Reactor Building:

1. The containment vessel and drywell purge radiation monitor can detect 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) level of Cs^{137} in any reactor building subcompartment listed in <Table 12.3-12>.

2. The containment vessel and drywell purge monitor can detect during purge a noble gas concentration of 1 MPC (0.1 DAC) in the drywell (drywell at purge); containment pool area (refueling operation); containment free space, and the RWCU HX area (normal operation).

3. In the following locations, a noble gas concentration of 10 MPC (1 DAC) can be detected by the containment vessel and drywell purge exhaust radiation monitor:

   (a) RWCU fill and drain backwash receiver tank area

   (b) 654'-0", RWCU valve nest area

   (c) RWCU fill and drain holding pump room

   (d) RWCU fill and drain room

   These areas are radiation Zone V areas and as such are not normally entered.

4. With the use of the containment atmospheric radiation monitor (D17K680), sufficient radiological surveillance is available for the information of personnel in the reactor building.

5. During refueling operations when the reactor is opened, radioactive substances from the reactor coolant may locally contaminate the air and not be detectable for some time on the exhaust monitor. Radiation Protection performs localized surveys as deemed necessary to ensure airborne radioactivity is controlled per <10 CFR 20>.

b. Radwaste Building:

1. The radwaste building ventilation exhaust radiation monitor can detect a 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) of Cs^{137} in any radwaste building subcompartment listed in <Table 12.3-13>.

2. The radwaste building ventilation exhaust radiation monitor can detect a 10 MPC (1 DAC) level of noble gas in any radwaste building subcompartment listed in <Table 12.3-13> except in the RWCU sludge decant pump room A where a 20 MPC (2 DAC) level can be detected.

To assess the relative potential for airborne radioactivity in the areas where there are potential sources, it was assumed, that in a small subcompartment with a low exhaust flowrate, the following conditions exist:

(a) Cold primary coolant in the system.

(b) A concentration in the subcompartment for any of the following isotopes: 1 MPC (0.45 DAC) I^{131}, 1 MPC (1.0 DAC) Cs^{137}, and 1 MPC (0.1 DAC) noble gas Kr^{85}.

(c) Partition factor: 10^{-3} for iodine, 10^{-4} for particulates, 1.0 for noble gases.

Using these assumptions, it was determined by calculation that the amount of leakage from equipment and components required to yield a concentration of 1 MPC (1 DAC) for Cs^{137} or 1 MPC (0.1 DAC) noble gas Kr^{85} was unlikely. However, the leak rate required to yield a 1 MPC (0.45 DAC) level of I^{131} was found to be approximately 31.6 gal/hr. This leak rate is considered abnormal yet in any event, a 1 MPC (0.45 DAC) level of I^{131} in any subcompartment listed in <Table 12.3-13> can be detected by the exhaust radiation monitor. Indication of abnormal leakage from equipment is provided through the use of floor drains and sumps. Sump level alarms are provided to alert the operator when abnormal leakage conditions exist. Abnormal leakage conditions will also be monitored by periodic patrolling of the building by shift personnel.

c. Auxiliary Building:

1. The auxiliary building ventilation exhaust radiation monitor can detect a 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) of Cs^{137} in any building subcompartment listed in <Table 12.3-14>.

2. The auxiliary building ventilation exhaust radiation monitor can detect a 1 MPC (0.1 DAC) level of noble gas in building subcompartments as listed in <Table 12.3-14>, except for some of pump room B where less than 10 MPC (1 DAC) can be detected.

d. Intermediate Building:

The radiation Zone V areas in the intermediate building consist of spent fuel pool cooling and cleanup system equipment areas. These areas are exhausted through the fuel handling area ventilation system. The equipment located in the subcompartments listed in <Table 12.3-15> present no airborne radiological hazard to personnel occupying the intermediate building. Administrative controls for radiation Zone V areas, ventilation flow patterns and absence of potential sources where open access is permitted, will limit personnel exposure to airborne radioactivity.

The intermediate building ventilation exhaust monitor is considered adequate for personnel protection.

e. Fuel Handling Area:

1. The fuel handling area ventilation exhaust radiation monitoring system can detect a 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) of Cs^{137} in any subcompartment listed in <Table 12.3-16>.

2. The fuel handling area ventilation exhaust radiation monitors can detect a 1 MPC (0.1 DAC) level of noble gas in the cask storage pool, spent fuel pool and fuel transfer pool area as listed in <Table 12.3-16>. A less than 10 MPC (1 DAC) level of noble gas can be detected in the radiation Zone V spent fuel cooling and cleanup system equipment areas.

f. Heater Bay:

1. The heater bay ventilation exhaust radiation monitors can detect a 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) of Cs^{137} on any floor level in the building <Table 12.3-17>.

2. A noble gas concentration of 1 MPC (0.1 DAC) on the following floors can be detected.

   (a) Elevation 580’6”

   (b) Elevation 600’6”

3. A 1 MPC (0.1 DAC) level of noble gas at Elevation 620’-6” can be detected in summer and less than 15 MPC (1.5 DAC) in winter operation due to differences in the dilution factors. Less than 15 MPC (1.5 DAC) can be detected in the FDW Lube Oil purifier room in summer or winter operation.

4. A 25 MPC (2.5 DAC) level of noble gas at Elevation 647’-6” hallway can be detected in winter operation and less than 2 MPC (0.2 DAC) can be detected in summer operation.

g. Turbine Building:

1. The turbine building ventilation exhaust radiation monitor can detect a 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) of Cs^{137} in all three areas of interest in <Table 12.3-18>.

2. A noble gas concentration of 1 MPC (0.1 DAC) can be detected by the exhaust radiation monitors in all subcompartments except in the condenser vacuum pump and sample extraction areas where less than 25 MPC (2.5 DAC) can be detected.

h. Offgas Building:

1. The offgas building ventilation exhaust radiation monitor can detect a 1 MPC (0.45 DAC) level of I^{131} or 1 MPC (1 DAC) of Cs^{137} in any subcompartment listed in <Table 12.3-19>.

2. A noble gas concentration of 1 MPC (0.1 DAC) can be detected in areas listed in <Table 12.3-19> by the offgas building ventilation exhaust radiation monitor with the exception of the following areas where 10 MPC (1 DAC) can be detected:

   (a) Elevation 584’-0”

   (b) Elevation 568’-0”, Filter and demineralizer cubicles

   (c) Elevation 548’-6”, condensate demineralizer backwash receiving tank area and condensate filter backwash receiving tank area

   (d) Elevation 602’-6”, desiccant dryer area after-filter, prefilter room

(e) Elevation 624’-0”, hydrogen/oxygen analyzer area

These areas are either radiation Zone V areas, which are normally not entered, or Zone II areas.

In summary, the analysis of the adequacy of the airborne radioactivity monitoring system resulted in the following conclusions. A one MPC (1 DAC) level (equivalent to less than 10 MPC hours) of Cs^{137} or one MPC (0.45 DAC) level (equivalent to less than 10 MPC hours [4.5 DAC hours]) of I^{131} in any subcompartment can be detected by the airborne radioactivity monitoring system. These radioisotopes are representative of the iodine and particulate groups of concern in considering airborne radiation monitoring. Kr^{85} was considered to be representative of the noble gas group of radioisotopes, although noble gases are considered to be an external airborne hazard. A one MPC (0.1 DAC) level of Kr^{85} cannot be detected in certain areas by the airborne radioactivity monitoring system. These areas in most cases are either radiation Zone II or radiation Zone V areas. Areas throughout the plant were considered in detail for their potential in presenting airborne radioactive hazards. It should be emphasized that radiation Zone II areas (normal continuous occupancy areas) contain little or no equipment, components or piping capable of presenting a credible fixed or airborne radiation hazard.

Areas where the dose rate is greater than 100 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates are High Radiation Areas. Areas greater than 100 mrem/hr, but less than 1,000 mrem/hr are barricaded and areas greater than 1,000 mrem/hr are locked, guarded or identified with a flashing light. An air sample of the area will be taken if the Radiation Protection Section suspects the airborne radioactivity exceeds 0.25 MPC (0.30 DAC).

Areas where the dose rate is greater than 500 RAD/hr at one meter from the radiation source or from any surface that the radiation penetrates are Very High Radiation Areas. These areas are double locked, guarded or identified with a flashing light.

Radiation Zone IV areas (controlled, limited access) in the radwaste and offgas building ventilation systems were analyzed for their potential for presenting an airborne radioactivity hazard. It was determined that an MPC (DAC) level of Cs^{137} or Kr^{85} was unlikely. However, an abnormal leak could result in an MPC (DAC) level of I^{131}, yet the radiation monitoring system and sump systems in the building would provide indication to alert the operator of this abnormality.

<Table 12.3-10> lists the ventilation system, by identification number and name and the corresponding airborne monitors and sample points associated with each system. <Table 12.3-11> gives calculated system flow rates which can vary as much as ±15 percent with no major effects upon the detectability of radioactive contaminants in the air.

On the basis of the above discussion, the airborne radioactivity monitoring system is adequate to ensure conservatively sufficient surveillance of airborne radioactive concentrations. The system will provide indication to the operator that an airborne hazard exists, should that hazard be manifest.

The use of portable air samplers will allow radiation protection personnel to locate the airborne radioactivity by sampling the particular subcompartments in that area.

The combination of the airborne radioactivity monitoring system in conjunction with administrative controls restricting and limiting personnel access, standard radiation protection practices, ventilation flow patterns throughout the plant, plant equipment layout, lack of significant sources in generally accessible radiological controlled

areas, and restricted and locked, guarded or identified (with flashing lights) High Radiation Areas, is sufficient to ensure that airborne radioactivity levels will be conservatively acceptable in terms of the required duration of personnel access through each area of the plant. A general review of these concepts follows:

a. High radiation areas, where whole body dose levels may exceed 100 mrem in 1 hour at 30 cm, but less than 1,000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area. Entrance shall be controlled by requiring issuance of a Radiation Work Permit. High radiation areas, where whole body dose levels are 1,000 mrem/hr or greater shall be locked, guarded or identified with a flashing light and conspicuously posted. Entrance shall be controlled by requiring issuance of a Radiation Work Permit.

b. Air flow patterns are consistent with the basic ventilation design criteria of the plant. Clean, filtered outside air is supplied to continuous occupancy areas (corridors, clean areas); these areas are exhausted into rooms and areas of successively higher potential for airborne contamination. Air flow is such that reversal or exfiltration from potentially contaminated areas is normally precluded. This ventilation arrangement essentially eliminates the possibility of personnel exposure to airborne radioactivity in continuous occupancy areas <Section 12.3.3>.

c. Administrative and physical controls are provided for access to areas in which potential sources of hazardous levels of airborne radioactivity from piping and equipment are located. Access shall require a Radiation Work Permit should airborne activity levels warrant one.

d. Maintenance, in a radiologically restricted area, requires a RWP.

e. Radiation protection programs are discussed in <Section 12.5>.

Additional comments are as follows:

a. Auxiliary Building Ventilation Exhaust:

A major hazard has not been identified in this building; however, a monitor is provided for prompt detection of any unusual releases. With the sensitivity as discussed in <Section 12.3.4.2.2>, the monitor will provide direct indication of excessive release. Analysis of the particulate and halogen filters will allow evaluation of the potential airborne hazards and indicate the need for respiratory equipment, if warranted. With a weekly analysis of the activity build-up on the sampling filters, a gross sensitivity of approximately 1.5×10^{-11} μCi/cc can be interpreted. This level is well within <10 CFR 20> limits.

b. Radwaste Building Ventilation Exhaust:

The monitor sensitivity is as discussed in <Section 12.3.4.2.2>. The hazard condition referenced in <Section 15.7.2> will release 6.40×10^{-3} curies of I^{131}. With an approximate room volume of 4,000 ft^3, a ventilation turn-over rate of 6/hour and a total building exhaust of 30,000 CFM, the concentration of I^{131} in the building exhaust is expected to be within the detection range of the monitor. At this level, respiratory equipment may be required to enter the hazardous area.

c. Reactor Building Purge Exhaust:

The failure of an instrument line has been identified as a possible hazard as indicated in <Section 15.6.2>. With a purge air flow, an I^{131} activity of 2×10^{-2} μCi/g in the reactor coolant, an instantaneous mixing of the leaked activity in the whole containment volume and a monitor sensitivity for iodine of approximately 2×10^{-9} μCi/cc (for five minute sampling), a high radiation alarm can be obtained in the control room following the leakage of less than 10 gallons. A plate out factor of 2 is taken into consideration for this evaluation.

With a reactor coolant pressure of 1,000 psi, and a flow restrictor of 1/4 inch, the initial flow rate through the break, assuming 100 percent flashing will be in excess of 10 gpm. Thus a monitor response to the potential hazard of less than 10 minutes is expected.

d. Fuel Handling Building Exhaust:

The drop of a channeled spent fuel bundle has been identified in <Section 15.7.4> as a hazard for personnel in this building.

Assuming that 7.49×10^{2} Ci of Kr^{85} <Table 15.7-34> are released and mixed instantaneously into the whole volume of the fuel handling building, the resulting concentration is expected to be 1.7×10^{-2} μCi/cc. This activity level is well within the range capability of the monitor.

The response time of the monitor is inversely proportional to the activity level and is expected to be negligible at the high anticipated levels which may be reached during this calculable fuel handling accident.

The particulate and iodine filters are removable for laboratory analysis to verify and identify activity levels and to provide a backup to the continual monitoring of the areas of surveillance.

Portable air samplers with appropriate filters will be used to determine localized exposure levels and to permit the proper selection of respiratory protective equipment for the occupied portions of these buildings.

Radiation monitors located in each of the auxiliary, radwaste, reactor, and fuel handling buildings take representative samples of the building exhaust air. Iodine and particulate filters are used to collect samples, and will accommodate the identification of specific types of contamination which can be localized and cleaned up. Radiation protection personnel will maintain a regular sampling routine in these areas to complement the radiation monitors.

12.3.4.4 System Setpoints

The alarm setpoints are adjustable and are set and revised as necessary based on ALARA, licensing requirements and operational experience gained throughout plant maturation. Those setpoints, which do not cause plant equipment actuations and are not fixed by other licensing requirements, will normally be placed at values consistent with other requirements of the radiation protection program, in order that significant changes in airborne radiation levels will be alarmed. Based on this philosophy, the resulting operationally meaningful alarms may then be used to direct further investigation or survey activities, as appropriate, in order to ascertain the cause of the change in radiation level.

12.3.5 REFERENCES FOR SECTION 12.3

1. E.A. Straker, et al., "The Morse Code, A Multigroup Neutron and Gamma Rate Monte Carlo Transport Code," ORNL-4585, Oak Ridge National Laboratory (1970).

2. "G-3, General Purpose Gamma Ray Scattering Code," Rev. 1A, GAI Code N105.

3. W. A. Woolson, et al., "Calculation of the Dose at Site Boundaries from N-16 Radiation in Plant Components," JRB 72 5076J, December 8, 1972.

4. Arnold, E. D. and Maskewitz, B. F., "SDC - A Shield Design Calculation Code for Fuel Handling Facilities," ORNL-3041, March 1966.

5. Malenfont, R. E., "QAD, A Series of Point-Kernel General Purpose Shielding Programs," Los Alamos Scientific Laboratory Report No. 3573, April 5, 1967.

