CHAPTER 12.0

RADIATION PROTECTION

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

TABLE OF CONTENTS (Continued)

LIST OF TABLES

CHAPTER 12 - LIST OF FIGURES

*Refer to Section 1.6 and Table 1.6-3. Controlled drawings were removed from the USAR at Revision 17 and are considered incorporated by reference.

NRC QUESTIONS PERTAINING TO CHAPTER 12.0

Section	Question Number	Section	Question Number
12.1.2.5b	331.1		
12.2.1.3	331.2		
12.2.1.2.3	331.3		
T12.2-7	331.4		
12.3.4.2.2.2.2	331.5		

CHAPTER 12.0

RADIATION PROTECTION

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

12.1.1 POLICY CONSIDERATIONS

The Operating Agent is committed to a company policy that supports the applicable Regulatory Guides (see Table 12.1-1) and parts of Title 10 of the Code of Federal Regulations in maintaining occupational radiation exposure (ORE) as low as reasonably achievable (ALARA). A Radiation Protection Operative Policy, initiated by the President and Chief Executive Officer through the Operating Agent Corporate Policy Manual has been implemented to meet this commitment. The Manager Chemistry/Radiation Protection is responsible for the technical content of the program and ensuring professional ALARA input. The Operating Agent establishes radiation protection practices for applicable plant activities and provides a qualified health physics organization to accomplish this goal. Utility management recognizes and emphasizes the importance of each individual's responsibilities to maintain occupational radiation exposures (ORE) - ALARA.

12.1.1.1 ALARA and Planning Committees

Under the ALARA Program, the ALARA Committee is responsible for integrating regulatory requirements with management policy by providing a multidisciplined forum for the discussion of radiological problems. A member of management serves as chairman of the ALARA Committee.

The station ALARA Coordinator participates in outage planning and scheduling of maintenance and testing activities during operations to evaluate dose reduction mechanisms and provide the necessary directives to ensure OREs, through the support of a functional onsite Health Physics Program, are maintained ALARA.

12.1.1.2 Health Physics

The ALARA Program maintains, through the Radiation Protection Manual and the Operative Policy contained in the Corporate Policy Manual: 1) that personnel work closely together to ensure that each Nuclear Division's ALARA responsibilities are carried out and that the site is updated to current regulatory standards; 2) an adequate training program to educate all nuclear station personnel in the areas of ALARA practices; 3) a method of reviewing the effectiveness of the station ALARA Program; 4) the formation of an ALARA committee to resolve areas of activity which offer the potential for increasing radiation exposures.

The Health Physics ALARA group is responsible for analyzing the latest regulatory criteria and providing general Health Physics support. This section also performs ALARA reviews of non-expedited plant modification requests. Expedited plant modification requests (Category I) ALARA reviews will be performed by Engineering.

The plant manager is responsible for ensuring that a station ALARA program is developed and that all station personnel support it. The plant manager is responsible for all operational aspects of the station, including the WCGS Health Physics Program, and support of the station Radiation Protection Manager.

The Radiation Protection Manager is responsible for developing and implementing the WCGS Health Physics Program and maintaining plant occupational radiation exposures ALARA. This manager is the site expert on radiation protection and is responsible for supporting operations and maintenance through radiological safety procedures, manpower and the provision of radiological protection equipment in controlled areas. This manager participates in site planning activities such as, but not limited to, the emergency plan, outages, equipment procurement, and personnel training, which require radiological input.

To ensure freedom from operation and maintenance pressures, this manager retains the administrative freedom to report technical information of an immediate nature directly to the plant manager. Any ALARA concerns can be appealed to the ALARA Committee and reported to corporate management.

The WCGS Health Physics organization~~s~~ works to consider the radiological implications of WCGS from long range perspectives. The organization ensures the application of sound Health Physics, radiation protection, and ALARA philosophies.

12.1.2 DESIGN CONSIDERATIONS

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

12.1.2.1 Plant Design

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

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

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

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

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

12.1.2.2 Scale-Model Program

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

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

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

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

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

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

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

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

12.1.2.4 Design Reviews by SNUPPS

Operating Agent reviews of safety-related systems, structures, and equipment were coordinated through the SNUPPS Technical Committee, which was composed of senior-level utility engineers, one from each SNUPPS utility (Wolf Creek and Callaway). It was the responsibility of the Technical Committee to obtain comments from appropriate personnel within their companies and to bring those comments to meetings of the Committee, where decisions are made on the basis of discussion and eventually a vote. Throughout the duration of the SNUPPS project, the Technical Committee met or had a telephone conference call on the average of once every week and a half, and total meeting days per year of the Technical Committee averaged about 50. The SNUPPS technical director participated in Technical Committee meetings.

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

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

During design and construction of the first SNUPPS unit (Callaway), SNUPPS staff and qualified personnel from WCGS participated in a review of the effectiveness of design to the ALARA objectives. Periodically during construction, preoperational testing, and start-up of the SNUPPS units, qualified personnel from WCGS participated in ALARA reviews of Wolf Creek.

12.1.2.5 Examples of Radiation Protection Design Reviews

a. Radiation Zone Drawings

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

b. Reactor Cavity

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

c. Radioactive Demineralizers

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

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

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

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

d. Airborne Radioactivity in Containment

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

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

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

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

4. Provision of a containment atmospheric control system (Section 9.4.6) to remove airborne iodine from the containment. This reduces the potential airborne exposure to operators entering the containment and also results in reduced environmental concentrations of iodine when purging the containment.

e. Steam Generator Maintenance

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

12.1.2.6 Decommissioning

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

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

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

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

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

12.1.3 OPERATIONAL CONSIDERATIONS

The Health Physics staff works with other WCGS groups to coordinate their input and ensure that proper radiological surveillance and controls for maintenance, operations, waste handling, inservice inspection, decommissioning and other activities are performed in a manner that maintains occupational exposures ALARA. This includes work on such systems as the NSSS, the Residual Heat Removal System, the Fuel Handling System, the Liquid Waste Management Systems, the Gaseous Waste Management Systems, the Solid Waste Management Systems, the Fuel Pool Cooling and Cleanup System and other systems that collect, store, contain or transport liquid, gaseous and solid radioactive material.

Routine operational practices used at WCGS to promote an ALARA philosophy and objectives are; the employment of a radiological staff that meets Regulatory Guide 1.8 requirements, the proper training of personnel and the preparation and preplanning of potentially high radiation exposure jobs. Briefings, surveys, critiques, etc, are administrative tools, which are used to ensure that station doses are maintained ALARA. The development and implementation of radiation protection procedures, practices and techniques in conjunction with adequate supervisory and radiation protection surveillance, provides a system that ensures ALARA is adequately reflected in station activities.