6. Boling, M. A. and W. A. Rhoads, "ANISIN/DT FII, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering." AL-66-MEMD-171.

7. Grove Engineering, Inc., "Microshield User's Manual," Version 2.0.

8. Grove Engineering, Inc., "Microskyshine User's Manual," Version 1.10.

9. NRC Safety Evaluation Report dated June 28, 1994 accompanying Amendment 62 to the Technical Specification.

10. <10 CFR 50.68>

11. Los Alamos National Laboratory, “Monte Carlo N-Particle Version 5 (MCNP5)”, Version 1.60

TABLE 12.3-1

RADIATION ZONE DESIGNATIONS AND CODE

(DELETED)

TABLE 12.3-2

RADIATION SHIELD THICKNESSES

	Equipment Identification		Shield Thickness (ft of concrete)
a.	AUXILIARY BUILDING		
	E12B001	RHR heat exchanger	2
	E12C002	RHR pump	2
	E21C001	LPCS pump	2
	E22C001	HPCS pump	2
	E51C001	RCIC pump	2
	G33C001	RWCU pump	3
	G40B005	ADHR heat exchanger	1-1/2[1]
b.	REACTOR BUILDING		
	G33B001	RWCU heat exchanger (regen.)	4
	G33B002	RWCU heat exchanger (non-regen.)	4
	G36A003	RWCU fill and drain backwash receiving tank	3
	G36C001	RWCU fill and drain holding pump	2
	G36D001	RWCU fill and drain	3-1/2
	G50C012	RWCU backwash transfer pump	2
c.	INTERMEDIATE BUILDING		
	G41A002	Fuel pool surge tank	1
	G41A003	Fuel transfer tube drain tank	2
	G41B001	Fuel pool heat exchanger	2
	G41C003	Fuel pool circulating pump	2
	G41C005	Fuel transfer tube drain pump	2
	G41D001	Fuel pool fill and drain	2-1/2
	G50A022	Fuel pool fill and drain backwash receiving tank	2-1/2
	G50C027	Fuel pool fill and drain transfer pump	2-1/2
d.	RADWASTE BUILDING		
	G50A001	Liquid waste collection tank	2
	G50A002	Liquid waste sample tank	2
	G50A003	Floor drains collection tank	2
	G50A004	Floor drains sample tank	2
	G50A005	Chemical waste tank	2
	G50A006	Concentrated waste tank	2-1/2
	G50A007	Chemical waste distribution tank	1
	G50A009	Spent resin tank	3
	G50A011	Condensate fill and drain settling tank	3
	G50A013	RWCU settling tank	3

NOTE:
[1] The ADHR heat exchanger is shielded by a partial height concrete shield wall supplemented by 5-6 layers of 15 psf lead blankets above the concrete wall.

TABLE 12.3-2 (Continued)

	Equipment Identification		Shield Thickness (ft of concrete)
d.	RADWASTE BUILDING (Continued)		
	G50A014	Waste sludge settling tank	2
	G50A024	Waste collection filtrate tank	2
	G50A025	Floor drains filtrate tank	2
	G50C001	Waste collector transfer pump	1
	G50C002	Waste sample pump	1
	G50C003	Floor drains collector pump	1
	G50C004	Floor drains sample pump	1
	G50C005	Chemical waste pump	1
	G50C006	Chemical waste distribution pump	1
	G50C008	Spent resin pump	2
	G50C010	Condensate sludge discharge mix pump	1
	G50C011	Condensate sludge decant pump	1
	G50C013	RWCU sludge discharge mix pump	2
	G50C014	RWCU sludge decant pump	1
	G50C015	Waste sludge discharge mix pump	1
	G50C016	Waste sludge decant pump	1
	G50C017	Waste collection filtrate pump	1
	G50C018	Floor drains filtrate pump	1
	G50C026	Conc. waste transfer pump	2
	G50D001	Waste collector filter	2
	G50D002	Floor drains filter	2
	G50D003	Waste demineralizer	2-1/2
	G50D004	Floor drains demineralizer	2-1/2
	G50Z001	Waste evaporator cond.	2-1/2
e.	TURBINE POWER COMPLEX		
	N23D001	Condensate filter	2-1/2
	N24A001	Condensate demineralizer cation regen. tank	2-1/2
	N24A002	Condensate demineralizer anion regen. tank	2-1/2
	N24A003	Condensate demineralizer mix and hold tank	2-1/2
	N24D001	Condensate demineralizer	2-1/2
f.	OFFGAS BUILDING		
	N64B010	Offgas cooler condenser	3
	N64D011	Offgas prefilter	3
	N64D012	Offgas charcoal absorber	3
	N64D016	Offgas after-filter	1
	N64D030	Offgas desiccant dryer	3
g.	TURBINE ROOM		
	N25B001	Moisture separator reheater	3
	N31C001	Main turbine	3
	N64B001	Offgas preheater	4
	N64B002	Offgas condenser	4
	N64D005	Offgas catalytic recombiner	4

TABLE 12.3-2 (Continued)

Equipment Identification		Shield Thickness (ft of concrete)
h. HEATER BAY		
N21B003	Compressure feedwater heater no. 3	2
N21B004	Direct contact heater	2-1/2
N27B001	Intermediate pressure feedwater heater no. 5	2
N27B002	High pressure feedwater heater no. 6	2
N27B003	Heater no. 5 drain cooler	1-1/2
N27C003	Main feedwater pump drive turbine	2[1]
N33B002	Steam seal evaporator	2

NOTE:

[1] Except at knockout opening in north wall of the "A" feedpump turbine room due to the distance from sources in the room. No shielding is required at this opening to meet radiation zoning criteria on <Figure 12.3-4>. This exception does not apply to the "B" feedpump turbine room.

TABLE 12.3-3

COMPARISON OF NON-ESF CHARCOAL FILTER SYSTEMS TO <REGULATORY GUIDE 1.140> CRITERIA

Regulatory Position	System Design Feature
1.a	The design conforms with this position.
1.b	The design conforms with this position.
1.c	The design conforms with this position.
1.d	The design conforms with this position.
2.a	All non-ESF charcoal plenums are Seismic Category I and consist of the following components in sequence: prefilters, HEPA charcoal and HEPA. Fans, ducts, dampers, and related instrumentation are also provided.
2.b	The design conforms with this position except that HEPA filter arrangements are not always 3 high by 10 wide.
2.c	The design conforms with the intent of Section 5.6 of ERDA 76-21.
2.d	The design conforms with this position.
2.e	The design conforms with this position.
2.f	The design conforms with this position. Duct and housing leak tests will be performed in accordance with Section 6 of ANSI N510-1980 instead of ANSI N510-1975.
3.a	The relative humidity of exhaust air for the non-ESF charcoal filter systems is not expected to exceed 70 percent.
3.b	The design conforms with this position.
3.c	The design conforms with this position.
3.d	The design conforms with the intent of Section 4.4 of ERDA 76-21.
3.e	The design conforms with this position.

TABLE 12.3-3 (Continued)

Regulatory Position	System Design Feature
3.f	The design conforms with this position.
3.g	The design conforms with this position. The original or replacement batch of impregnated activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of this Regulatory Guide.
3.h	The design conforms with this position. The adsorbent shall meet the requirements of Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI N509-1976.
3.i	The design conforms with this position.
3.j	The design conforms with this position.
3.k	The design conforms with this position.
3.l	The design conforms with this position.
4.a	The design conforms with the intent of the recommendations of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.
4.b	The maximum length of component plus 2 ft 6 inches is provided due to space limitations imposed by equipment room size. It was determined that this is adequate for the replacement of the prefilters and HEPA filters, and is consistent with the manufacturer's recommendation.
4.c	The design conforms with this position.
4.d	Preoperational Phase Testing meets the intent of this position. The testing will be performed while active construction is in progress on the project, but sufficiently complete to ensure that the installed HEPA filters and charcoal are not subjected to airflow that would invalidate inplace testing.

TABLE 12.3-3 (Continued)

Regulatory Position	System Design Feature
5.a	Testing procedures will meet the intent of this position. Visual inspection will be performed in accordance with the provisions of Section 5 of ANSI N510-1980 instead of ANSI N510-1975.
5.b	Testing procedures will meet the intent of this position. The airflow distribution testing will be performed in accordance with the provisions of Section 8.3.2 of ANSI N510-1980 instead of ANSI N510-1975.
5.c	Testing procedures will meet the intent of this position. The inplace leak test on upstream HEPA filter banks will be performed in accordance with the provisions of Section 10 of ANSI N510-1980 instead of ANSI N510 1975. In addition Section 10 of ANSI N510-1980 requires Sections 8 and 9 to be performed as prerequisites to the inplace leak test on the HEPA filter bank and the Section 8 and 9 testing will be performed in accordance with the provisions of Sections 8 and 9 of ANSI N510-1980. Inplace leak testing on downstream HEPA filter banks will not be performed. Testing frequency will meet the intent of the provision but may be based upon refueling outage intervals for systems M14 and M38.
5.d	Testing procedures will meet the intent of this position. The inplace leak test on the charcoal adsorber stage will be performed in accordance with the provisions of Section 12 of ANSI N510-1980 instead of ANSI N510-1975. In addition Section 12 of ANSI N510-1980 requires Sections 8 and 9 be performed as prerequisites to the inplace leak test on the charcoal adsorber stage and the Section 8 and 9 testing will be performed in accordance with the provisions of Section 8 and 9 of ANSI N510-1980. Testing frequency will meet the intent of the provision but may be based upon refueling outage intervals for systems M14 and M38.
6.a(1)	Testing procedures will meet the intent of this position per 5.d above.

TABLE 12.3-3 (Continued)

Regulatory Position	System Design Feature
6.a(2)	Initially installed charcoal will conform with the requirements of this position. New activated carbon meets the requirements of Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 1, <Regulatory Guide 1.140>, March, 1978.
6.a(3)	Plant operating procedures will conform with the requirements of this position. Laboratory testing for non-ESF adsorbers will be conducted in accordance with the specification for testing of ESF adsorbers. Non-ESF adsorbers are tested with no test parameter exceptions. Testing frequency will meet the intent of the provision but may be based upon refueling outage intervals for systems M14 and M38.
6.b	The design conforms to this position. The preoperational testing procedures conform to this position. The plant operating procedures conform to this position, with the exception that utilization of adsorbent samples removed from the subject bed may be used to refill the samplers. The new unused activated carbon used to replace a bed on failure to meet the applicable tests of Table 2 will meet the requirements of Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 1 of <Regulatory Guide 1.140>, March 1978. Testing frequency will meet the intent of the provision but may be based upon refueling outage intervals for systems M14 and M38.

TABLE 12.3-4

DETECTORS ASSOCIATED WITH UNIT 1[1]

Detector[4]	Channel[2]	Location[5]
1D21N030[3]	Personnel air lock	Containment at Elev. 600′-6″
1D21N040	CRD HCU west	Containment at Elev. 620′-6″
1D21N050	RWCU fill and drain receiver tank area	Containment at Elev. 642′-0″ east
1D21N060[3]	TIP drive area	Containment at Elev. 600′-6″
1D21N070	RWCU fill and drain area	Containment at Elev. 664′-7″
1D21N080[3][6]	Upper pool area	Containment at Elev. 689′-6″
1D21N110	Auxiliary building, 574′ east	Auxiliary building at Elev. 574′-10″ east
1D21N120	Auxiliary building, 574′ west	Auxiliary building at Elev. 574′-10″ west
1D21N130	Turbine room east	Turbine room at Elev. 647′-6″ east
1D21N140	CRD HCU east	Containment at Elev. 620′-6″
1D21N160[3]	Turbine room west	Turbine room at Elev. 647′-6″ west
1D21N170	Turbine building, 605′	Turbine building at Elev. 605′-6″
1D21N180	Hotwell pump area	Turbine building at Elev. 577′-6″
1D21N190	Turbine building sump area	Turbine power complex at Elev. 548′-6″
1D21N200	Offgas building, 584′	Offgas building at Elev. 584′-0″
1D21N210	Condensate filter pump area	Turbine power complex at Elev. 568′-6″
1D21N220	Offgas after-filter area	Offgas building at Elev. 602′-6″

TABLE 12.3-4 (Continued)

Detector[4]	Channel[2]	Location[5]
1D21N230	High pressure feedwater heater area	Heater bay at Elev. 600’-6”
1D21N240	Feedpump area	Heater bay at Elev. 647’-6”
1D21N400	Control room	Control complex at Elev. 654’-6”
1D21N410	Offgas holdup area	Turbine building at Elev. 577’-6”

NOTES:

(1) Readout modules located on Panel 1H13-P803 and recorded on Panel 1H13-P600.
(2) All channels have a local alarm and indicating unit except 1D21N400.
(3) In addition to a local alarm and indicating unit, Channels 1D21N030, 1D21N060 and 1D21N080 have remote warning units, and Channel 1D21N160 has a remote warning light. Channel 1D21N060 also has a remote meter mounted on the TIP drive control panel and uses flashing (stroboscopic) blue high radiation alarm lights.
(4) Range of detectors is 0.1 to 10^4 mR/hr.
(5) Detector and local alarm and indicating units located inside containment are housed in unpainted aluminum NEMA Type 4 enclosures. All other equipment is housed in NEMA Type 12 enclosures.
(6) (Reference 9) and (Reference 10) should be reviewed prior to making any modification that would impact this instrument’s ability to be used as an area radiation monitor.

TABLE 12.3-5

(DELETED)

TABLE 12.3-5 (Continued)

(DELETED)

TABLE 12.3-6

DETECTORS ASSOCIATED WITH COMMON AREAS[1]

Detector[4]	Channel[2]	Location
D21N250[3]	Radwaste 574′ west	Radwaste building at Elev. 574′-10″ west
D21N260[3]	Radwaste 574′ east	Radwaste building at Elev. 574′-10″ east
D21N270[3]	Radwaste 602′	Radwaste building at Elev. 602′-0″
D21N280	Process sample room	Radwaste building at Elev. 623′-6″
D21N290[3]	Radwaste evaporator area	Radwaste building at Elev. 623′-6″
D21N310	Fuel pool cleanup fill and drain area	Intermediate building at Elev. 599′-0″
D21N320[5]	Fuel preparation pool	Intermediate building at Elev. 620′-6″
D21N330[5]	Spent fuel storage pool	Intermediate building at Elev. 620′-6″
D21N420	Fuel pool cooling circulating pump area	Intermediate building at Elev. 574′-10″

NOTES:

(1) Readout modules located on Panel H13-P906 and recorded on Panel H13-P907.
(2) All channels have an alarm and indicating unit.
(3) In addition to an alarm and indicating unit, Channels D21N250, D21N260, D21N270, and D21N290 have a remote meter and remote alarm in the radwaste building control room.
(4) Range of detectors is 0.1 to 10^4 mR/hr.
(5) (Reference 9) and (Reference 10) should be reviewed prior to making any modification that would affect these instruments' ability to be used as area radiation monitors.

TABLE 12.3-7

DETECTORS ASSOCIATED WITH LOCAL CHANNELS

Detector[3]	Channel	Location[4]
1D21N340	Unit 1 drywell (portable)	Containment at Elev. 655′-0″ (Azimuth approximately 160°)
D21N380[1]	Waste compactor area	Radwaste building at Elev. 623′-6″
D21N370[1]	Solid radwaste drumming area	Radwaste building at Elev. 623′-6″

NOTES:

(1) Channels D21N380 and D21N370 are locally mounted channels and have a remote meter and a remote alarm in the radwaste building control room.
(2) (Deleted)
(3) Range of detectors is 0.1 to 10^4 mR/hr.
(4) Detector and alarm and indicator units located inside containment are housed in unpainted aluminum NEMA Type 4 enclosures. All other equipment is housed in NEMA Type 12 enclosures.

TABLE 12.3-8

DETECTOR DESIGN REQUIREMENTS

Type	G-M tube
Range, mR/hr	0.1 to 10^4
Energy dependence	±15% (80 KeV to 1.5 MeV)
Circuitry	Solid state preamplifier
Mounting	Wall bracket
Remote Capability	Up to 1,500 feet
Exposure	Capability of withstanding a total integrated dose of 10^5 rads
Enclosure	NEMA 12 (NEMA 4 in containment)
Dead Time	20 μsec.

TABLE 12.3-9

READOUT MODULE DESIGN REQUIREMENTS

Response Time	Meter response is approximately 2.5 seconds for full scale deflection. Time constant of 60, 6, .06 seconds at 0.01, 0.1, 1 mR/hr respectively.
Susceptibility	The input signal is shielded to prevent gross fluctuations and false trips due to normal electromagnetic interference caused by electric motors, circuit breaker closure, welding.
Stability	Drift is less than ±3% of the measured point over a period of 30 days at environmental design center. The system is capable of operation on 120 volt ac, 60 hertz and will operate within specifications under voltage or frequency changes of ±10%. For the operating temperature range the shift due to temperature will be less than 0.5% per °C.
Accuracy	The overall accuracy of the system will be the actual reading relative to the true reading within ±25% of any decade at a reference energy in the range of 0.1 to 2.5 MeV.
Precision	The precision will be ±10% of any single measurement level at the environmental design center.

TABLE 12.3-10

AIRBORNE RADIATION MONITOR SUBGROUP
UNIT 1 AND COMMON

Radiation Monitor Subsystem[2]	Sample Point	Instrument Channels[1][3]	Function of Subsystem	Location
D17K720 Radwaste Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of filter trains	GSP HSP PSP	Local, control room and radwaste control room indication and alarms. Ventilation supply fan trip on high radiation.	Radwaste Bldg. 623’-6” East
1D17K700 Auxiliary Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of filter trains	GSP HSP PSP	Local and control room alarms and indication. Ventilation supply fan trip on high radiation.	Aux. Bldg. 620’-6” West
D17K730 Intermediate Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork downstream of exhaust fan	GSP H&P Filters	Local and control room alarms and indication. Supply fan trip on high radiation.	Intermediate Bldg. 682’-6” S.W.
D17K710 Fuel Handling Area Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of the exhaust filters	GSP HSP PSP	Local and control alarms and indication. Supply fan trip on high radiation. Fuel handling area evac. alarm on high rad.	Intermediate Bldg. 682’-6” N.W.
1D17K760 Offgas Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of exhaust filter trains	GSP HSP PSP	Local and control room alarms and indication.	Offgas Bldg. 635’-0”
1D17K660 Containment Vessel and Drywell Purge Exhaust Radiation Monitor	Ventilation ductwork outside containment upstream of exhaust filter trains	GSP HSP PSP	Local and control room alarms and indication. Drywell and containment evac. alarm.	Intermediate Bldg. 654’-6”
D17K770 Control Room Airborne Radiation Monitor	Ventilation supply duct downstream of common supply plenum	GSP HSP PSP	Local and control room alarms and indication. High radiation on gas channel shifts ventilation into emergency recirculation mode.	Control Complex 679’-6”

TABLE 12.3-10 (Continued)

Radiation Monitor Subsystem[2]	Sample Point	Instrument Channels[1][3]	Function of Subsystem	Location
1D17K670 Drywell Atmospheric Monitor	Drywell, 617’-3” Elev.	GSP HSP PSP	Local and control room indication and alarms and drywell evacuation alarm. Isolation of hydrogen purge Valves M51F090 and M51F110 upon noble gas high alarm.	Fuel Handling Bldg. 620’-6”
1D17K680 Containment Atmospheric Monitor	Recirculated containment air, 674’ Elev.	GSP HSP PSP	Local and control room indication and alarms and containment and drywell evac. alarm.	Intermediate Bldg. 665’-0”

NOTES:

(1) Analog signals are recorded.
(2) Tag Numbers prefixed by 1D17 are components associated with Unit 1.
Tag Numbers prefixed by D17 are components associated with the common areas of the plant.
(3) GSP = Gas chamber scintillator-photomultiplier
HSP = Halogen cartridge scintillator-photomultiplier
PSP = Particulate filter scintillator-photomultiplier
H = Halogen
P = Particulate

TABLE 12.3-11

ISOKINETIC PROBES

UNIT 1

	Ventl. System	Monitor No.[1]	Isokinetic Probe No.[1]	Duct (inches)	Flow CFM
M14	Containment Vessel and Drywell Purge	1D17K660	1D17N661A	48 x 48	5,000
			1D17N661B	48 x 48	25,000
M15	Reactor Bldg. Annulus Exhaust Gas Treatment	1D17K690A	1D17N691A	14 x 16	400min[2]
		1D17K690B	1D17N691B	14 x 16	400min[2]
M36	Offgas Bldg. Vent. System	1D17K760	1D17N761	30 x 46	16,700[3]
M38	Auxiliary Bldg. Vent. System	1D17K700	1D17N701	30 x 90	29,325

COMMON

M25	Control Room HVAC and Emerg. Recirc. System	D17K770	D17N771	38 x 32	11,540
M31	Radwaste Bldg. Vent. System	D17K720	D17K721A	32 x 90	30,000
			D17K721B	32 x 90	30,000
M33	Intermediate Bldg. Vent. System	D17K730	D17N731	46 x 46	27,400
M40	Fuel Handling Area Vent. System	D17K710	D17N711	40 x 60	30,000

NOTES:

(1) Unit 1 has 1 preceding the number, i.e., 1D17K---.
Common areas have no prefix to the number, i.e., D17K---.
(2) With no recycle to the annulus space, 2,000 CFM is possible.
(3) Based on total Offgas Vent Stack flow.

TABLE 12.3-12

REACTOR BUILDING SUBCOMPARTMENT VENTILATION DATA
M11, M14 SYSTEMS

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Reactor Operation:	5,000	-
642′ level, RWCU F/D backwash rec. tank area (radiation Zone 5 area)	500[1]	0.10 max.
654′ level east RWCU valve area (radiation Zone 5 area)	200[1]	0.04 max.
664′ level, RWCU F/D valve & holding pump room (radiation Zone 5 area)	750[1]	0.15 max.
664′ level, RWCU F/D room (2) (radiation Zone 5 area)	120[1] (each)	0.024 max.
664′ level north, RWCU HX (radiation Zone 5 area)	2,300[1]	0.46 max.
Reactor Shutdown Mode of Operation:	25,000	-
Drywell area	16,500	0.66
Containment free space	8,500	0.34
642′ level, RWCU F/D backwash rec. tank area (radiation Zone 5 area)	500[1]	0.017 max.
664′ level north, RWCU HX (radiation Zone 5 area)	2,300[1]	0.077 max.
654′ level east RWCU F/D valve nest area (radiation Zone 5 area)	200[1]	0.007 max.
664′ level, RWCU F/D valve & holding pump room (radiation Zone 5 area)	750[1]	0.025 max.
664′ level, RWCU F/D room (radiation Zone 5 area)	120[1] (each)	0.004 max.

NOTE:

[1] Exhaust flow rate is a result of M11 Supply Air.

TABLE 12.3-13

RADWASTE BUILDING SUBCOMPARTMENT VENTILATION DATA
M31 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	30,000	-
574'-0" Level:		
Floor drain collection pump room A and B	400 (each)	0.013
Fuel pool sludge decant pump room	200	0.0066
Equipment drain sump pump	200	0.0066
RWCU sludge decant pump room A	100	0.0033
RWCU sludge decant pump room B	200	0.0066
Floor drain sump room	200	0.0066
Condensate sludge decant pump room	400	0.013
Waste sample pump room	600	0.02
Floor drain sample pump room	600	0.02
Chemical waste pump room A and B	600 (each)	0.02
Corridor	1,700	0.0567
Waste collector transfer pump room A and B	400 (each)	0.03
Radiation Zone 5 Areas:		
Fuel pool sludge discharge mixing pump area A and B	400 (each)	0.013
RWCU sludge discharge mixing pump area A and B	400 (each)	0.013
Condensate sludge discharge mix pump area A and B	400 (each)	0.013
Fuel pool F/D backwash settling tank room area A and B	400 (each)	0.013
RWCU F/D backwash settling tank A and B	400 (each)	0.013
Condensate filter backwash settling tank A and B	400 (each)	0.013
602' Level:		
Chemical waste distillate tank area	900	0.03
Detergent drain tank & pump area	300	0.01
Corridor	1,800	0.06
Radiation Zone 5 Areas:		
Spent resin pump area A and B	400 (each)	0.013
Concentrated waste transfer pump area A and B	400 (each)	0.013
Floor drain collection tank area A and B	600 (each)	0.02

TABLE 12.3-13 (Continued)

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Radiation Zone 5 Areas: (Continued)		
Waste collection tank area A and B	700 (each)	0.023
Spent resin tank area A and B	400 (each)	0.013
Concentrated waste tank area A and B	400 (each)	0.013
Chemical waste tank area A and B	700 (each)	0.023
Floor drain sample tank area A	300	0.01
Floor drain sample tank area B	600	0.02
Waste sample tank area A	600	0.02
Waste sample tank area B	300	0.01
623′ Level:		
Process sample room	2,250	0.075
Radwaste control panel area	Air Supplied Only	-
Dry Active Waste Handling Area	3,500	0.1167
Corridor	6,600	0.2200
Loading and drum storage area	3,610	0.1203
Reverse osmosis unit room	300	0.01
Chemical treatment room	300	0.01
Ventilation exhaust equipment room	1,200	0.04
Exhaust system area	1,000	0.033
Valve station	130	0.0043
Radiation Zone 5 Area:		
Chemical waste evaporator rooms A and B	900 (each)	0.0333
Waste demineralizer area	140	0.0047
Floor drain demineralizer area	140	0.0047
Waste cement mixing pump room A and B[1]	400 (each)	0.0133
Waste mixing dewatering tank room A and B	400 (each)	0.0133
646′ Level:		
Filter, precoat pump and tank room	1,540	0.0513
Waste collector filtrate pump room	200	0.0067
Waste collector filtrate room	570	0.0190
Floor drain filtrate pump room	200	0.0067
Floor drain filtrate room	570	0.0190
616′ Level:		
Full drum storage area	2,000	0.0667
Decontamination area	500	0.0167

NOTE:

[1] Either portions or all of the equipment located in these areas are abandoned. However, the M31 System is operational. Therefore, this information is being retained in this table.

TABLE 12.3-14

AUXILIARY BUILDING SUBCOMPARTMENT VENTILATION DATA
M38 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	29,325	-
574′ level - Corridor	1,800	0.061
Radiation Zone 5 Areas		
568′ level: LPCS pump room	2,280	0.078
RHR-A pump room	5,415	0.185
RCIC pump room	2,105	0.072
RHR-C pump room	1,545	0.053
RHR-B pump room	5,420	0.185
HPCS pump room	2,280	0.078
599′ level - Corridor and containment vessel/turbine building water chiller area/ADHR heat exchanger room	9,885	0.337
599′ level - RWCU pump room A	3,175	0.108
RWCU pump room B	915	0.031
614′ level - Steam tunnel	4,000	0.136
620′ level - Northwest corridor	3,615	0.123
Northeast corridor and ventilation supply equipment area	8,240	0.281
- pipe chase access room	910	0.031

TABLE 12.3-15

INTERMEDIATE BUILDING SUBCOMPARTMENT VENTILATION DATA
M33 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	27,400	-
574′ level-General area	900	0.033
Tool Decon Area	1,100	0.040
Tool Storage Area	900	0.033
599′ level-General area	5,400	0.197
Control complex controlled access entry	100	0.004
Electrical equipment room	1,200	0.044
620′ level	8,300	0.303
Annulus exhaust gas treatment rooms (4)	500 (each)	0.018
654′ level-General area & recombiner area	4,800	0.175
Containment vessel & drywell purge (2)	1,000 (each)	0.036
Containment vessel & drywell purge (2)	1,200 (each)	0.044
682′ level-General area	6,000	0.218
Fuel handling area exhaust filter rooms (3)	800 (each)	0.029

TABLE 12.3-16

FUEL HANDLING AREA SUBCOMPARTMENT VENTILATION DATA
M40 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	30,000	-
574' level		
Corridor, north and south	1,000 (each)	0.033
Radiation Zone 5 Areas:		
Control rod drive pump room, north and south	2,500 (each)	0.083
Fuel pool cooling and cleanup circulating pump	300	0.01
Fuel pool cooling and cleanup F/D transfer pump room	200	0.0067
Fuel pool cooling and cleanup F/D backwash rec. tank room	250	0.0083
Intermediate building floor and equipment drain sump pump room	250	0.0083
Postaccident sample room	300	0.01
599' level		
Corridor, north and south	1,000 (each)	0.033
Control rod drive maintenance area	8,100	0.27
Radiation Zone 5 Areas:		
Pool Area: (Cask storage pool, spent fuel pool, Fuel transfer pool)	15,300	0.51
Fuel pool cooling and cleanup heat exchanger	400	0.0133
Fuel pool cooling and cleanup F/D room A, B	200	0.0067
Fuel pool cooling and cleanup F/D room C, D	200	0.0067
Hot I&C repair room	500	0.0167

TABLE 12.3-17

HEATER BAY SUBCOMPARTMENT VENTILATION DATA
M41 SYSTEM

Subcompartment	SUMMER OPERATION[1] Exhaust Flow Rate (CFM)	SUMMER OPERATION[1] Dilution Factor	WINTER OPERATION[2] Exhaust Flow Rate (CFM)	WINTER OPERATION[2] Dilution Factor
Normal Operation	360,000	-	198,000	-
560′ level (Heater area) DC feedwater heater, MCV access area	107,000	0.30	53,500	0.27
580′ level (Hot water heating equipment)	97,000	0.269	48,500	0.245
Rad. Zone 5 areas: DC feedwater heaters and auxiliary boiler evaporator	120,000	0.333	60,000	0.303
600′ level (Hallway)	28,500	0.0792	14,250	0.072
Rad. Zone 5 areas: intermediate feedwater heater, HP feedwater heater	158,500	0.440	79,250	0.400
620′ level (Hallway)	21,500	0.0597	750	0.004
FDW Lube Oil Purifier Room	1,500	0.004	750	0.004
Rad. Zone 5 area: Auxiliary Condenser	166,000	0.461	79,600	0.402
647′ level (Hallway)	14,000	0.0389	400	0.002
Rad. Zone 5 areas:				
Steam seal evaporator	83,000	0.23	39,800	0.20
Feedwater pump area (2)	41,500 (each)	0.11	19,900 (each)	0.10
Feedwater heater area	60,000	0.17	22,400	0.11

NOTES:

(1) With louvers open.
(2) With louvers closed.