The operation and calibration procedures developed and implemented by the WCGS Health Physics staff provide specific guidelines, techniques and methods that incorporate guidance from Regulatory Guides 8.8 and 8.10. Past operational experience of the Health Physics personnel and information from other Nuclear Power Plants were used in program development and implementation to minimize occupational exposure problems. A description of the Health Physics program is given in Section 12.5.3.

The Manager Chemistry/Radiation Protection through a Superintendent Chemistry/Radiation Protection has the primary responsibility for developing and implementing the WCGS Health Physics Program. Implementation of the program is through the use of administrative exposure controls and procedures, Health Physics procedures, employee training, Health Physics review of plant procedures, job preparation, pre-planning of work which may produce significant exposures, and a Radiation Work Permit (RWP) program by which Health Physics regulates access to controlled areas. A description of the RWP System is given in Section 12.5.3.

To ensure that Health Physics is an integral part of the WCGS plant operation, the Manager Chemistry/Radiation Protection is a member of the Plant Safety Review Committee.

The Manager Chemistry/Radiation Protection through a Superintendent Chemistry/Radiation Protection periodically reviews exposure records for the purpose of identifying exposure by category, location, job function, etc, to enable the recommendation of changes to plant procedures, operating methods, etc, where such changes may reduce exposure.

12.1.4 Quality Assurance of Maintenance of ALARA

To verify that radiation protection functions are being performed as required and that a high level of radiological safety is maintained, review and evaluation through audit of the radiation protection program is performed biennially by the Operating Agent Quality & Performance Improvement Division. The performance of audits and review of radioactivity monitoring (fixed and portable), radioactivity sampling, contamination measurement and analysis, internal and external personnel monitoring, instrument storage, calibration and maintenance, decontamination, and the respiratory protection program including testing and contamination control to assure program effectiveness is consistent with the position of NUREG 0761.

The audit program is conducted in accordance with established auditing principles:

1. Use of inspectors/auditors with training and expertise in the area being audited.

2. Use of audit personnel with no responsibilities or "vested interests" in the areas being reviewed.

3. Documentation of audit results and findings, and review by management.

4. Performance of corrective or followup actions as appropriate.

Additional quality for these items is built into HP procedures utilizing quality provisions from regulatory guides.

Adequacy of permanent and temporary biological shielding within the Radiation Controlled Areas (RCAs) is verified by periodic radiation surveys.

The programs which control the monitoring activities are administered to meet the requirements of 10 CFR 20.

12.1.5 REFERENCE

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

TABLE 12.1-1

Regulatory Criteria Applicable to The Operating Agent's Health Physics Program

Number	Title	Revision	Issuance
1.8	Personnel Selection and Training	Draft Rev. 2	2/79
8.2	Guide For Administrative Practices in Radiation Monitoring	-	2/2/73
8.4	Direct-Reading and Indirect Reading Pocket Dosimeters	-	2/26/73
8.7	Occupational Radiation Exposure Records System	-	5/73
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA)	3	6/78
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	-	9/73
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable	1-R	9/75
8.13	Instruction Concerning Prenatal Radiation Exposure	1	11/75
8.14	Personnel Neutron Dosimeters	1	8/77
8.15	Acceptable Programs for Respiratory Protection	-	10/76
8.26	Applications of Bioassay for Fission and Activation Products	-	9/80
ANSI N13.5-1972	American National Standard performance specfications for direct reading and indirect reading pocket dosimeters for x - and gamma radiation	-	1972
ANSI N13.6-1966	American National Standard practice for occupational radiation exposure records systems	R1972	1972
ANSI/ANS 3.1-1978	American National Standard for selection and training of Nuclear Power Plant Personnel		1978

On June 24, 1997 an exemption from the requirements of 10CFR70.24 was granted by the NRC, therefore Regulatory Guide 8.12 is no longer applicable. On November 12, 1998 the NRC issued 10CFR50.68, which provides eight criteria that may be followed in lieu of criticality monitoring per 10CFR70.24 and revised 10CFR70.24 to make any exemption ineffective so long as the licensee elects to comply to 10CFR50.68.

12.2 RADIATION SOURCES

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

12.2.1 CONTAINED SOURCES

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

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

12.2.1.1 Containment

12.2.1.1.1 Reactor Core

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

12.2.1.1.2 Reactor Coolant System

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

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

12.2.1.1.3 Secondary Coolant Cycle

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

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

12.2.1.1.4 Auxiliary Systems

Residual heat removal system - see Section 12.2.1.2.1.

Chemical and volume control system - see Section 12.2.1.2.2.

Boron Recycle system - see Section 12.2.1.5.

12.2.1.1.5 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant source of radiation in the containment after plant shutdown for refueling. A reactor operating time necessary to establish fission product buildup near equilibrium for the reactor at rated power is used in determining the source strength. Shielding requirements for spent fuel transfer are based on the fission product activity present 100 hours after shutdown to conservatively take credit for the time elapsed prior to the initiation of refueling operations. Source terms for spent fuel are listed in Table 12.2-4.

12.2.1.2 Auxiliary Building

12.2.1.2.1 Residual Heat Removal System

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

12.2.1.2.2 Chemical and Volume Control System

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

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

12.2.1.2.3 Nuclear Sampling System

The sampling systems for (a) reactor coolant, (b) containment sump water, and (c) containment atmosphere have been evaluated to meet normal sample conditions and the guidelines in WCAP-14986-A, Rev. 2, "Post Accident Sampling system Requirements: A Technical Basis," to assure that a recovery sample can be taken if needed. Steps will be taken to assure that exposure is minimized while taking recovery samples.

By Amendment 137, the in-line post-accident sampling system is no longer used, although the equipment is still in place. The sample panel for this system is located in Room 1312 of the auxiliary building.

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

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

12.2.1.3 Fuel Building

12.2.1.3.1 Spent Fuel Storage and Transfer

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

12.2.1.3.2 Fuel Pool Cooling and Cleanup System

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

12.2.1.4 Turbine Building

12.2.1.4.1 Main Steam Supply and Power Conversion Systems

Potential radioactivity in the main steam supply and power conversion systems is a result of steam generator tube leaks and fuel cladding defects, as discussed in Section 12.2.1.1.3. This radioactivity is sufficiently low that no radiation shielding for equipment in the turbine building, except the condensate demineralizers, is required in order to meet the shielding design requirements. The isotopic concentrations for a condensate demineralizer bed and other secondary system sources are listed in Table 12.2-8.

12.2.1.5 Radwaste Building

12.2.1.5.1 Boron Recycle and Liquid Radwaste Systems

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

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

12.2.1.5.2 Gaseous Radwaste System

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

12.2.1.5.3 Solid Radwaste System

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

12.2.1.6 Sources Resulting from Design Basis Accidents

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

12.2.1.7 Stored Radioactivity

The principal sources of activity not stored inside the plant structures are the reactor makeup water storage tank (RMWST) and the refueling water storage tank (RWST). The RMWST is expected to contain concentrations of radionuclides which yield a surface dose rate of 0.5 mrem/hr or less. The RWST is expected to have a maximum contact dose of less than 10 mrem/hr when the water is returned from the refueling pool. This is rapidly reduced by processing through the fuel storage pool purification filter and demineralizer. Spent fuel is stored in the fuel storage pool until prepared for shipment offsite. Storage space is allocated in the radwaste building for the storage of spent filter cartridges and solidified spent resins, evaporator bottoms, and chemical wastes. Radioactive wastes stored inside designed storage areas are shielded so anticipated access outside the structure will meet Zone A specifications (see Figure 12.3-2). If it becomes necessary to store radioactive wastes outside the plant structures, adequate radiation protection measures are taken by the health physics staff. Tabulations of the activities within these tanks are provided in Table 11.1-6 (Sheets 1,2, and 3).

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

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

An analysis of operating plant measurements of average airborne radionuclide concentrations (Ref. 6) and their respective DACs for various stations in the auxiliary building (these stations include the waste handling area and the waste gas decay tank rooms, which for WCGS are located in the radwaste building) indicates that the concentration to DAC ratios in all stations is well within the limits of 10CFR20.1204(G) to consider these airborne activities as insignificant. It is expected that WCGS will not have airborne radioactivity concentrations significantly greater than the operating plant data for corresponding locations. However, it is possible that, within these stations, there may be rooms where maximum airborne concentrations can occur due to localized leakage of radioactive fluids, but these rooms house equipment and components that handle highly radioactive fluids and consequently are D or E zones and, therefore, are normally not occupied. Also the ventilation systems in the auxiliary and radwaste buildings are designed in such a manner that airborne contamination from high radiation zones does not generally spread into low radiation zones, since the airflow is from regions of lower potential for contamination to those with higher potential for contamination. Consequently, negligible airborne radioactivity concentrations are expected in those areas of the auxiliary and radwaste buildings which are accessible (see Table 12.2-10). Airborne radioactivity concentrations in the turbine building are also expected to be negligible, since possible leaks into the turbine building are only from the secondary side, and, also, the turbine building ventilation exhaust is high (at least 90,000 cfm). For example, airborne concentrations are calculated to be 2.3×10^{-13} μCi/cc and 6.9×10^{-12} μCi/cc for I-131 and Xe-133, respectively in the turbine building.

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

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

12.2.2.1 Model For Calculating Airborne Concentrations

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

$$C_i(t) = \frac{(LR)_i \, A_i \, (PF)_i \, (1-e^{-\lambda_{Ti} t})}{V\lambda_{Ti}} \qquad (1)$$

where

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

and

A_i = activity concentration of the ith leaking or evaporating radioisotope in μCi/gm

$(PF)_i$ = partition factor or the fraction of the leaking activity that is airborne for the i^{th} radioisotope

λT_i = Total removal rate constant for the i^{th} radioisotope in sec-1 from the applicable region

= $(\lambda di + \lambda e)$

$\lambda di + \lambda e$ are the removal rate constants in sec^{-1} due to radioactive decay for the i^{th} radioisotope and the exhaust from the applicable region

t = time interval between the start of the leak and the time at which the concentration is evaluated in seconds

V = free volume of the region in which the leak occurs in cc

$C_i(t)$ = airborne concentration of the i^{th} radioisotope at time t in $\mu Ci/cm^3$ in the applicable region

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

$$C_{Eqi} = (LR)_i \; A_i \; (PF)_i \; /V \tau Ti \qquad (2)$$

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

12.2.3 REFERENCES

1. C. M. Lederer, et. al., Table of Isotopes, Lawrence Radiation Laboratory, University of California (March, 1968).

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

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

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

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

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

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

TABLE 12.2-1

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

Energy Group	Neutron Flux (neutrons/cm^2-sec)
ϕ_1 (E > 1.0 Mev)	7.6×10^8
ϕ_2 (5.53 Kev $\leq$ E $\leq$ 1.0 Mev)	1.2×10^{10}
ϕ_3 (0.625 ev $\leq$ E $\leq$ 5.53 Kev)	7.1×10^9
ϕ_4 (E < 0.625 ev)	1.8×10^9

TABLE 12.2-2

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

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

TABLE 12.2-3

PRESSURIZER SHIELDING SOURCE TERMS

Liquid Volume

Isotope	Activity (μCi/gm)
N-16	1.8 (max)
Cr-51	1.68E-03
Mn-54	3.10E-04
Fe-55	1.61E-03
Fe-59	9.30E-04
Co-58	1.53E-02
Co-60	2.02E-03
Br-83	2.53E-04
Br-84	3.08E-05
Br-85	3.39E-07
Rb-86	1.48E-04
Rb-88	1.32E-03
Sr-89	6.84E-04
Sr-90	2.11E-05
Sr-91	1.29E-04
Y-89M	6.16E-08
Y-90	1.35E-05
Y-91M	8.44E-05
Y-91	1.33E-04
Y-93	7.05E-06
Zr-95	5.72E-05
Nb-95	4.84E-05
Nb-95M	2.92E-05
Mu-99	7.34E-02
Tc-99	3.68E-09
Ru-103	4.14E-05
Ru-106	1.00E-05
Te-125M	2.75E-05
Te-127M	2.73E-04
Te-127	3.26E-04
Te-129M	1.27E-03
Te-129	8.23E-04
Te-131M	6.13E-04
Te-131	1.15E-04
Te-132	1.24E-02
I-129	8.55E-13
I-130	5.17E-04
I-131	3.83E-01
I-132	1.71E-02
I-133	1.46E-01
I-134	9.09E-04
I-135	2.68E-02
Cs-134	5.24E-02

Liquid Volume

Isotope	Activity (μCi/gm)
Cs-135	1.04E-10
Cs-136	2.10E-02
Cs-137	3.79E-02
Ba-137M	3.58E-02
Ba-140	3.55E-04
La-140	3.43E-04
Ce-141	1.32E-04
Ce-143	2.19E-05
Ce-144	6.86E-05
Pr-143	8.67E-05
Pr-144	6.86E-05

Steam Volume

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

TABLE 12.2-4

SPENT FUEL SHUTDOWN SOURCES
(FULL CORE)

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

Values given in Mev/cc-sec.