TABLE 12.3-18

TURBINE BUILDING SUBCOMPARTMENT VENTILATION DATA
M35 SYSTEM

Subcompartment	SUMMER OPERATION[1] Exhaust Flow Rate (CFM)	SUMMER OPERATION[1] Dilution Factor	WINTER OPERATION[2] Exhaust Flow Rate (CFM)	WINTER OPERATION[2] Dilution Factor
Normal Operation	360,000	-	198,000	-
Condenser bay area (radiation Zone 5)	96,000	0.27	96,000	0.48
Turbine operating floor	180,000	0.50	118,000	0.60
Hot well pump & L.P. heater area	25,500	0.07	25,500	0.129
Condenser vacuum pump area	300	0.001	300	0.002
Sample extraction area	600	0.002	600	0.003
Steam seal exhaust area	24,700	0.07	24,700	0.125

NOTES:

(1) With louvers open
(2) With louvers closed

TABLE 12.3-19

OFFGAS BUILDING VENTILATION EXHAUST SUBCOMPARTMENT VENTILATION DATA
M36 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	15,400	-
(Based on total Offgas Vent Stack flow)	16,700	
548' level		
Radiation Zone 5 areas:		
Turbine power complex		
a. Condensate demineralizer backwash rec. tank	600	0.035
b. Condensate filter backwash rec. tank area	700	0.041
568' level		
Turbine power complex, corridor	2,200	0.131
Turbine power complex, condensate filter pump area	1,500	0.089
Condensate demin. cubicles (6)	500 (total)	0.029
Condensate filter cubicles (8)	500 (total)	0.029
Caustic & acid storage tank room	2,000	0.119
Offgas exhaust plenum area	600	0.035
Radiation Zone 5 area:		
Turbine power complex, Cation regen. tank area	500	0.029
577' level		
(Radiation Zone 5 area) Turbine building, holdup pipe area	1,000	0.059
584' level		
Offgas building, corridor	700	0.041

TABLE 12.3-19 (Continued)

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Radiation Zone 5 areas:		
Offgas cooler condenser room	200	0.011
Offgas regenerator room A and B	250 (each)	0.014
Charcoal absorber rooms	1,000 (Total)	0.059
602’ level		
Radiation Zone 5 areas:		
Offgas building, corridor	1,500	0.089
Desiccant dryer area	450	0.026
After filter and prefilter rooms (3)	200 (each)	0.011
605’ level		
Radiation Zone 5 areas:		
Steam jet air ejector room A and B	800 (each)	0.047
Preheater area	800	0.047
620’ level		
Offgas building, floor area	2,500	0.149
Offgas sample panels	400 (Total)	0.023
624’ level		
Turbine building, lab. hoods	2,000 (Total)	0.119
Turbine building, hydrogen oxygen analyzer area (A and B)	300 (each)	0.017

12.4 DOSE ASSESSMENT

The estimates of exposure use both design radiation zones and the associated occupancy times described in <Section 12.3.1>, along with operational data gathered from similar BWR plants. Design radiation levels used have been developed from conservative assumptions which indicate maximum radiation levels and not those anticipated for normal plant operation.

Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. These assessments, including any reference to Unit 2, are considered historical.

Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993. During normal plant operation, the site ALARA Committee has the responsibility to determine the annual person-rem exposure for the Perry Nuclear Power Plant. This will be done based on anticipated routine and outage conditions, work loads and number of personnel on site.

12.4.1 ESTIMATES OF PERSONNEL OCCUPANCY REQUIREMENTS

Occupancy requirements throughout the plant are based on operating experience and project manpower needs, and were considered in the establishment of the design radiation zones described in <Section 12.3.1>. To estimate occupancy requirements, plant personnel are categorized into five groups according to work function. Feedback information from operating facilities indicates that contract workers will be called upon to do some tasks as indicated in <Table 12.4-3>.

Essentially all of the dose received by contract personnel will be from maintenance activities. <Table 12.4-1> lists the estimated size of each group and the estimated occupancy requirements for each group in Zones I, II and III. Personnel requirements are for both Unit 1 and Unit 2. Estimates were not made for Zones IV and V since routine occupancy is not anticipated in these zones during normal operation or during anticipated operational occurrences.

12.4.2 ESTIMATES OF ANNUAL PERSON-REM DOSES

Annual doses to plant personnel are estimated based on the assumption of 2,000 hours per year work for each employee.

Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. These assessments, including any reference to Unit 2, are considered historical.

Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993. It is anticipated that the general radiation levels for each zone will be less than the design values stated for the zone, although isolated higher levels will exist in certain areas within the zone.

The estimated person-rem doses for the various categories of plant personnel discussed in <Section 12.4.1> are listed in <Table 12.4-2>. <Table 12.4-3> and <Table 12.4-4> list the percentages of exposures by job function and work function, respectively, as reported by operating nuclear power plants for the year 1982 (Reference 1). The total annual dose for plant operation is conservatively estimated to be 598.3 person-rem/unit of which 183.8 person-rem/unit will be received by contract personnel. <Table 12.4-5> lists several of the larger operating BWR facilities and the associated person-rem dose for the years 1980, 1981 and 1982 (Reference 1). This indicates that the person-rem dose does vary considerably but is generally higher than the estimated person-rem dose for PNPP operation.

A further estimate of the person-rem doses has been made by identifying specific tasks anticipated to occur at the plant. Various data from operating plants and in current publications (Reference 1), (Reference 2), (Reference 3), (Reference 4), (Reference 5), (Reference 6), (Reference 7), and (Reference 8) were used to identify these tasks, the manpower effort required to complete each task and the radiation levels associated with performing each task. The guidance of

<Regulatory Guide 8.19> has been followed to complete these dose estimates. <Table 12.4-6>, <Table 12.4-7>, <Table 12.4-8>, <Table 12.4-9>, <Table 12.4-10>, <Table 12.4-11>, and <Table 12.4-12> summarize the person-rem doses per tasks. These tables give person-rem estimates on a per unit basis. <Table 12.4-6> indicates that special maintenance tasks represent a relatively small percentage of the annual person-rem dose. Routine maintenance and refueling make up approximately 70 percent of the annual person-rem dose at operating facilities. Disagreement in the estimated values for PNPP and industry averages can be partially accounted for by design changes at PNPP <Section 12.2.1>, and in the way various tasks have been categorized.

Particular mention should be made regarding feedwater sparger repair. This task is a very special maintenance task and may not occur during the projected life of the plant. However, a person-rem dose has been added for this task under the assumption that it will occur once during the 40 year life of the plant. For other special maintenance tasks it has been assumed that frequency of occurrence will coincide with refueling outages. In actuality, the frequency or special maintenance will be quite variable.

In addition to the facility design aspects that will promote ALARA, both the ALARA-oriented radiation protection program <Section 12.5> and the utilization of applicable operational data and guidance will be very important in ensuring that personnel doses are ALARA.

12.4.3 ESTIMATED INHALATION DOSES

Radiation doses associated with airborne radioactivity have not been analyzed in terms of tasks due to the lack of sufficient industry data. Inhalation doses were estimated using the airborne sources of radioactivity described in <Section 12.2.2>. The ventilation system has generally been designed to move air from areas of unlimited occupancy (no potential airborne radioactivity sources) to areas of limited

occupancy (potential airborne radioactivity sources). Appropriate health physics procedures will be established to measure the radiological conditions in areas with potential airborne contamination, ensuring that radiation doses are maintained ALARA. Where required, respirators will be used to further reduce inhalation doses.

<Table 12.4-13> lists specific areas of the plant where airborne activity may be present in quantities that would result in a measurable dose to the whole body and, as a result of gaseous iodines, a dose to the thyroid. The areas included are the reactor building (during normal operation and during refueling), the fuel handling building, the radwaste building, and the turbine building. For each area man-hours per work function have been estimated using (Reference 4) and (Reference 6). The exposure rates have been determined from specific activities listed in <Section 12.2.2>. Sources of airborne activity will be primarily from valve and pump leakage. During refueling operations the refueling and spent fuel pools will release small amounts of airborne activity to the reactor building and the fuel handling building, respectively. However, as a result of pool cleanup systems, it is anticipated that contributions from these sources will be minimal. The resulting annual person-rem doses per unit are listed in <Table 12.4-13>.

An additional airborne activity source will come from the actuation of the safety/relief valves after an isolation scram (Type 2 event). Type 1 and Type 2 events of steam discharges to the suppression pool are discussed in <Section 12.2>. <Table 12.4-14> gives the resulting doses to personnel in the reactor building as they exit after a Type 2 event. It has been assumed that at the initiation of the isolation scram, an operator is located at the TIP drive floor. The operator egress is at the personnel airlock 180° from the TIP floor area at the same elevation. Operator egress is conservatively assumed to take four minutes.

The dose rates used are those calculated for the immediate area above the suppression pool. Normal ventilation is assumed and airborne concentrations are not corrected for plate out on walls. The dose assessment methodology including pool retention factors and average radiohalogen carry-over factors as referred to in <Section 12.2.2.1>.

No dose calculations regarding Type 1 events are presented since resulting personnel doses would be negligible.

12.4.4 ESTIMATED ANNUAL DOSE OUTSIDE THE NUCLEAR FACILITY AT THE BOUNDARY OF THE RESTRICTED AREA

Potential direct radiation doses to individuals outside the nuclear facility will arise from the following:

a. Skyshine and direct dose from turbines

b. Direct dose from stored radwaste

c. Direct dose from the external surfaces of buildings

d. Dose from the gaseous radioactive plume

12.4.4.1 Skyshine and Direct Dose from Turbines

The dose analyses for normal operation of both units were based on an 80 percent load factor, and 50 and 24 percent occupancy factors for offsite and onsite exposures, respectively. For distances beyond 300 feet, a single lumped source was assumed in the turbine building; for distances less than 300 feet, all major sources were considered separately. The resultant doses at selected locations are given in <Table 12.4-15> and a curve of the dose rate (mrem/hr) versus distance from the turbine building is presented in <Figure 12.3-12>.

Projection of doses to non-plant personnel assumed that non-plant personnel working at Unit 2 could receive a radiation dose when Unit 1 is in commercial operation. A study (Reference 9) has analyzed several two unit stations where construction personnel are completing a unit next to an operating one. This study indicated that construction personnel are likely to receive radiation doses that are not distinguishable from background radiation levels. For the PNPP, however, a conservative dose analysis for non-plant personnel has been performed based on the following assumptions:

a. The activities scheduled during this period on Unit 2 can be divided into three basic categories: completion of construction, startup/testing and site engineering. The total duration of these activities is scheduled to be 74 months between commercial operation of Unit 1 and Unit 2.

b. The relative orientation of the structures is as shown on <Figure 1.2-1>.

c. The startup/testing activities on Unit 2 consist of such items as fuel loading, preoperational testing and startup and power testing. The bulk of this work will be accomplished in the last 24 of the 74 month period between commercial operation of Unit 1 and Unit 2.

d. The occupational composition and size for the startup/testing work for a single shift is conservatively assumed as follows:

    1. Engineers - 25 required

    2. Electricians - 20 required

    3. Pipe fitters - 20 required

    4. Crane operators - 4 required

5. Operating engineers - 4 required

6. Utility personnel - 20 required

7. Laborers - 10 required

8. Technicians - 50 required

e. There will be three 9-hour shifts per day.

f. During any shift, the following time schedules are assumed:

1. A portion of the work force (25 percent) spends 5 hours on the turbine building operating floor, 2 hours outside all plant structures and 2 hours inside plant structures. It is conservatively assumed that while inside a plant structure, the individuals are shielded by a minimum of two feet of concrete and are located at a distance of 300 feet from the Unit 1 turbine building.

2. The remainder of the work force spends 2 hours outside all plant structures and 7 hours inside plant structures.

g. The completion of construction activities on Unit 2 consists of installation of equipment and piping, electrical wiring, closing of concrete construction openings, completion of painting and architectural items and completion of final roadways and landscaping in the area around Unit 2.

h. The size of the work force needed for the completion of Unit 2 construction, based on present schedule resource loading, is given on <Figure 12.4-1>.

i. The work force is assumed to be composed of the occupational crafts listed in <Table 12.4-16>.

j. During a working day, the following time schedules are assumed:

   1. A portion of the work force (40 percent) spends 2 hours outside all plant structures and 7 hours inside plant structures. It is conservatively assumed that while inside a plant structure, the individuals are shielded by a minimum of two feet of concrete and are located at a distance of 300 feet from Unit 1 turbine building.

   2. The remaining 60 percent of the work force is conservatively assumed to spend all 9 hours of the work shift outside plant structures.

k. The site organization is a field engineering group housed onsite to complete portions of the final design and handle field construction problems. The estimate of the work force does not include operations, radiation protection, maintenance, and I&C personnel since their occupational exposures are discussed in <Section 12.4.2>.

l. The size of the site organization work force is shown on <Figure 12.4-2>.

m. During the working day, the following schedule is assumed: the average worker spends 5 hours inside the warehouse/office, 2 hours outside all plant structures and 2 hours inside plant structures.

The radiation dose to the startup/test personnel, construction personnel and site organization personnel will arise from radiation levels at the external surfaces of buildings <Section 12.4.4.3>, from the gaseous radioactive plume <Section 12.4.4.4> and from the predominant source

(skyshine). A summary of direct doses from these sources is given in <Table 12.4-17>. The resulting person-rem information is summarized in <Table 12.4-18>.

12.4.4.2 Direct Doses From Stored Radwaste

The interim/temporary storage of radioactive wastes were evaluated for compliance with the NRC <Generic Letter 81-38>. Facilities and structures utilized for the interim storage of radioactive waste have been designed to assure that such storage does not result in an exposure (direct and skyshine) to the nearest site boundary greater than an additional 1 millirem per year above normal plant operations.

12.4.4.3 Direct Doses From the External Surfaces of Buildings

All external walls of buildings have been designed to attenuate radiation sources from within to comply with Zone I conditions (0.5 mrem/hr). For calculational purpose, an expected radiation dose of 0.25 mrem/hr has been used. It has been assumed that the dose rate at the surface of the building decreases inversely proportional to the distance from the wall. In addition to distance, air attentuation also will decrease the doses. <Table 12.4-18> lists distances from the building and associated direct dose rates and annual doses out to the exclusion boundary. Assumed occupancies are also given in <Table 12.4-19>.

12.4.4.4 Doses From Gaseous Radioactive Plume

A small direct dose will result from noble gases released from Unit 1 and Unit 2. The methodology used to calculate this dose has been taken from <Regulatory Guide 1.109>. The meteorological data has been taken from <Section 2.3> and the quantity of nuclides released from <Section 11.3.3>. An 80 percent load factor and 50 and 24 percent

occupancy factors for offsite and onsite exposures, respectively, were again assumed in calculating the expected annual dose from this source <Table 12.4-20>.

12.4.5 REFERENCES FOR SECTION 12.4

1. U.S. Nuclear Regulatory Commission, <NUREG-0713>, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1982," December 1983.

2. U.S. Nuclear Regulatory Commission, <NUREG-0322>, "Ninth Annual Occupational Radiation Exposure Report, 1976," October 1977.

3. U.S. Nuclear Regulatory Commission, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants, Design Stage Man-rem Estimates," <Regulatory Guide 8.19>, May 1978.