TABLE 12.2-5

RADIATION SOURCES RESIDUAL HEAT REMOVAL SYSTEM

Isotope	Activities (μCi/gm)	Isotope	Activities (μCi/gm)
Cr-51	1.51E-03	I-131	4.37E-01
Mn-54	2.48E-04	I-132	6.36E-02
Fe-55	1.28E-03	I-133	5.46E-01
Fe-59	7.98E-04	I-134	3.17E-03
Co-58	1.28E-02	I-135	2.06E-01
Co-60	1.60E-03	Xe-131M	3.94E-02
Br-83	2.49E-03	Xe-133M	2.07E.01
Br-84	2.28E-05	Xe-133	1.06E+01
Br-85	negligible	Xe-135M	3.56E-02
Kr-83M	1.34E-02	Xe-135	5.42E-01
Kr-85M	1.20E-01	Xe-137	negligible
Kr-85	1.68E-02	Xe-138	8.llE-07
Kr-87	1.48E-02	Cs-134	4.10E-02
Kr-88	1.57E-01	Cs-135	4.48E-13
Kr-89	negligible	Cs-136	2.llE-02
Rb-86	1.39E-04	Cs-137	2.95E-02
Rb-88	1.75E-01	Cs-138	4.47E-04
Rb-89	5.28E-08	Ba-137M	2.79E-02
Sr-89	5.73E-04	Ba-140	3.58E-04
Sr-90	1.64E-05	La-140	2.54E-04
Sr-91	8.01E-04	Ce-141	1.14E-04
Y-89M	5.15E-08	Ce-143	6.03E-05
Y-90	2.58E-06	Ce-144	5.41E-05
Y-91M	5.30E-04	Pr-143	8.18E-05
Y-91	1.06E-04	Pr-144	5.41E-05
Y-93	4.25E-05		
Zr-95	4.79E-05		
Nb-95	4.00E-05		
Nb-95M	1.45E-06		
Mo-99	1.32E-01		
Tc-99	2.00E-10		
Ru-103	3.59E-05		
Ru-106	8.00E-06		
Te-125M	2.32E-05		
Te-127M	2.24E-04		
Te-127	5.63E-04		
Te-129M	1.12E-03		
Te-129	7.66E-04		
Te-131M	1.82E-03		
Te-131	3.34E-04		
Te-132	2.08E-02		
I-129	2.63E-14		
I-130	2.75E-03		

TABLE 12.2-6

CHEMICAL AND VOLUME CONTROL SYSTEM SOURCES

LETDOWN MIXED BED DEMINERALIZER

Isotope	Activity (μCi/cc)
Cr-51	3.30E+01
Mn-54	3.36E+01
Fe-55	2 23E+02
Fe-59	2.80E+01
Co-58	6.94E+02
Co-60	2.96E+02
Br-83	6.26E-01
Br-84	7.47E-02
Br-85	8.17E-04
Rb-86	1.15E+00
Rb-88	1.78E+00
Sr-89	2.30E+01
Sr-90	3.25E+00
Sr-91	3.42E-01
Y-89M	2.07E-03
Y-90	3.21E+00
Y-91M	2.25E-01
Y-91	5.13E+00
Y-93	1.88E-02
Zr-95	2.40E+00
Nb-95	3.44E+00
Nb-95M	2.40E+00
Mo-99	3.02E+02
Tc-99	9.73E-04
Ru-103	1.12E+00
Ru-106	1.14E+00
Te-125M	1.04E+00
Te-127M	1.72E+01
Te-127	1.74E+01
Te-129M	2.99E+01
Te-129	1.92E+01
Te-131M	1.95E+00
Te-131	3.64E-01
Te-132	5.49E+01
I-129	1.12E-06
I-130	1.41E+00
I-131	2.83E+03
I-132	6.73E+01
I-133	4.30E+02
I-134	2.21E+00
I-135	6.92E+01
Cs-134	3.88E+03
Cs-135	1.73E-05

TABLE 12.2-6 (Sheet 2)

Isotope	Activity (μCi/cc)
Cs-136	1.22E+02
Cs-137	3.26E+03
Ba-137M	3.08E+03
Ba-140	3.66E+00
La-140	3.98E+00
Ce-141	2.96E+00
Ce-143	7.16E-02
Ce-144	7.20E+00
Pr-143	9.56E-01
Pr-144	7.20E+00

Reactor Coolant Filter

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

NOTE:

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

TABLE 12.2-6 (Sheet 3)

Regenerative Heat Exchanger Shell Side and Excess Letdown and Letdown Heat Exchanger Tube Side		Volume Control Tank Liquid Volume and Seal Water Heat Exchanger		Volume Control Tank Gaseous Volume	
Isotope	Activities (μCi/gm)	Isotope	Activities (μCi/cc)	Isotope	Activity (μCi/cc)
Cr-51	1.90E-03	Cr-51	1.73E-04	Kr-83M	2.69E-01
Mn-54	3.10E-04	Mn-54	2.82E-05	Kr-85M	2.01E+00
Fe-55	1.60E-03	Fe-55	1.46E-04	Kr-85	2.40E-01
Fe-59	1.00E-03	Fe-59	9.10E-05	Kr-87	6.11E-01
Co-58	1.60E-02	Co-58	1.46E-03	Kr-88	3.11E+00
Co-60	2.00E-03	Co-60	1.82E-04	Kr-89	3.19E-03
Br-83	1.00E-02	Br-83	1.00E-03	Xe-131M	5.65E-01
Br-84	5.42E-03	Br-84	5.42E-04	Xe-133M	2.96E+00
Br-85	6.25E-04	Br-85	6.25E-05	Xe-133	1.52E+02
Kr-83M	4.54E-02	Rb-86	8.08E-05	Xe-135M	3.88E-02
Kr-85M	2.25E-01	Rb-88	1.90E-01	Xe-135	7.16E+00
Kr-85	1.67E-02	Sr-89	6.65E-05	Xe-137	6.88E-03
Kr-87	1.32E-01	Sr-90	1.90E-06	Xe-138	1.17E-01
Kr-88	4.23E-01	Sr-91	1.24E-04		
Kr-89	1.13E-02	Y-90	2.28E-07		
Rb-86	1.77E-04	Y-91M	6.84E-05		
Rb-88	4.17E-01	Y-91	1.22E-05		
Sr-89	7.29E-04	Y-93	6.46E-06		
Sr-90	2.08E-05	Zr-95	5.46E-06		
Sr-91	1.35E-03	Nb-95	4.55E-06		
Y-90	2.50E-06	Mo-99	1.60E-02		
Y-91M	7.50E-04	Ru-103	4.10E-06		
Y-91	1.33E-04	Ru-106	9.10E-07		
Y-93	7.08E-05	Te-125M	2.64E-06		
Zr-95	6.00E-05	Te-127M	2.55E-05		
Nb-95	5.00E-05	Te-127	7.74E-05		
Mo-99	1.75E-01	Te-129M	1.27E-04		

TABLE 12.2-6 (Sheet 4)

Regenerative Heat Exchanger Shell Side and Excess Letdown and Letdown Heat Exchanger Tube Side

Isotope	Activities (µCi/gm)
Ru-103	4.50E-05
Ru-106	1.00E-05
Te-125M	2.90E-05
Te-127M	2.80E-04
Te-127	8.50E-04
Te-129M	1.40E-03
Te-129	1.60E-03
Te-131M	2.50E-03
Te-131	1.10E-03
Te-132	2.70E-02
I-130	4.37E-03
I-131	5.62E-01
I-132	2.08E-01
I-133	7.92E-01
I-134	9.79E-02
I-135	3.96E-01
Xe-131M	3.98E-02
Xe-133M	2.17E-01
Xe-133	1.08E+01
Xe-135M	3.00E-02
Xe-135	6.44E-01
Xe-137	2.04E-02
Xe-138	9.89E-02
Cs-134	5.21E-02
Cs-136	2.71E-02
Cs-137	3.75E-02
Ba-137M	3.33E-02
Ba-140	4.58E-04
La-140	3.12E-04

Regenerative Heat Exchanger Shell Side and Excess Letdown and Letdown Heat Exchanger Tube Side

Isotope	Activities (µCi/gm)
Ce-141	1.46E-04
Ce-143	8.33E-05
Ce-144	6.87E-05
Pr-143	1.04E-04
Pr-144	6.87E-05

Volume Control Tank Liquid Volume and Seal Water Heat Exchanger

Isotope	Activities (µCi/cc)
Te-129	1.46E-04
Te-131M	2.28E-04
Te-131	1.00E-04
Te-132	2.46E-03
I-130	4.37E-04
I-131	5.62E-02
I-132	2.08E-02
I-133	7.92E-02
I-134	9.79E-03
I-135	3.96E-02
Cs-134	2.38E-02
Cs-136	1.24E-02
Cs-137	1.71E-02
Ba-137M	3.04E-03
Ba-140	4.18E-05
La-140	2.85E-05
Ce-141	1.33E-05
Ce-143	7.60E-06
Ce-144	6.27E-06
Pr-143	9.50E-06
Pr-144	6.27E-06

NOTE:

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

TABLE 12.2-6 (Sheet 5)

Boric Acid Tanks

Isotope	Activities (μCi/cc)
Cr-51	4.16E-06
Mn-54	8.56E-07
Fe-55	4.49E-06
Fe-59	2.41E-06
Co-58	4.08E-05
Co-60	5.63E-06
Br-83	2.54E-08
Br-84	1.92E-12
Br-85	negligible
Rb-86	8.40E-06
Rb-88	1.87E-09
Rb-89	5.94E-14
Sr-89	7.32E-06
Sr-90	5.90E-08
Sr-91	6.39E-08
Y-89M	6.58E-10
Y-90	5.00E-08
Y-91M	4.27E-08
Y-91	3.54E-07
Y-93	3.61E-09
Zr-95	1.52E-07
Nb-95	1.35E-07
Nb-95M	1.18E-07
Mo-99	8.72E-05
Tc-99	1.45E-11
Ru-103	1.06E-07
Ru-106	2.77E-08
Te-125M	7.23E-08
Te-127M	7.40E-07
Te-127	7.68E-07
Te-129M	3.21E-06
Te-129	2.05E-06
Te-131M	5.16E-07
Te-131	9.41E-08
Te-132	1.59E-05
I-129	5.08E-15
I-130	3.06E-07
I-131	7.46E-04
I-132	1.67E-05
I-133	1.09E-04
I-134	2.26E-09
I-135	1.06E-05
Cs-134	3.56E-03
Cs-135	3.71E-09

TABLE 12.2-6 (Sheet 6)

Boric Acid Tanks

Isotope	Activities (μCi/cc)
Cs-136	1.llE-03
Cs-137	2.59E-03
Cs-138	7.90E-08
Ba-137M	2.45E-03
Ba-140	7.64E-07
La-140	8.17E-07
Ce-141	3.31E-07
Ce-143	1.92E-08
Ce-144	1.90E-07
Pr-143	1.93E-07
Pr-144	1.90E-07

TABLE 12.2-7

FUEL STORAGE POOL WATER ACTIVITIES**

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

* Other isotopes will be present in much lower concentrations.

** Typical spent fuel pool concentrations at operating plants with similar cleanup systems.

TABLE 12.2-8

SECONDARY SYSTEM ACTIVITIES

Condensate Demineralizer		High TDS Regenerant Collector Tank		Secondary Liquid Waste Drain Collector Tank		Secondary Liquid Waste Monitor Tank	
Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)	Isotope	Activity (μ Ci/cc)
Cr-51	1.85E-06	Cr-51	2.04E-07	Cr-51	3.98E-11	Cr-51	2.38E-12
Mn-54	4.96E-07	Mn-54	6.43E-08	Mn-54	8.99E-12	Mn-54	6.43E-13
Fe-55	2.01E-06	Fe-55	2.61E-07	Fe-55	3.60E-11	Fe-55	2.61E-12
Fe-59	1.33E-06	Fe-59	1.73E-07	Fe-59	2.67E-11	Fe-59	1.72E-12
Co-58	1.86E-05	Co-58	2.41E-06	Co-58	3.57E-10	Co-58	2.41E-11
Co-60	2.27E-06	Co-60	2.94E-07	Co-60	4.05E-11	Co-60	2.94E-12
Br-83	4.96E-07	Br-83	6.43E-08	Br-83	8.83E-11	Br-83	5.83E-13
Br-84	2.94E-08	Br-84	3.81E-09	Br-84	5.23E-12	Br-84	7.03E-18
Br-85	4.27E-11	Br-85	5.54E-12	Br-85	7.62E-15	Br-85	6.31E-66
Rb-86	1.52E-07	Rb-86	1.96E-08	Rb-86	3.95E-12	Rb-86	1.94E-13
Rb-88	7.73E-08	Rb-88	1.00E-08	Rb-88	1.53E-11	Rb-88	3.08E-22
Sr-89	9.40E-07	Sr-89	1.22E-07	Sr-89	1.86E-11	Sr-89	1.21E-12
Sr-90	2.10E-08	Sr-90	2.73E-09	Sr-90	3.75E-13	Sr-90	2.73E-14
Sr-91	4.21E-08	Sr-91	5.46E-09	Sr-91	7.13E-12	Sr-91	3.01E-14
Y-89M	8.46E-11	Y-89M	1.10E-11	Y-89M	1.67E-15	Y-89M	1.09E-16
Y-90	1.73E-08	Y-90	2.25E-09	Y-90	1.36E-13	Y-90	2.29E-14
Y-91M	2.86E-08	Y-91M	3.70E-09	Y-91M	4.84E-12	Y-91M	2.01E-14
Y-91	1.50E-07	Y-91	1.95E-08	Y-91	2.90E-12	Y-91	1.94E-13
Y-93	2.19E-09	Y-93	2.84E-10	Y-93	3.68E-13	Y-93	1.61E-15
Zr-95	9.26E-08	Zr-95	1.20E-08	Zr-95	1.79E-12	Zr-95	1.20E-13
Nb-95	9.43E-08	Nb-95	1.22E-08	Nb-95	1.78E-12	Nb-95	1.23E-13
Nb-95M	6.62E-08	Nb-95M	8.59E-09	Nb-95M	2.61E-13	Nb-95M	8.80E-14
Nb-99	4.89E-05	Nb-99	6.35E-06	Nb-99	3.14E-09	Nb-99	5.82E-11
Tc-99	5.99E-12	Tc-99	7.77E-03	Tc-99	2.59E-17	Tc-99	7.96E-18
Ru-103	4.37E-08	Ru-103	5.66E-09	Ru-103	8.89E-13	Ru-103	5.63E-14
Ru-106	9.93E-09	Ru-106	1.29E-09	Ru-106	1.79E-13	Ru-106	1.29E-14