4. Vance, J., Weaver, C. L., Lepper, E. M., "A Preliminary Assessment of the Potential Impacts on Operating Nuclear Power Plants of a 500 mrem/hr Occupational Exposure Limit," Report to the Nuclear Regulatory Staff by the Atomic Industrial Forum, April 1978.

5. Verna, B. J., "Radioactive Maintenance, Parts 1-4," Nuclear News, September 1978, November 1975, January 1976, March 1976.

6. General Electric Information Document, "Mark III Containment Dose Reduction Study," 22A5718, 1/29/80.

7. Murphy, T. D. et al., "Occupational Exposure at Light Water Cooled Power Reactors, 1969-1975," U.S. Nuclear Regulatory Commission, <NUREG-0109>, August 1976.

8. Pelletier, C. A., et al., "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plant," September 1974.

9. Endres, G. W. R., Garcia, W. T., and Shipler, D. B., BNWL-SA-6103, "Dose to Construction Workers at Operating Reactor Sites," presented at ACRS Meeting on Radiological and Site Evaluation, Washington, D.C., July 11, 1978.

TABLE 12.4-1

ESTIMATED MANPOWER NEEDS AND OCCUPANCY REQUIREMENTS, UNIT 1 AND UNIT 2

Group	Number	Zone I (%)	Zone II (%)	Zone III (%)
Administrative	100	97	3	0
Radiation Protection	90	80	18.5	1.5
Technical Section	200	93	6	1
Operations Section	200	86	13	1
Maintenance Section	200	70	28.5	1.5
Contractors	300	75	24	1
Total	1,090			

TABLE 12.4-2

PERSON-REM ESTIMATES FOR NORMAL PLANT OPERATIONS, ANTICIPATED OPERATIONAL OCCURRENCES AND ROUTINE MAINTENANCE, UNIT 1 AND UNIT 2

Group	Zone I (0.5 mrem/hr)	Zone II (2.5 mrem/hr)	Zone III (25 mrem/hr)
Administrative	48.5	0	0
Radiation Protection	36.0	41.6	33.8
Technical Section	93.0	30.0	50.0
Operations Section	86.0	65.0	50.0
Maintenance Section	70.0	142.5	75.0
Contractors	112.6	180.0	75.0

Subtotals: 828.9 person-rems (station personnel)
367.6 person-rems (contract personnel)

Total: 1,196.5 person-rems

TABLE 12.4-3

BOILING WATER REACTORS
PERCENTAGES OF EXPOSURE BY JOB FUNCTION[1]

Job Function	Utility	Contractor	Total
Maintenance	22.6	55.8	78.4
Operations	6.4	0.9	7.3
Radiation Protection	2.9	3.1	6.0
Supervisory	1.6	0.4	2.0
Engineering	2.6	3.7	6.2
Totals	36.1	63.9	100

NOTE:

[1] <NUREG-0713> (Reference 1).

TABLE 12.4-4

BOILING WATER REACTORS
PERCENTAGES OF EXPOSURE BY WORK FUNCTION[1]

Work Function	Utility	Contractor	Total
Reactor Operations	8.5	2.5	11
Routine Maintenance	4.0	5.0	9
Inservice Inspection	2.8	7.2	10
Special Maintenance	0.4	1.6	2
Waste Processing	0.5	0.5	1
Refueling	34.7	32.3	67
Totals	50.9	49.1	100

NOTE:

[1] <NUREG-0713> (Reference 1).

TABLE 12.4-5

YEARLY OPERATIONAL PERSON-REM FOR SELECTED BWR PLANTS[1]

		1980	1981	1982
1.	Dresden 1, 2 & 3	2,105	2,802	2,923
2.	Monticello	531	1,004	993
3.	Nine Mile Point	591	1,592	1,264
4.	Peach Bottom 2 & 3	2,302	2,506	1,977
5.	Quad Cities 1 & 2	4,838	3,146	3,757
6.	Vermont Yankee	1,338	731	205
	Ave. Person-rem/Unit	1,170	1,178	1,112

NOTE:

[1] <NUREG-0713> (Reference 1).

TABLE 12.4-6

SUMMARY OF TOTAL OCCUPATIONAL RADIATION EXPOSURE ESTIMATES BY TASK

Function	Dose (person-rems/yr-unit)	Percentage of Total person-rem dose
Routine operation	43	11
Routine maintenance	38	10
Waste processing	3	1
Refueling	251	66
Inservice inspection	39	10
Special maintenance	6	2
Total person-rems/yr-unit	380	

TABLE 12.4-7

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE OPERATIONS AND SURVEILLANCE

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
Control room	0.1	6,000	2	-	1.2
Walking and checking:					
Turbine and feedwater	100	0.1		1/shift	10.0
heat exchanger	1.0	1.0	1	1/shift	1.0
Containment cooling system	1.0	1.0	1	1/day	0.36
Standby liquid control system	1.0	1.0	1	1/day	0.36
ECCS and process equip	1.0	1.0	1	1/shift	1.0
C&I panels and equip in containment	1.0	1.0	1	1/shift	1.0
Fuel pool system	1.0	0.4	1	1/day	0.1
RWCU	1.0	0.5	1	1/shift	0.6
CRD system	1.0	0.5	1	1/shift	0.6
Recirc flow control	1.0	0.6	1	1/day	0.22
Misc auxiliary building	1.0	1.0	1	1/shift	1.0
	100	0.1	1	1/shift	10.0
Traversing incore probe system	10	0.1	1	1/shift	1.2
Misc. in containment	1.0	1.0	1	1/day	0.4
Instrument calibration in containment	1.0	0.6	1	1/week	0.03

TABLE 12.4-7 (Continued)

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
Radiochemistry	1.0	1.0	2	1/day	0.73
Radiation protection surveys	1.0	4.0	1	1/day	1.46
	15	1.0	1	1/day	5.48
	100	0.5	1	1/week	2.6
Sample stations in reactor building	5.0	0.5	1	1/shift	2.7
Other local samples	5.0	0.1	1	1/day	0.15
Remote sampling	1.0	0.3	1	1/day	0.1
Containment personnel lock	1.0	0.05	3	1/shift	0.16
Total					43

TABLE 12.4-8

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE MAINTENANCE

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
RWCU filter precoat	1.0	2.0	1	1/week	0.1
RWCU pump and valve	150	1.5	2	1/week	23.4
TIP system	10	2.0	2	1/month	0.5
CRD	2.5	10.0	1	1/week	1.3
HVAC systems	1.0	8.0	1	1/week	0.4
Sample stations	5.0	1.5	1	1/day	2.7
Misc. auxiliary	2.0	4.0	2	1/week	0.8
building pumps (LPCS, HPCS, RCIC, etc.)	25.0	0.2	2	1/week	0.5
Feedwater and	25.0	0.5	2	1/week	1.3
condensate pumps valves and heat exchangers	1.0	1.5	2	2/weeks	0.3
Condensate demineralizer and heat exchangers filters	1.0	1.5	2	1/day	1.1
Total					32.4

TABLE 12.4-9

OCCUPATIONAL DOSE ESTIMATES DURING WASTE PROCESSING

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
Operation of liquid waste processing system	0.2	112	1	1/week	1.1
Operation of solid waste processing system	1.0	30	1	1/week	1.5
Total					3

TABLE 12.4-10

OCCUPATIONAL DOSE ESTIMATES DURING REFUELING

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
Reactor pressure vessels head and internals - removal and installation	60	40	10	1/year	27
Refueling operations	10	100	15	1/year	18.0
Fuel sipping	2.0	100	2	1/year	0.4
CRD replacement	260	35	5	1/year	54
CRD repair	15	200	6	1/year	20
Low Power range monitor	95	20	4	1/year	9
MSIV repair	75	100	6	1/year	52
RHR system	50	27	8	1/year	11
RWCU pump	180	35	3	1/year	22
RWCU valve and heat exchanger	110	45	6	1/year	36
Turbine inspection	3	80	10	1/3 years	0.8
Turbine overhaul	3	250	20	1/20 years	0.75
Total					251

TABLE 12.4-11

OCCUPATIONAL DOSE ESTIMATES DURING INSERVICE INSPECTION

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
Providing access: installation of platforms, ladders, etc., removal of thermal insulation	200	10	4	1/year	8
Drywell	100	50	6	1/year	30
Reactor building	5	50	2	2/year	1
Total					39

TABLE 12.4-12

OCCUPATIONAL DOSE ESTIMATES DURING SPECIAL MAINTENANCE

Activity	Average Dose Rate (mrem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (person-rem/ year)
Feedwater sparger	800	60	244	1/40 years	6
Special cleaning evolutions or one-time plant modifications of radiological significance as defined by the Perry staff.					See Note[1]
Total					6

Note:

[1] No dose is specified for activity. Dose will be estimated on a case-by-case basis based upon the specific radiological conditions, task duration, and manpower requirements to accomplish the activity.

TABLE 12.4-13

PERSONNEL RADIATION DOSES FROM AIRBORNE ACTIVITY

	Routine Surveillance	Routine Operation	Instrument Calibration	Main tenance	Refueling	ISI	Total man-hrs/yr	Dose Rates (rem/hr) WB	Dose Rates (rem/hr) Thyroid	Yearly Person-rem Doses (person-rem/yr-unit) WB	Yearly Person-rem Doses (person-rem/yr-unit) Thyroid
Reactor building (during operation)	5,000	1,100	70	2,200	---	- -	8,370	1.8×10^{-4}	4.3×10^{-3}	1.5	36.0
Reactor building (refueling)	---	---	40	2,600	1,500	440	4,580	Neg[1]	3.8×10^{-4}	Neg	1.7
Fuel handling building	150	---	20	300	1,500	- -	1,970	Neg	1.5×10^{-5}	Neg	3.0×10^{-2}
Radwaste building	150	7,400	20	300	---	- -	7,870	Neg	1.2×10^{-4}	Neg	9.4×10^{-1}
Turbine building	1,200	---	40	5,000	---	- -	6,240	2.5×10^{-4}	9.0×10^{-5}	1.6	5.6×10^{-1}
Total										3.1	37.7

NOTE:

[1] Negligible.

TABLE 12.4-14

SAFETY/RELIEF VALVE DISCHARGERS
DOSE FOR TYPE 2 EVENT

Organ	Dose
Whole body + eye (γ)	150 mrem/event
Skin (β)	440 mrem/event
Thyroid	.87 mrem/event

TABLE 12.4-15

ESTIMATED SKYSHINE DOSES[1]

Distance (ft)	Occupancy Factor[2]	Dose (mrem/yr)
500	0.19	258
1,500	0.19 (40 hrs/wk)	19
2,900 (Exclusion boundary)	.4 (50% occupancy)	1.6

NOTES:

[1] Two units operating.
[2] Plant factor (80 percent) times percentage occupancy.

TABLE 12.4-16

WORK FORCE BY OCCUPATIONAL CRAFTS

Occupational Craft	Approximate Percentage of Work Force (%)
Boilermakers	7
Electricians	18
Ironworkers	9
Pipe fitters	15
Painters	7
Laborers	11
Operating engineers	6
Carpenters	9
All other crafts	18
Total	100

TABLE 12.4-17

SUMMARY OF DIRECT DOSES[1]

Source	Distance (ft) 500	1500	2,900 Exclusion Boundary
Skyshine	258	19	1.6
Surfaces of buildings	7.2×10^{-1} See Note[2]	6.3×10^{-2} See Note[2]	2.6×10^{-3}
Radioactive plume	13.6	7.8	1.1

NOTES:

[1] Doses in mrem/yr.
[2] Estimated from <Table 12.4-18>.

TABLE 12.4-18

DOSE TO NON-PLANT PERSONNEL

Function/Location	Direct Dose Buildings	Radioactive Plume	Skyshine	Daily Dose	Dose Per Job Duration
A. Startup/test	(mrem/hr)	(mrem/hr)	(mrem/hr)	(mrem/day)	(person-rem)
1. 25% of work force					
5 hrs/day in turbine building[1]	Negligible	Negligible	0.046	0.23	2.0
2 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0032	0.028
2 hrs/day outside plant[3]	0.0008	0.025	0.16	0.37	3.2
2. 75% of work force					
2 hrs/day outside plant[3]	0.0008	0.025	0.16	0.37	9.6
7 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0112	0.29
				Total	15.1 person-rem
B. Construction workers[5]					
1. 40% of work force					
2 hrs/day outside plant[4]	0.00026	0.008	0.06	0.14	111.8
7 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0112	8.9

TABLE 12.4-18 (Continued)

Function/Location	Direct Dose Buildings	Radioactive Plume	Skyshine	Daily Dose	Dose Per Job Duration
2. 60% of work force 9 hrs/day outside[4]	0.00026	0.008	0.06	0.61	731.0
				Total	851.7 person-rem
C. Site Organization[5]	(mrem/hr)	(mrem/hr)	(mrem/hr)	(mrem/day)	(person-rem)
2 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0032	2.8
2 hrs/day outside plant[3]	0.0008	.025	.16	.37	327.6
5 hrs/day at warehouse office[6]	0.00026	0.008	.02	.14	124.0
				Total	454.4 person-rem

NOTES:

(1) Assume Unit 2 turbine is 666 feet from Unit 1 turbine.
(2) Assume 300 feet from any radiation source plus 2 feet concrete shielding.
(3) Assume 300 feet from any radiation source.
(4) Assume 600 feet from any radiation source.
(5) Values based on one 9 hour shift per day, 5 days per week.
(6) Assume 600 feet from radiation source for direct dose and plenum, and 1,000 feet for skyshine.

TABLE 12.4-19

DIRECT DOSE FROM THE EXTERNAL SURFACES OF BUILDINGS

Distance (ft)	Dose Rate (mrem/hr)	Occupancy Factor[1]	Annual Dose (mrem/yr)
100	2.0×10^{-3}	0.19	3.3
300	7.8×10^{-4}	0.19	1.3
600	2.6×10^{-4}	0.19	4.3×10^{-1}
1,200	5.6×10^{-5}	0.19	9.3×10^{-2}
2,000	8.0×10^{-6}	0.19	1.3×10^{-2}
2,900 (Exclusion Boundary)	7.3×10^{-7}	0.4	2.6×10^{-3}

NOTE:

[1] Plant factor (80%) times percentage occupancy.

TABLE 12.4-20

DOSE FROM GASEOUS RADIOACTIVE PLUME

	Distance (ft)		
	500	1,500	2,900
Expected dose rate (mrem/hr)	8.0×10^{-3} See Note[1]	2.2×10^{-3} See Note[2]	7.0×10^{-4} See Note[3]
Expected annual dose (mrem)	13.6	7.8	1.1

NOTES:

[1] A X/Q at 200 meters is used.
[2] A X/Q at 400 meters is used.
[3] A X/Q at 800 meters is used.