TABLE 12.2-8 (Sheet 2)

Condensate Demineralizer		High TDS Regenerant Collector Tank		Secondary Liquid Waste Drain Collector Tank		Secondary Liquid Waste Monitor Tank	
Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)	Isotope	Activity (μCi/cc)
Te-125M	2.29E-08	Te-125M	2.96E-09	Te-125M	4.47E-13	Te-125M	2.95E-14
Te-127M	2.39E-07	Te-127M	3.10E-08	Te-127M	4.48E-12	Te-127M	3.10E-13
Te-127	2.61E-07	Te-127	3.39E-08	Te-127	8.06E-12	Te-127	3.25E-13
Te-129M	1.28E-06	Te-129M	1.66E-07	Te-129M	2.66E-11	Te-129M	1.65E-12
Te-129	8.26E-07	Te-129	1.07E-07	Te-129	1.83E-11	Te-129	1.06E-12
Te-131M	2.78E-07	Te-131M	3.61E-08	Te-131M	3.08E-11	Te-131M	2.98E-13
Te-131	5.17E-08	Te-131	6.70E-09	Te-131	5.77E-12	Te-131	5.44E-14
Te-132	6.79E-06	Te-132	8.81E-07	Te-132	3.86E-10	Te-132	8.18E-12
I-129	1.44E-15	I-129	1.87E-16	I-130	2.34E-10	I-129	1.88E-20
I-130	1.45E-06	I-130	1.88E-07	I-131	7.85E-08	I-130	1.18E-11
I-131	2.45E-03	I-131	3.18E-04	I-132	2.68E-09	I-131	3.09E-08
I-132	1.98E-05	I-132	2.57E-06	I-133	6.17E-08	I-132	2.82E-11
I-133	4.60E-04	I-133	5.97E-05	I-134	2.03E-10	I-133	4.52E-09
I-134	1.14E-06	I-134	1.48E-07	I-135	1.19E-08	I-134	1.91E-14
I-135	6.77E-05	I-135	8.77E-06	Cs-134	1.20E-09	I-135	3.71E-10
Cs-134	6.00E-05	Cs-134	7.78E-06	Cs-135	7.31E-18	Cs-134	7.77E-11
Cs-135	7.70E-13	Cs-135	9.99E-14	Cs-136	5.94E-10	Cs-135	6.54E-19
Cs-136	2.04E-05	Cs-136	2.65E-06	Cs-137	8.66E-10	Cs-136	2.60E-11
Cs-137	4.37E-05	Cs-137	5.67E-06	Ba-137M	8.21E-10	Cs-137	5.67E-11
Ba-137M	4.14E-05	Ba-137M	5.36E-06	Ba-140	9.02E-12	Ba-137M	5.36E-11
Ba-140	3.42E-07	Ba-140	4.44E-08	La-140	7.56E-12	Ba-140	4.36E-13
La-140	3.63E-07	La-140	4.71E-08	Ce-141	3.70E-12	La-140	4.67E-13
Ce-141	1.77E-07	Ce-141	2.29E-08	Ce-143	6.63E-13	Ce-141	2.27E-13
Ce-143	6.34E-09	Ce-143	8.23E-10	Ce-144	1.88E-12	Ce-143	6.91E-15
Ce-144	1.03E-07	Ce-144	1.34E-08	Pr-143	1.84E-12	Ce-144	1.34E-13
Pr-143	7.36E-08	Pr-143	9.55E-09	Pr-144	1.90E-12	Pr-143	9.40E-14
Pr-144	1.03E-07	Pr-144	1.34E-08			Pr-144	1.34E-13

TABLE 12.2-9

CONSERVATIVE BASIS ACCUMULATED RADIOACTIVITY IN THE GASEOUS WASTE PROCESSING SYSTEM AFTER FORTY YEARS OPERATION

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

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

TABLE 12.2-10

FORT CALHOUN OPERATING DATA (REF. 6)
I. AVERAGE AIRBORNE RADIOACTIVITY CONCENTRATIONS
(µC/cc)

Isotope	Piping Penetration Rm.	Letdown Ht Exch Rm.	Waste Evap Rm.	Aux. Bldg. Station 1	Aux. Bldg. Station 2	Aux. Bldg. Station 3	Aux. Bldg. Station 4
I-131	2.4E-12	6.9E-12	3.9E-11	1.3E-10	2.6E-11	4.7E-12	1.9E-11
Cs-134	--	7.1E-14	4.1E-13	3.6E-13	8.0E-13	9.0E-15	8.8E-13
Cs-136	--	--	--	--	--	--	--
Cs-137	--	1.9E-13	2.1E-12	2.7E-12	2.2E-11	1.6E-12	3.1E-12
H-3	--	--	--	2.0E-9	1.1E-8	1.2E-9	1.0E-9
C-14	--	--	--	1.0E-9	1.3E-9	3.8E-10	3.0E-10
Cr-51	--	--	4.1E-13	1.1E-12	1.4E-13	3.2E-13	5.1E-13
Mn-54	--	--	1.1E-13	--	2.8E-14	9.4E-16	--
Fe-59	--	--	--	6.4E-14	--	4.3E-14	--
Co-57	--	--	6.3E-14	2.2E-12	1.3E-14	3.8E-13	3.5E-13
Co-58	--	1.8E-13	4.1E-13	6.7E-13	3.6E-13	1.6E-13	1.3E-13
Co-60	2.5E-13	1.9E-13	7.2E-14	1.4E-13	8.1E-14	3.3E-15	--
Zn-65	--	--	--	--	--	--	--
Zr-95	--	--	--	--	2.6E-14	--	--
Nb-95	--	--	--	--	1.3E-15	--	--
Ru-103	--	--	--	5.8E-14	1.7E-14	3.6E-15	--
Ru-106	--	--	2.8E-13	--	--	--	1.8E-13
Ag-110m	--	--	--	--	--	--	--
Sb-124	--	--	2.2E-13	3.1E-13	2.0E-13	7.7E-14	6.2E-14
Sb-125	--	--	1.9E-14	1.2E-13	5.7E-15	6.4E-14	3.6E-14
Ba-140	--	--	--	4.0E-13	5.1E-14	--	--
La-140	--	--	--	3.0E-12	--	--	--
Ce-141	--	--	7.2E-14	3.7E-13	9.1E-14	6.1E-13	4.4E-13
Eu-152	--	--	1.1E-13	3.2E-15	2.4E-14	7.2E-14	3.7E-13
Eu-154	--	--	3.6E-15	7.1E-13	--	--	--
Eu-155	--	--	4.8E-14	--	1.7E-14	2.0E-12	8.4E-13

TABLE 12.2-10 (Sheet 2)

FORT CALHOUN OPERATING DATA
II. RATIO OF AIRBORNE RADIOACTIVITY CONCENTRATIONS TO MPC (C/MPC)

Isotope	Piping Penetration Rm.	Letdown Ht Exch Rm.	Waste Evap Rm.	Aux. Bldg. Station 1	Aux. Bldg. Station 2	Aux. Bldg. Station 3	Aux. Bldg. Station 4
I-131	2.6E-4	7.7E-4	4.4E-3	1.5E-2	2.9E-3	5.2E-4	2.1E-3
Cs-134	--	7.1E-6	4.0E-5	3.6E-5	7.9E-5	9.0E-7	8.8E-5
Cs-136	--	--	--	--	--	--	--
Cs-137	--	1.9E-5	2.1E-4	2.7E-4	2.2E-3	1.6E-4	3.1E-4
H-3	--	--	--	4.0E-4	2.1E-3	1.6E-4	3.1E-4
C-14	--	--	--	2.5E-4	3.4E-4	9.5E-5	7.4E-5
Cr-51	--	--	2.0E-7	5.3E-7	7.0E-8	1.6E-7	2.6E-4
Mn-54	--	--	2.8E-6	--	7.0E-7	2.3E-8	--
Fe-59	--	--	--	1.3E-6	--	8.5E-7	--
Co-57	--	--	3.2E-7	1.1E-5	6.3E-8	1.9E-6	1.8E-6
Co-58	--	3.5E-6	8.3E-6	1.3E-5	7.0E-6	3.2E-6	2.7E-6
Co-60	2.8E-5	2.1E-5	8.0E-6	1.5E-5	9.0E-6	3.7E-7	--
Zn-65	--	--	--	--	--	--	--
Zr-95	--	--	--	--	8.7E-7	--	--
Nb-95	--	--	--	--	1.3E-8	--	--
Ru-103	--	--	--	7.3E-7	2.2E-7	4.5E-8	--
Ru-106	--	--	4.7E-5	--	--	--	2.9E-5
Ag-110m	--	--	--	--	--	--	--
Sb-124	--	--	1.1E-5	1.5E-5	1.0E-5	3.8E-6	3.1E-6
Sb-125	--	--	6.4E-7	4.1E-6	1.9E-7	2.2E-6	1.2E-6
Ba-140	--	--	--	1.0E-5	1.3E-6	--	--
La-140	--	--	--	3.0E-5	--	--	--
Ce-141	--	--	3.6E-7	1.9E-6	4.5E-7	3.0E-6	2.2E-6
Eu-152	--	--	5.7E-6	1.6E-7	1.2E-6	3.6E-6	1.8E-5
Eu-154	--	--	9.0E-7	1.8E-4	--	--	--
Eu-155	--	--	6.9E-7	--	2.5E-7	2.8E-5	1.2E-5

NOTES:

1. The concentrations shown are average concentrations for all samples at each location analyzed for each particular radioisotope.

2. Dash indicates that the radioisotope was not detected on any sample at that location.

3. The following list (Table 12.2-10, Sheet 3) gives the areas in the Fort Calhoun auxiliary building served by the four sampling stations. Investigation revealed that the sampling room was the principal source of airborne radioactivity in Station 1, and that the charging pump room was the principal source of airborne radioactivity in Station 4. However, Station 1 samples were drawn from the ventilation duct into which the hood from the sampling station was exhausted. Consequently, the atmosphere in the sample room had much lower concentration. Furthermore, occupancy in the sampling room is limited. As for the charging pump room, this is an E Zone, and normally, no occupancy is expected in an E Zone.

TABLE 12.2-10 (Sheet 3)

FORT CALHOUN OPERATING DATA
III. AUXILIARY BUILDING SAMPLING STATION FEEDS (REF. 6)

Station 1

1. Letdown heat exchanger room
2. Mechanical penetration area
3. Shutdown heat exchanger room
4. Valve room
5. Pipe penetration area
6. Personnel air lock area
7. Sampling room*

Station 2

1. Cask decon. room
2. Fuel arrival area
3. Fuel storage area
4. Drum storage area
5. Waste baler room
6. Spent resin storage room
7. Volume control tank room
8. Waste evaporator room
9. Waste holdup tank rooms
10. Spent fuel heat exchanger room
11. Safety injection pump rooms
12. Charging pump room
13. Charging pump valve room
14. Fuel pool area

Station 3

1. Waste decay tank rooms
2. Shutdown cooling heat exchanger room
3. Shutdown cooling heat exchanger valve room
4. Component heat exchanger room

Station 4

1. Spent fuel heat exchanger room
2. Safety injection pump rooms
3. Charging pump room*
4. Charging pump valve room

*Principal source in the station.