12.5 RADIATION PROTECTION PROGRAM

12.5.1 ORGANIZATION

The Manager, Radiation Protection Section is designated as the Radiation Protection Manager as defined in <Regulatory Guide 1.8>. The Radiation Protection Manager is responsible for directing all activities associated with radiation protection and other radiological control services required to support plant operation and maintenance activities. This includes all radiation protection activities and for conducting the plant radiological survey activities required to ensure that personnel exposure to radiation and radioactive materials is maintained within regulatory guidelines and that such exposure is kept as low as reasonably achievable (ALARA). If the Manager, Radiation Protection Section, does not meet the Radiation Protection Manager qualifications specified in <Regulatory Guide 1.8>, an individual who meets the Radiation Protection Manager qualifications specified in <Regulatory Guide 1.8>, shall be designated as the technical specification qualified Radiation Protection Manager, who shall be responsible for all of the aforementioned activities for which the Radiation Protection Manager is responsible, including reviewing, approving and signing/countersigning all associated documents. The Radiation Protection Manager has direct reporting authorization to the General Plant Manager and/or the Site Vice President and shall have sufficient operational freedom to ensure independence from operating pressures in order to carry out radiation protection duties. The Manager, Radiation Protection Section is responsible for staffing, budgeting, department coordination activities, and other non-technical specification Radiation Protection Manager related duties. The Manager, Radiation Protection Section, reports to the General Plant Manager.

The Radiation Protection Supervisors are responsible for the oversight of the Radiation Protection Technicians and the implementation of the operational Radiation Protection Program.

The Radiation Protection Technicians perform the various radiological surveys and associated analysis to ensure compliance with the radiation protection program. At least one Radiation Protection Technician is provided for each shift.

Qualification and training requirements for the Radiation Protection Section's supervisory positions are described in <Section 13.1.2.2>, <Section 13.1.3.2>, and <Section 13.2.3>.

12.5.2 EQUIPMENT, INSTRUMENTATION AND FACILITIES

12.5.2.1 Facilities

Radiation protection and chemistry facilities are located on Elevations 599’0” and 620’6” of the Control Complex. These facilities include but are not limited to the following rooms and areas at Elevation 599’0”:

a. Oil Lab

b. Chemistry Office

c. (Deleted)

d. Chemistry Counting Room

e. Low Level Chemistry Laboratory

f. High Level Chemistry Laboratory

g. Respirator Cleaning

h. (Deleted)

i. Men’s and Women’s Locker Rooms and Lavatories

j. (Deleted)

k. Personnel Decontamination Room

l. Lunch Room

m. (Deleted)

n. Respirator Issue Room

See <Figure 1.2-4> for the arrangement of these facilities.

These facilities include, but are not limited to the following rooms and areas at elevation 620’-6”:

a. Health Physics Storage & Frisking Room

b. Health Physics and Radiation Protection Offices and Support Facilities

c. Audio/Visual Room

d. Whole Body Counting Room

e. RRA Access Control Point

See <Figure 1.2-5> for the arrangement of these facilities.

Portable radiation survey equipment, airborne radiation monitoring equipment and other miscellaneous equipment, will be stored in the Radiation Protection Service Room, and other areas designated by Radiation Protection. Respiratory protective equipment will be repaired at locations determined by Radiation Protection and cleaned in the Respirator Cleaning and Issue Area.

Locker and lavatory facilities for men and women shall contain personal effects lockers for workers as well as toilet, washroom and shower accommodations. Protective clothing may be issued from clothing storage areas in the plant and/or any local access control point, as may be warranted. The protective clothing storage areas will be located at various places throughout the plant.

Health physics sample counting equipment is readily available and is generally located in the Health Physics Storage & Frisking Room. Equipment includes beta counters, an alpha counter, an iodine counting system, and radiation check sources and equipment.

The Radiochemistry Low Level and High Level Laboratories are adjacent to the Chemistry Counting Room. Analysis capabilities will include gamma isotopic analysis, beta and alpha low level analysis and other miscellaneous special sample counting analyses.

Personnel decontamination will be performed in the Radiation Protection Personnel Decontamination Room which is designed for decontamination. Standup showers, hand sinks and other personnel decontamination equipment are located in this room.

Contaminated equipment may be decontaminated in the small tool decontamination facility located in the Service Building Hot Shop. Shop equipment necessary for proper decontamination, e.g., Freon hi-pressure ultrasonic degreasers, liquid abrasive bead blaster, wash sinks, etc., will be provided. Ventilation systems will be designed to maintain a habitable environment for personnel performing decontamination. Drainage from the decontamination areas will be collected or directed to radwaste.

12.5.2.2 Access and Egress to Radiation Protection Controlled Areas

Access to and from the plant shall normally be via the Health Physics RRA Access Control Point, through the double doors and the Service Building hall. Additional access points may be established as deemed necessary. Contaminated materials and radioactive samples are normally brought to the Service Building Hot Shop through the Service Building hall. Contamination detection instrumentation is strategically located for personnel contamination surveys.

12.5.2.3 Health Physics Instrumentation

12.5.2.3.1 Laboratory Instruments

Instrumentation located in the Counting Room, Alternate Counting Room, Health Physics Storage & Frisking Room, Low Level Chemistry Laboratory, and High Level Chemistry Laboratory, will allow radiation protection personnel to ascertain the radioactive material concentrations in survey samples. Typical samples would be: contamination survey smears, airborne survey particulate filters and charcoal halogen cartridges. Typical laboratory instruments are listed on <Table 12.5-1>. Each counting system will be checked and calibrated at regular intervals with radioactive sources traceable to National Institute of Standards and Technology (NIST) in accordance with Health Physics and Chemistry Instructions. Counting efficiency, background count rates and detector voltage settings will be checked periodically. Instrumentation will also be calibrated after repair.

12.5.2.3.2 Portable Survey Instruments

Portable instruments are normally stored in the Health Physics Storage & Frisking Room and, as required, at any inplant access control point for plant maintenance and/or repair. These instruments allow radiation protection personnel to perform alpha, beta, gamma, and neutron surveys, for area radiation, airborne and surface contamination monitoring.

Each portable instrument will be calibrated according to Health Physics Instructions when in use, or prior to use after repair. Sufficient quantities of portable instrumentation will be available to permit calibration, maintenance or repair to instruments without causing a shortage of operational equipment. Typical portable equipment is listed in <Table 12.5-2>.

A large, heavily shielded, self-contained, multi-source calibrator is provided for calibrating gamma dose rate instrumentation. Other sources will be provided as required. Instruments may also be calibrated by a qualified consultant. All sources used for calibration will be traceable to NBS.

12.5.2.3.3 Personnel Monitoring Instruments

Personnel monitoring shall be provided by use of thermoluminescent dosimeters (TLDs), direct reading pocket ionization dosimeters and/or direct reading electronic dosimeters. All personnel entering a Radiologically Restricted Area will be issued personnel monitoring. TLDs will be used to measure exposure to beta-gamma radiation. This badge will contain thermoluminescent chips or cards with suitable energy filters. TLDs will be evaluated and processed by a processor holding a current National Voluntary Laboratory Accreditation Program (NVLAP) accreditation at least annually for personnel records. Personnel doses will be ascertained as prescribed by Health Physics Instructions. The exposure history established by the TLD readings shall constitute the official record of personnel exposure at PNPP. To meet the requirements of <Regulatory Guide 8.4>, direct reading dosimeters may constitute the official record when used for the measurement of x-ray or gamma radiation for plant visitors who must enter a Radiologically Restricted Area.

Anytime an individual is expected to enter a neutron radiation area, neutron dose should be calculated by using a methodology of determination of neutron dose equivalent rates as measured by a neutron rem meter instrument and known occupancy times or the neutron to gamma ratio methodology until the TLD results for the period of time for the exposure is received.

Direct reading dosimeters are issued to personnel as necessary for indication of dose. These dosimeters provide “up-to-the-minute”

indication of radiation exposure. These dosimeters may also be used to monitor the extremities. The dosimeters will be calibrated semi-annually, or anytime damage is suspected.

TLD results greater than 100 millirems are compared to the pocket dosimeter readings upon receipt. Discrepancies between the pocket dosimeter readings and TLD results of greater than 25 percent are evaluated by the Radiation Protection Section.

Annually or as required by the Radiation Protection Manager, an analysis of dose by task is performed to determine which operations or tasks can be modified to reduce exposure.

Personnel contamination survey instruments shall include Geiger-Mueller friskers and portal monitors. These instruments will be calibrated according to Health Physics Instructions at least annually when in use, or prior to use after repair. Personnel internal exposures will be evaluated by a bioassay program as described in <Section 12.5.3.6.2>.

Typical personnel monitoring instruments are listed on <Table 12.5-3>.

12.5.2.3.4 Radiation Protection Equipment

Portable air samplers are used to determine airborne radioactive material concentrations. Portable air samplers will be calibrated for flow semi-annually in accordance with <Regulatory Guide 8.25> and Health Physics Instructions. Typical surveys will be performed for particulate and radioiodine airborne concentrations.

Portable continuous air monitors will be used in various plant locations, to provide local information and trending data. Alarm setpoints are variable in accordance with Health Physics Instructions. Audible and visual alarms are provided to warn local personnel of airborne radioactive concentrations in excess of specified limits.

Respiratory equipment will be provided and stored in the Respirator Cleaning and Issue Area or any remote controlled access point in the plant, as required. Emergency respiratory equipment shall be stored at strategic locations within the plant. The equipment will be maintained and used in accordance with applicable plant procedures and Health Physics Instructions. These instructions are prepared in accordance with <10 CFR 20>.

Protective clothing will be provided for personnel working in contaminated areas. Specific requirements for clothing will be prescribed by radiation protection personnel based on actual or anticipated radiological conditions. An adequate inventory of protective clothing will be maintained and available at in-plant clothing storage areas and access control points. This clothing includes: lab coats, coveralls, booties, hoods, rubber overshoes, rubber gloves, cotton glove liners, and waterproof suits. All respiratory equipment complies with <10 CFR 20>.

Typical contamination control supplies will include: cleaning cloths, drum liners, plastic bags, wet/dry vacuum cleaners, step-off pads, masking tape, radiation tape, radiation rope, various signs, mops, sponges, smears, and other supplies necessary for good radiological housekeeping.

A typical listing of radiation protection equipment is provided in <Table 12.5-4>.

12.5.2.3.5 Other Radiation Protection Instruments

The Area Radiation Monitoring system (ARMS) is installed in areas where it is desirable to have constant dose rate indication. Area monitors display the dose rates of strategic plant locations locally and on Control Room panel modules. The Area Monitors provide audible and visual alarm indication when a preset dose rate is exceeded. Airborne

Radioactivity Monitors (ABRMs) are provided for strategic plant locations when personnel exposure to airborne gaseous and particulate radionuclides and radioiodine may be anticipated.

12.5.2.3.6 Emergency Equipment

Sufficient radiation protection equipment will be available to personnel responding to an accident. The equipment includes dosimeters, portable survey meters, respiratory protection devices, protective clothing, and air samplers. The equipment will be stored in strategic locations in the plant and the Emergency Response Facilities. This equipment will be inventoried and checked for operability at regular scheduled intervals.

12.5.3 HEALTH PHYSICS INSTRUCTIONS

Adherence to PNPP procedures will help ensure that personnel radiation exposures are kept ALARA, and within the limits of <10 CFR 20>. Plans and procedures used at PNPP will meet the criteria provided in <Regulatory Guide 1.33>, <Regulatory Guide 8.2>, <Regulatory Guide 8.8>, and <Regulatory Guide 8.10>.

12.5.3.1 Radiation and Contamination Surveys

Radiation Protection personnel will normally perform radiation and contamination surveys of all accessible areas in the plant. The surveys will be performed on an appropriate frequency, depending on the probability of radiation and contamination levels changing, and the frequency with which the areas are occupied. Surveys related to specific operations and maintenance activities may be performed prior to, during, and/or after the activity, based on information required to keep radiation exposures ALARA.

12.5.3.2 Procedures and Methods Ensuring ALARA

<Section 12.1.3> describes the operational considerations to keep radiation exposures ALARA.

Procedures and instructions address the requirements of <10 CFR 20>, and <Regulatory Guide 1.33>, <Regulatory Guide 8.2>, <Regulatory Guide 8.8>, and <Regulatory Guide 8.10>, as discussed in <Section 1.8>.

Records to keep radiation exposures ALARA are maintained in accordance with <Regulatory Guide 8.7>.

Methods to maintain exposures ALARA will not only be included in Radiation Work Permits, but will also be contained in applicable procedures. Some examples of methods that will be used to maintain exposures ALARA are discussed for the following operations:

12.5.3.2.1 Refueling

After the reactor coolant system is depressurized, it will be degassed and sampled to verify that the gaseous activity is minimized prior to removing the reactor head. After flooding the reactor well and the refueling pool, purification of the pool water shall be continued to maintain minimal radioactivity in the water and, therefore, radiation exposures ALARA.

Movement of spent fuel assemblies will be done with the assemblies under a sufficient depth of water to provide shielding to keep radiation exposures ALARA.

12.5.3.2.2 Inservice Inspection

Review of system drains; maintenance inspection histories, photographs, slides, and radiological survey reports may be required as preparation

for entering radiologically restricted areas. RWPs, if required, will be used to specify the radiological protection measures required to keep personnel exposures ALARA.

12.5.3.2.3 Radwaste Handling

Radiation exposure to personnel handling radwaste will be ALARA due to plant design. Liquids, spent filter media and resins are remotely processed in shielded cubicles.

12.5.3.2.4 Spent Fuel Handling, Loading and Shipping

All movements of spent fuel will be done under at least eight feet of water to provide shielding and cooling. The water is purified to reduce the concentration of radioactive materials. After the fuel is loaded into the shipping cask, the cask will be decontaminated once it is removed from the pool.

12.5.3.2.5 Normal Operations

Major radiation hazards in the plant are minimized by plant design. Equipment and piping containing large quantities of radioactive materials are housed in shielded cubicles and pipe chases. Most equipment is operated remotely to keep operator radiation exposures ALARA.

An Area Radiation Monitoring (ARM) system is provided so personnel can move about and work in the plant with reasonable assurance that radiation levels are below those requiring special monitoring precautions. The ARM system indicates, alarms and records abnormal gamma radiation levels in areas where radioactive materials may be present, stored, handled, or inadvertently introduced.

The plant ventilation system is designed to keep areas of possible radioactive contamination at a negative air pressure to minimize the spread of contamination. Airflow from most areas is monitored for particulate, iodine and gaseous radioactivity. The Airborne Radioactivity Monitoring system indicates, alarms and records abnormal airborne radioactivity levels, allowing personnel to move about and work in the plant with reasonable assurance that airborne radioactivity concentrations are below those requiring special monitoring.

12.5.3.2.6 Routine Maintenance

All maintenance work at PNPP that involves systems that contain, collect, store, or transport radioactive materials and may cause radiation exposure will require an RWP, as determined by Radiation Protection, based on currently determined radiation, contamination or airborne activity levels. The RWP will specify radiological hazards associated with a job and the protective measures required for performing the job.

When applicable, procedures will specify portions of radioactive systems or components which are to be isolated, flushed and/or drained. This will reduce the radiation levels in the maintenance area.

Where applicable, special tools will be used for remote handling of components. This will help keep radiation exposures ALARA by increasing the distance between the workers and the sources of radiation, and by decreasing the workers exposure time. Temporary shielding will be used as deemed necessary.

12.5.3.2.7 Sampling

Periodic sampling and analysis of process streams will identify radiation sources, help verify that process stream monitors are providing accurate information and thus help maintain exposures ALARA.

Most of the sampling of radioactive systems will be performed in chemical fume hoods. The fume hoods provide negative air pressure and minimize the possible spread of contamination. RWP requirements, if necessary, will specify appropriate protective measures to be taken during sampling. The possibility of radioactive spills and radiation exposure will be maintained ALARA during sample transport by the use of special shielded or remote handling and transportation devices.

12.5.3.2.8 Calibration

Periodic calibration of radiation detection instruments will help keep radiation exposures ALARA by assuring that the instruments are accurate and are providing reliable information. Portable radiation detection instruments are calibrated in accordance with manufacturers recommendations and Health Physics Instructions.

Portable survey meters will be calibrated using an enclosed and shielded calibrator.

Portable sources used to calibrate fixed instruments like the Area Radiation Monitoring system are in shielded containers that slip over the detectors, keeping radiation exposure to personnel ALARA.

12.5.3.2.9 Plant Cleanliness

Plant cleanliness is maintained in accordance with <Regulatory Guide 1.39> as discussed in <Section 1.8> and <Section 17.2>.

12.5.3.3 Controlling Access and Stay Time

Personnel who have not satisfactorily demonstrated comprehension of the information presented in Plant Access Training and Radiation Worker

Training will not be allowed to enter radiologically restricted areas unless escorted by someone who has satisfactorily completed the training.

All maintenance work performed on systems that contain, collect, store, or transport radioactive materials and may cause radiation exposure, will require an RWP, as determined by Radiation Protection, based on currently determined radiation, loose contamination or airborne activity levels.

The RWP will specify the radiological hazards associated with a job, including the radiation exposure rate. If necessary, full time radiation protection coverage will be provided and appropriate "stay time" will be used to maintain radiation exposures ALARA.

For additional details, see <Section 12.5.2.2>.

12.5.3.4 Contamination Control

Contamination surveys will be performed as described in <Section 12.5.3.1>. Contamination limits for areas, tools and clothing are described in the plant procedures and instructions.

Plant design will also help prevent the spread of contamination. The ventilation systems keep a negative air pressure in areas of possible high contamination. By keeping air flow into these areas, the possibility of airborne contamination leaving these areas will be minimized.

Contamination detection instrumentation will be placed at strategic locations throughout the plant. Step-off pads may be used to differentiate between contaminated areas and clean areas. After removing protective clothing, workers will monitor themselves for

contamination. If contamination is found, personnel will be decontaminated as quickly as possible to minimize the radiation exposure and prevent the spread of the contamination.

All tools used in contaminated areas will not be taken from radiologically restricted areas unless surveyed and released by radiation protection personnel.

12.5.3.5 Training Programs

All individuals working in or frequenting any portion of a radiologically restricted area shall be kept informed of the storage, transfer or use of radioactive materials and/or of radiation levels in the areas, and will be instructed in conformity with <10 CFR 19.12> in the following areas:

a. The health protection problems associated with radiation or radioactive materials

b. The precautions or procedures to minimize exposure

c. The purpose and use of protective devices

d. The appropriate response to warning signals made in the event of an unusual occurrence or malfunction at the plant

e. Applicable sections of the license and Title 10 of the Code of Federal Regulations for the protection of personnel from exposures to radiation or radioactive materials.

As a minimum, the above listed information and the requirements of <Regulatory Guide 8.13> and <10 CFR 20> will be presented in general employee training. Personnel who have not satisfactorily demonstrated

comprehension of the information presented in the Plant Access Training and Radiation Worker Training will not be allowed to enter radiologically restricted areas unless escorted by someone who has satisfactorily completed the training.

PNPP shall provide current copies of the following documents at designated locations:

a. A notice describing the following documents including where they may be examined:

   1. <10 CFR 19>

   2. <10 CFR 20>

   3. PNPP licenses

   4. PNPP Operating Procedures applicable to licensed activities

b. Any notice of violation involving radiological working conditions, any proposed or actual imposition of civil penalty, and order issued for imposing requirements or for modifying, suspending or revoking a license, and any response from PNPP.

c. Form NRC-3.

At the request of a worker, the PNPP Radiation Protection Section shall supply:

a. A report of the total effective dose equivalent received by that worker during the previous full year from January 1 through December 31.

b. A report of the total effective dose equivalent received by a terminated worker while that worker was engaged in activities pursuant to PNPP licenses.

c. A report of the actual or estimated total effective dose equivalent received by a terminated worker during the final calendar year or portion thereof that the worker was engaged in activities pursuant to PNPP licenses. In addition, a worker shall be provided a report of any exposure reported to the NRC for that specific worker.

12.5.3.6 Personnel Dosimetry

12.5.3.6.1 Personnel External Exposures

Personnel monitoring will be assigned to any individual entering a radiologically restricted area as per <10 CFR 20>.

Thermoluminescent dosimeters (TLDs) are the primary and official method of measuring an individual's occupational radiation exposure to the whole body; however, direct reading dosimeters may be used as the official record when used for the measurement of x-ray or gamma radiation.

Exposure data for all personnel will be recorded on Form NRC-5, or the equivalent. These records will be maintained by FENOC and will be preserved indefinitely or until the NRC authorizes their disposal. Current exposure status will be made available to each supervisor and individual, as required, to assist in keeping individual radiation exposures ALARA. Each worker shall receive exposure reports in accordance with <10 CFR 19.13>.

Personnel monitoring, and equipment for personnel monitoring and surveys, will be in conformance with the requirements of <10 CFR 20> and <Regulatory Guide 8.4>, <Regulatory Guide 8.9> and <Regulatory Guide 8.28>, as discussed in <Section 1.8>.

12.5.3.6.2 Internal Exposures

The bioassay program, composed of an internal screening component and a diagnostic bioassay component to determine individual doses, is implemented in accordance with <Regulatory Guide 8.26> and <10 CFR 20>.

All personnel who take part in the respiratory protection program are routinely screened for internal contamination during each exit from the radiologically restricted area and/or the protected area. If an individual alarms an exit monitor, the individual is evaluated for external contamination and decontaminated as necessary. After decontamination, if continued exit monitor alarms are being experienced, the radiation protection staff will perform a diagnostic bioassay of the individual to determine and evaluate the extent of internal contamination. The evaluation will identify and quantify the radionuclides in the body and determine the committed effective dose equivalent (CECE) resulting from internal contamination. CEDE exceeding 10% of the annual limit on intake (ALI) will be summed with the deep dose equivalent (DDE) and become part of the individual's annual dose record.

Training in the use and care of respiratory protection devices are given by qualified and experienced instructors. The training program is based on the hazards to be encountered and the types of respirators to be worn. Training will be given to personnel who will use respirators.

All personnel who are expected to continue using respirators will be retrained at least annually to retain a high degree of proficiency and help maintain radiation exposures ALARA.

12.5.3.7 Evaluation and Control of Potential Airborne Radioactivity

Fixed continuous air monitors and portable air monitors and air samplers are used to determine the concentrations of airborne radioactivity throughout the plant.

The fixed air monitors, described in <Section 12.3.4>, provide continuous data to indicate trends throughout the various plant areas. Particulate filters and charcoal cartridges are removed periodically to identify the specific nuclides encountered.

Portable air samplers are used to collect particulate and charcoal grab samples of areas of specific concern, for example, in preparation and conduct of specific work functions, to verify significant indicated changes by one or more fixed air monitors, or periodic air sampling throughout the plant.

In-plant iodine analysis is accomplished by collection of iodine samples utilizing charcoal cartridges with particulate prefilters and low volume air samplers. The cartridges and filters are analyzed by gamma spectroscopy and gross beta methods.

In the event the counting facilities are lost due to high background counting levels, an alternate counting room is established in the Technical Support Center, located in the service building. Noble gas interference is reduced by using silver zeolite cartridges or by flushing activated charcoal iodine cartridges with air.

<Table 12.5-4> lists the quantities of air samplers available. <Table 12.3-10> lists the airborne radiation monitors.

12.5.3.8 Radioactive Material Handling and Storage Methods

Handling of radioactive samples is described in <Section 12.5.3.2.7>. Various other types and quantities of radioactive sources are used to calibrate equipment. Recognized methods for the safe handling of radioactive materials, such as those recommended by the National Council on Radiation Protection and Measurements, will be used to maintain potential total effective dose equivalents ALARA.

All radioactive sources will be used or handled by, or under the direction of, Radiation Protection personnel. Individuals using these sources will be familiar with the radiological restrictions, regulations and limitations placed on their use. These limitations will help protect both the user, the source integrity and other personnel in the work vicinity.

Special Nuclear Material is maintained, handled and stored in accordance with <10 CFR 70>, excluding adherence to <10 CFR 70.24>. Instead, PNPP has chosen to comply with <10 CFR 50.68(b)>.

TABLE 12.5-1

LABORATORY EQUIPMENT

Type	Quantity	Range	Sensitivity/Accuracy	Remarks
Alpha Counter	1	$1\text{-}10^6$ cpm	30% of 2π (Cm-244)	Propor-tional Counter
Beta Counter	2	$1\text{-}10^6$ cpm	>10% (Cs-137)	G-M tube
Gamma Spectro-meter (Iodine counter)	1	2^{20} counts/ channel	>15% (Co^{60}) 2.3 KeV (FWHM) Co^{60}	HPGe

TABLE 12.5-2

PORTABLE SURVEY INSTRUMENTS

Type	Quantity	Range	Sensitivity/ Accuracy	Remarks
G-M Pancake	As required	0-50k cpm	>10% (Cs-137)	-
G-M Hand	As required	0-50k cpm	>10% (Cs-137)	mR/hr and cpm
G-M mR/hr	As required	0.1-2,000 mR/hr	±10% of Full Scale	Wide Range
Ion Chamber mR/hr	As required	1-1,000 mR/hr	±10% From 60 KeV to 1.3 MeV	-
mR/hr Tele-scoping	As required	0.01 R/hr - 999 R/hr	±15% From 70 KeV to 1.3 MeV	-
Ion Chamber R/hr	As required	1 mR/hr - 19.99 kR/hr	±15% of Full Scale	-
mrem/hr Neutron	As required	0-5k mrem/hr	.025 eV (Thermal) to approx. 10 MeV	-
R-Chamber	As required	0-250R	-	For calibration
Alpha-Scintillation	As required	0-50,000 cpm	28% of 2 π	-

TABLE 12.5-3

PERSONNEL MONITORING INSTRUMENTS

Type	Quantity	Range	Sensitivity/ Accuracy
TLD	As required	10mR-3000R	±30%
Dosimeter, Pocket	As required	0-500mR	±20%
Dosimeter, Pocket	As required	0-5R	±20%
Dosimeter, Pocket	As required	0-100R	±20%
Dosimeter Charger	As required	N/A	N/A
Portal Monitors	As required	N/A	200 nano Ci, ±20%
Personnel Friskers	As required	0-50k cpm	±10%

TABLE 12.5-4

RADIATION PROTECTION EQUIPMENT

Type	Quantity	Range	Remarks
Large Calibrator	1	approx. 500R/hr maximum	Cs-137 traceable to NIST
High Vol. Air Sampler	3	Approx. 11 cfm	
Low Vol. Air Sampler	3	Variable to 4 cfm	
Self-Contained Breathing Apparatus	10	<50 MPC(D)	10,000(PD)
Full Face Masks	200	N/A	CF Mode
Full Face Filters	300	N/A	-
Respirator Hoses	20	N/A	-
Respirator Junction Box	3	N/A	To 4 Respirators Per Junction Box
Waterproof Suits	30	N/A	-
Clothing Sets (Coveralls, Hoods)	2,500	N/A	-
Booties	5,000	N/A	
Rubbers (Pr.)	3,000	N/A	-
Rubber Gloves (Pr.)	5,000	N/A	-
Cotton Gloves (Pr.)	2,000 Doz.	N/A	-
Vacuum Cleaners	1	N/A	
Portable HEPA Filter Units	3	N/A	

12.6 DESIGN REVIEW OF PLANT SHIELDING FOR SPACES/SYSTEMS WHICH MAY BE USED IN POSTACCIDENT OPERATIONS OUTSIDE CONTAINMENT

12.6.1 INTRODUCTION

An accident equivalent to that described in <Regulatory Guide 1.3> would release large fractions of the nuclear core fission product inventory. Once released these fission products could be transferred to various areas in the plant creating high radiation areas and limiting personnel access. This review determines if these postaccident radiation fields unduly limit personnel access to areas necessary for mitigation of an accident. Corrective actions for problems identified are also determined. This design review of plant shielding for spaces and systems required for postaccident operations outside containment is in accordance with "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations" <NUREG-0578> Section II.B.2, as clarified by <NUREG-0660> and <NUREG-0737>.

The review is based on the following guidelines:

(Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.)

a. The postaccident dose rate in areas requiring continuous occupancy should not exceed 15mR/hr (control room and onsite technical support center.)

b. The postaccident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 rem whole body (total effective dose equivalent) or its equivalent (sample stations, panels, motor control centers, etc.)

c. The integrated dose to safety equipment as a result of the accident should be less than the dose for which the equipment has been qualified to ensure that the capability of the equipment to perform its safety function has not been degraded.

d. The minimum radioactive source term used in the evaluation should be equivalent to the source term recommended in the <Regulatory Guide 1.3>.

The occupancy and radiation design objectives for this review are given in <Table 12.6-1>.

12.6.2 RADIOACTIVE SOURCE RELEASE

The initial core inventory is shown in <Table 12.6-2>. The percents of radioactive fission product core inventory assumed to be released from the fuel rods are:

a.	Noble gas (Kr, Xe)	100%
b.	Halogens (I, Br)	50%
c.	Alkali metal (Cs, Rb)	50%
d.	Others	1%

This release is assumed to occur and be distributed instantaneously at the start of the accident.

12.6.3 RADIOACTIVE SOURCE DISTRIBUTION

The following radioactive source distributions were used to determine the shielding review concentration:

a. Source A - Liquid in the suppression pool and other systems not isolated from the core at the start of the accident and containing only liquid from a depressurized source is assumed to contain the following percents of core inventory of radioactive fission products:

1.	Noble gases (Kr, Xe)	0%
2.	Halogens (I, Br)	50%
3.	Alkali metals (Cs, Rb)	50%
4.	Others	1%

These fission products are assumed to be uniformly mixed in a volume of 117,105 cubic feet which is the low level water volume of the suppression pool.

b. Source B - For evaluating vital areas, the drywell atmosphere is assumed to contain the following percents of radioactive fission product core inventory:

1.	Noble gases (Kr, Xe)	100%
2.	Halogens (I, Br)	25%
3.	Others	0%

These fission products are assumed to be uniformly mixed in volume of the drywell air space of 277,685 cubic feet.

c. Source C - For evaluating vital areas, the primary containment atmosphere is assumed to contain the following percents of radioactive fission product core inventory:

1.	Noble gases (Kr, Xe)	100%
2.	Halogens (I, Br)	25%
3.	Others	0%

These fission products are assumed to be uniformly mixed in the free volume of the containment of 1,141,014 cubic feet.

d. Source D - Until the reactor vessel is depressurized, gases in the steam lines and any other vapor containing lines not isolated from the reactor coolant system are assumed to contain the following percents of radioactive fission product core inventory:

1.	Noble gases (Kr, Xe)	100%
2.	Halogens (I, Br)	25%
3.	Others	0%

These fission products are assumed to be uniformly mixed in the volume of the reactor coolant system steam space (9,189 cubic feet). This is a highly conservative source estimate because steam usage would deplete this source shortly after the start of the accident.

e. Source E - For equipment qualification inside containment, the larger of the two following source terms are used:

1. Source terms:

a)	Noble gases (Kr, Xe)	100%
b)	Halogens (I, Br)	50%
c)	Alkali metals (Cs, Rb)	50%
d)	Others	1%

These fission products are assumed to be released to the containment atmosphere.

2. Source terms:

a)	Noble gases (Kr, Xe)	100%
b)	Halogens (I, Br)	50%
c)	Alkali metals (Cs, Rb)	50%
d)	Others	1%

These fission products are assumed to be uniformly mixed in a volume no greater than the liquid volume of the reactor coolant system (11,838 cubic feet).

f. Source F - For qualification of equipment inside containment for non-LOCA events which do not depressurize the primary system, the following source terms are used:

1.	Noble gases (Kr, Xe)	10%
2.	Halogens (I, Br)	10%
3.	Others	0%

These fission products are assumed to be uniformly mixed in the volume of the reactor coolant system (11,838 cubic feet).

The initial radioactive source terms used are given in <Table 12.6-3>.

12.6.4 SYSTEMS CONTAINING RADIOACTIVE SOURCES

A review of PNPP identified systems which were likely to contain highly radioactive fluid following a design basis LOCA. The radioactive material is assumed to be instantaneously mixed in those systems connected either to the reactor coolant or to the containment atmosphere, that are not isolated at the start of the accident. Non-essential systems that are isolated and have no postaccident function are not considered in this review.

12.6.4.1 LPCS, HPCS, RHR (LPCI mode), RCIC Systems

Following an accident, the LPCS, HPCS, RHR (LPCI mode), and RCIC (water side) systems draw water from the suppression pool and inject it into the reactor vessel for emergency core cooling. The suppression pool water is assumed to be the only injection water source, although the HPCS and RCIC systems would initially draw water from the condensate storage tank. The RCIC (steam side) system draws main steam from the reactor vessel until the reactor is depressurized.

12.6.4.2 RHR (Shutdown Cooling Mode)

After reactor depressurization, the RHR system cools recirculating reactor water. Essentially all noble gas would be released from the reactor water upon depressurization. Therefore, it is assumed there are no noble gases in the RHR system while it is performing its shutdown cooling function. Also, no reactor water dilution by the condensate storage tank or suppression pool water is assumed.

12.6.4.3 RHR (Suppression Pool Cooling Mode)

The suppression pool cooling function of the RHR system maintains the correct temperature of the suppression pool water by circulating it through the RHR heat exchanger and returning it to the suppression pool.

12.6.4.4 RWCU

The RWCU system is designed to automatically isolate following a low reactor water level signal caused by a LOCA. Since the system is not designated for post-LOCA recovery cleanup operations, it is not included in the LOCA review.

Since the RWCU system would not receive a low reactor water level signal following a high energy line break accident (HELBA) that did not depressurize the reactor coolant system, it would not automatically isolate. Therefore, it is assumed to contain the non-LOCA equipment qualification HELBA source (Source F).

12.6.4.5 Liquid Radwaste System

The liquid radwaste system is automatically isolated from parts of the system that may contain highly contaminated postaccident water. Therefore, the liquid radwaste system is not included in this review.

12.6.4.6 Sampling System

Following an accident it is necessary to obtain and perform radioisotopic and chemical analyses of reactor coolant and containment atmosphere samples. The samples are taken at the 574′ 10″ elevation of the intermediate building and analyzed in the control complex count room as shown on <Figure 12.6-1>. The postaccident sampling system and count room are shielded such that an operator may collect and analyze samples under degraded core conditions without excessive radiation exposures.

12.6.4.7 Offgas System

The offgas system removes noncondensible fission gases from the main condenser. Since the condenser is isolated from the primary system following an accident, the offgas system is also isolated and is not included in this review.

12.6.4.8 Annulus Exhaust Gas Treatment System (AEGTS)

The AEGTS processes the air in the annular space between the shield building and the primary containment to limit radioactive releases to the environment during an accident. This system is assumed to be operating after a LOCA and to contain the design basis leakage from the primary containment atmosphere.

12.6.4.9 Feedwater Leakage Control System

The Feedwater Leakage Control (FWLC) system consists of two subsystems designed to eliminate through-line air leakage in the feedwater piping by providing a positive seal for the stem, bonnet and seat of the outboard isolation valve on each feedwater line. FWLC is initiated post-LOCA as described in <Section 6.9>. The water supply is the suppression pool via the Low Pressure Core Spray/Residual Heat Removal A water leg pump for Division 1 and Residual Heat Removal B/C water leg

pump for Division 2. The source term for the suppression pool water is given in <Table 12.6-3> and was increased 2% in accordance with <Regulatory Guide 1.49>.

12.6.5 SHIELDING REVIEW

After determining the systems and postaccident source distribution to be used for the shielding review, the SDC (Reference 1) and SPOT 1 (Reference 2) shielding codes were used to calculate the associated postaccident radiation doses.

Each compartment radiation level is calculated at what is judged to be maximum radiation dose point. This point is on the surface of the major system component and includes contributions from piping and other simultaneously operating components in that compartment.

Calculated radiation levels in corridors include contributions from:

a. Unattenuated radiation penetrating adjacent compartments' shield walls.

b. Direct radiation from piping or equipment.

c. Radiation scattered over or around shield walls.

Potential dose contributions not considered in this review are:

a. Normal operating sources which may exist at the time of the accident.

b. Airborne sources from equipment leakage.

12.6.6 AREAS REQUIRING PERSONNEL ACCESS

Areas which may require occupancy to permit an operator to aid in the mitigation of an accident are vital areas. The evaluation to determine the necessary vital areas included the control room, technical support center, post-LOCA hydrogen control system, containment isolation system, sampling and sample analysis areas, remote shutdown panel, ECCS alignment functions, motor control centers, instrument panels, emergency power supplies, central alarm station, and radwaste control panels (located in the radwaste building control room).

A study at PNPP determined that several of the areas considered were not vital areas. The post-LOCA hydrogen control system is located inside the reactor building and is remotely operated from the control room.

The containment isolation reset controls and ECCS alignment controls are initiated from the control room. The safety-related motor control centers are located in the control complex. They require no local actions but are accessible for infrequent occupancy, if required. The radwaste control panel and radwaste operator's console (both of which are located in the radwaste building control room) are excluded, because the radwaste system is automatically isolated in the event of an accident and is not designed to process accident wastes. The emergency power supply areas (standby diesel generators) are not vital areas, because the required functions are initiated from the control room and require no local action, however, they are accessible for infrequent occupancy, if required.

The security center is not a vital area, because after an accident, security functions will be implemented from the secondary alarm station in the control room.

The vital areas for PNPP assumed to require postaccident personnel access to bring the plant to cold shutdown and to implement the emergency plan include:

a. For continuous occupancy - the Control Room and Technical Support Center.

b. For frequent occupancy - the Remote Shutdown Panel, if required.

c. For infrequent occupancy - the Sampling Station, Sample Analysis Area, Auxiliary Building elevation 620' east end in area of 1P57F0565B (outboard MSIV accumulator safety-related air isolation valve) and the Remote Shutdown Panel.

These areas meet the occupancy and radiation design objectives given in <Table 12.6-1>.

12.6.7 POSTACCIDENT RADIATION ZONE DRAWINGS AND SUMMARY

Postaccident radiation zone drawings are given in <Figure 12.6-1>, <Figure 12.6-2>, <Figure 12.6-3>, <Figure 12.6-4>, <Figure 12.6-5>, <Figure 12.6-6>, and <Figure 12.6-7>. These radiation zones represent the maximum expected radiation dose rate for each area at the start of the accident. Normal operating dose levels which may exist at the time of the accident are not shown on these drawings. A summary of major locations and accident dose rates is given in <Table 12.6-4>.

12.6.7.1 Radiation Dose Rates as a Function of Time Following an Accident

The decay curves for radiation dose rates as a function of time following an accident for the auxiliary and intermediate buildings are given in <Figure 12.6-8> and <Figure 12.6-9>. These curves may be used along with the radiation zone drawings and summary table to predict radiation dose rates at a given time after the accident.

12.6.8 REFERENCES FOR SECTION 12.6

1. Arnold, E. D. and Maskewitz, B. F., "SDC-Shield Design Calculation Code for Fuel Handling Facilities," ORNL-3041, March 1966.

2. Kamphouse, J. L., "SPOT1 Shield Code," Gilbert Associates, Inc., October 1979.

3. A. Tobia, Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products, Revision 3, Central Electricity Generating Board, RD/B/M2666, CNDC (73) P4, June 1973.

TABLE 12.6-1

OCCUPANCY AND RADIATION DESIGN OBJECTIVES

Required Occupancy	Dose Rate Limit	Integrated Dose Objective
Continuous	15 mR/hr	5 rem for duration
Frequent	100 mR/hr	5 rem for all activities
Infrequent	500 mR/hr	5 rem per activity
Accessway	5000 mR/hr	Included in above doses

TABLE 12.6-2

INITIAL CORE INVENTORY

ISOTOPE	μCi/WATT
I-131	2.7 + 4
I-132	3.8 + 4
I-133	5.5 + 4
I-134	5.9 + 4
I-135	5.1 + 4
Kr-83m	-
Kr-85m	7.2 + 3
Kr-85	2.9 + 2
Kr-87	1.2 + 4
Kr-88	1.8 + 4
Sr-89	2.4 + 4
Sr-90	2.3 + 3
Y-90	2.4 + 3
Sr-91	3.0 + 4
Y-91	3.1 + 4
Sr-92	3.3 + 4
Y-92	3.3 + 4
Sr-93	3.8 + 4
Y-93	3.9 + 4
Y-94	4.1 + 4
Rh-105m	5.1 + 3

TABLE 12.6-2 (Continued)

ISOTOPE	μCi/WATT
Ru-106	1.6 + 4
Rh-106	1.7 + 4
Rh-107	1.6 + 4
Sb-127	2.8 + 3
Te-127	3.6 + 3
Sb-128	4.1 + 3
Sn-128	2.6 + 3
Sb-129	9.0 + 3
Te-129m	1.6 + 3
Cs-139	4.9 + 4
Ba-139	5.0 + 4
Ba-140	4.8 + 4
La-140	5.0 + 4
Ba-141	4.6 + 4
La-141	4.6 + 4
Ce-141	4.6 + 4
Pr-142	3.5 + 3
Ba-142	3.9 + 4
La-142	4.2 + 4
Xe-131m	-
Xe-133m	1.5 + 3
Xe-133	5.5 + 4
Xe-135m	9.7 + 3

TABLE 12.6-2 (Continued)

ISOTOPE	μCi/WATT
Xe-135	7.4 + 3
Xe-138	4.7 + 4
Br-83	2.7 + 3
Br-84	5.4 + 3
Br-88	1.8 + 4
Rb-88	2.3 + 4
Br-89	2.3 + 4
Y-95	4.4 + 4
Zr-95	4.5 + 4
Nb-95	4.5 + 4
Zr-97	4.6 + 4
Nb-97	4.6 + 4
Mo-99	5.1 + 4
Tc-99m	4.5 + 4
Mo-101	4.6 + 4
Ru-103	4.4 + 4
Rh-103m	4.3 + 4
Tc-103	3.5 + 4
Ru-105	2.4 + 4
Rh-105	2.4 + 4
Te-129	8.6 + 3
Sb-130	1.3 + 4
Sb-131	2.2 + 4

TABLE 12.6-2 (Continued)

ISOTOPE	μCi/WATT
Te-131	2.4 + 4
Te-131m	4.3 + 3
Te-132	3.8 + 4
Te-133	2.4 + 4
Te-133m	3.2 + 4
Te-134	4.9 + 4
Cs-134	2.0 + 3
Cs-137	3.3 + 3
Ba-137m	3.0 + 3
Cs-138	5.0 + 4
Ce-143	4.1 + 4
Pr-143	4.0 + 4
Ce-144	3.5 + 4
Pr-144	3.5 + 4
Pr-145	2.3 + 4
Pr-146	2.3 + 4
Pr-147	1.7 + 4
Nd-147	1.8 + 4
Pm-148	3.8 + 3
Nd-149	1.0 + 4
Pm-149	1.5 + 4
Pm-151	5.4 + 3
Eu-156	4.8 + 3

TABLE 12.6-3

INITIAL RADIOACTIVE SOURCE TERMS (GAMMAS/CC-SEC)

GAMMA-ENERGY GROUP (MEV)[1]	SOURCE	SUPPRESSION POOL	RWCU SYSTEM	STEAM DOME	REACTOR BLDG ATMOSPHERE	REACTOR COOLANT SYSTEM
	%CORE NOBLE GAS	0	10	100	100	100
	%CORE HALOGENS	50	10	25	25	50
	% CORE SOLIDS	1/50 (Cs-Rb)	0	0	0	1/50 (Cs-Rb)
0.1 - 0.5		1.86 +9	5.46 +9	2.06 +10	1.34 +8	4.69 +10
0.5 - 1.0		6.84 +9	1.33 +10	3.32 +10	2.15 +8	7.17 +10
1.0 - 1.5		3.26 +9	3.26 +9	4.41 +10	2.86 +8	3.48 +10
1.5 - 2.0		6.63 +8	1.48 +9	1.18 +10	7.69 +7	1.09 +10
2.0 - 2.5		4.57 +8	1.16 +9	8.50 +9	5.52 +7	1.43 +10
2.5 - 3.0		9.31 +7	1.94 +8	1.29 +10	8.36 +7	2.72 +9
3.0 - 3.5		1.49 +7	2.39 +6	2.28 +9	1.48 +7	1.49 +8
3.5 - 4.0		2.76 +7	3.87 +7	1.14 +8	7.41 +5	2.76 +8
4.0 - 5.0		7.74 +5	5.80 +5	1.80 +6	1.16 +4	7.74 +6

Note:

[1] (Reference 3) of <Section 12.6.8>

TABLE 12.6-4

DOSE RATES

Location	Time 0 Dose Rate (mR/hr)
AUXILIARY BUILDING	
Steam Tunnel	2.10E+9
RWCU Pump Cubicle	4.83E+8
LPCS Pump Cubicle	3.89E+8
RCIC Pump Cubicle	1.37E+9
RHR HX Cubicle	5.57E+8
Corridor	
Outside RCIC pump room 568’-4”	3.50E+4
Outside LPCS pump room 568’-4”	4.41E+3
Outside RHR HX room 568’-4”	2.84E+4
@ Elev. 599’-0” (see Figure 12.6-2)	
North Corridor and East side	1.61E+7
West side	2.84E+4

TABLE 12.6-4 (Continued)

INTERMEDIATE BUILDING	
Above Elev. 646”-0”	3.11E+4
Elev. 620’-6”	1.67E+4
Elev. 599’-0”	2.72E+4
Elev. 574’-10”	1.34E+3
DIESEL GENERATOR BUILDING	5.09E+0
CONTROL COMPLEX	<15[1]
Control Room	<15[1]
CENTRAL ALARM STATION	3.23E+2[1]
GUARD HOUSE	1.11E+2[1]
TECHNICAL SUPPORT CENTER	<15[1]

NOTE:

[1] Represents average dose rate for 30 days.