TABLE 12.2-11

PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVE CONCENTRATIONS (REF. 7)

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

TABLE 12.2-11 (Sheet 2)

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

	CFM
D. Ventilation Rates	
1. Containment during power	4,000
2. Containment during refueling	20,000
3. Fuel building during power	20,000
4. Fuel building during refueling	20,000 (Note 1)

	CF
E. Volumes of the Regions	
1. Containment	2.5×10^6
2. Fuel building	8.2×10^5

	Hrs/yr
F. Maximum Annual Individual Occupancy	
1. Containment during power 5 hr/wk for 50 wks/year	250
2. Containment during fuel handling 10 hr/day for ~ 6.25 days when the refueling pool is full of water	62.5
3. Fuel building during power 5 hr/week for 50 weeks/year	250
4. Fuel building during refueling 10 hrs/day for ~ 10 days and 8 hrs/day for ~ 3 days	125

G. Miscellaneous Information	
1. Failed fuel percentage for fission products	0.12
2. Reactor coolant specific activities	Table 11.1-1

TABLE 12.2-11 (Sheet 3)

3. Refueling and fuel storage pool concentrations (Ci/gm). (These are the maximum concentrations during refueling)	I-131 3.21×10^{-5}	I-133 2.32×10^{-6}
4. Fuel storage pool cleanup rate during power (gpm) (for conservatism, no cleanup of fuel storage pool or refueling pool is assumed during refueling)	300	
5. Decay of isotopes in the pools during refueling	Not included for conservatism	
6. Duration of refueling pool evaporation (hrs)	150	
7. Duration of fuel storage pool evaporation during refueling (hrs)	320	
8. Duration of fuel storage pool evaporation during power (hours/year)	8440	
9. Tritium release to environment via containment ventilation exhaust during refueling (Ci) (based on a total tritium release of 1000 Ci via gaseous effluents, durations of evaporation quoted in items 6, 7 and 8, and evaporation rates from pools given in Item B)	24	
10. Tritium release to environment via fuel building ventilation exhaust during refueling (Ci) (same bases as given for Item 9)	104	
11. Tritium release to environment via fuel building ventilation exhaust during power (Ci) (same bases as given for Item 9)	1094	

NOTES:

1. The emergency exhaust flow rate may vary resulting in proportionally higher or lower airborne concentrations in the fuel building. Any difference is insignificant when compared to the allowable limits of 10 CFR 20.

TABLE 12.2-12

AIRBORNE RADIOACTIVITY CONCENTRATIONS

(μCi/cc)

Nuclide	Containment (power)	Containment (refueling)	Fuel Building (power)	Fuel Building (refueling)
H-3	2.34E-7	4.7E-6	3.10E-6	1.66E-5
Cr-51	1.25E-12	negligible	negligible	negligible
Mn-54	2.07E-13	negligible	negligible	negligible
Fe-55	1.07E-12	negligible	negligible	negligible
Fe-59	6.63E-13	negligible	negligible	negligible
Co-58	1.06E-11	negligible	negligible	negligible
Co-60	1.33E-12	negligible	negligible	negligible
Br-83	1.76E-11	negligible	negligible	negligible
Br-84	2.61E-12	negligible	negligible	negligible
Kr-83m	6.60E-8	negligible	negligible	negligible
Kr-85m	6.02E-7	negligible	negligible	negligible
Kr-85	1.18E-7	negligible	negligible	negligible
Kr-87	1.39E-7	negligible	negligible	negligible
Kr-88	8.34E-7	negligible	negligible	negligible
Kr-89	5.83E-10	negligible	negligible	negligible
Rb-88	8.01E-7	negligible	negligible	negligible
Rb-89	5.63E-10	negligible	negligible	negligible
Sr-89	3.54E-12	negligible	negligible	negligible
Sr-91	2.49E-13	negligible	negligible	negligible
Mo-99	5.06E-11	negligible	negligible	negligible
Te-127m	1.86E-13	negligible	negligible	negligible
Te-127	4.01E-13	negligible	negligible	negligible
Te-129m	9.26E-13	negligible	negligible	negligible
Te-129	6.58E-13	negligible	negligible	negligible
Te-131m	1.34E-12	negligible	negligible	negligible
Te-131	2.69E-13	negligible	negligible	negligible
Te-132	1.65E-11	negligible	negligible	negligible
I-130	1.95E-11	negligible	negligible	negligible
I-131	3.82E-9	2.14E-10	negligible	7.6E-10
I-132	3.66E-10	negligible	negligible	negligible
I-133	4.14E-9	1.55E-11	negligible	5.5E-11
I-134	7.41E-11	negligible	negligible	negligible
I-135	1.34E-9	negligible	negligible	negligible
Xe-131m	2.75E-7	negligible	negligible	negligible
Xe-133m	1.35E-6	negligible	negligible	negligible
Xe-133	7.24E-5	negligible	negligible	negligible
Xe-135m	7.64E-9	negligible	negligible	negligible
Xe-135	2.55E-6	negligible	negligible	negligible
Xe-137	1.26E-9	negligible	negligible	negligible
Xe-138	2.22E-8	negligible	negligible	negligible

TABLE 12.2-12 (Sheet 2)

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

NOTES:

1. Iodine airborne concentrations during refueling are calculated very conservatively, assuming no purification by fuel pool cleanup system and no decay in the pool and a partition factor of 1 at the water-air interface at pool surface.

2. Continuous pool cleanup, decay in the pool, and lower evaporation rates are expected to reduce iodine air concentrations to negligible levels during power operation in the fuel building.

   Xe-133 and I-131 air concentrations in the containment ventilation exhaust duct are expected to be <2 x 10^{-2} and 1.6 x 10^{-6} Ci/cc, respectively during reactor head venting. However, the containment airborne concentrations for these isotopes are expected to be significantly less during head venting since the radioactivity is directly piped to the exhaust duct. The maximum value for Xe-133 is based on operating plant measurements normalized for a reactor system which has continuous stripping of the noble gases in the volume control tank. The maximum value for I-131 is based on operating plant measurements.

3. Airborne concentrations in the fuel building are based upon 9,000 cfm ventilation exhaust airflow rate during refueling. Actual flow rate may be lower resulting in proportionally higher concentrations. Any difference is insignificant when compared to the allowable limits of 10 CFR 20.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

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

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

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

12.3.1.1.1 Common Equipment and Component Designs for ALARA

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

FILTERS - To reduce exposure, spent liquid system radioactive filters are normally handled with a remote tool during removal from the filter housing and during transfer for packing and shipment from the site for disposal. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. Each filter vessel is located within individual shielded compartments, provided with a ventilation supply and return and compartment drainage capabilities. The ventilation return ducts are equipped with a removable access panel through which a portable radiation monitor can be lowered into the filter compartment to obtain radiation levels within the compartment. Care has been taken to ensure that adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste building.

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

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

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

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

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

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

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

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

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

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

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

All manually-operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown have provisions for reach rods extending through or over the valve gallery wall. The valve gallery shield walls are designed for maximum expected filter activities.

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

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

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

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

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

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

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

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

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

12.3.1.1.2 Common Facility and Layout Designs for ALARA

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

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

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

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

Potentially radioactive piping is located in appropriate radiation design zone and restricted areas. Process piping is monitored where required to ensure that access is controlled to limit exposure (Section 12.5).

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Exposure from routine in-plant inspection is controlled by locating, whenever possible, inspection points in properly shielded low background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is provided. In high radiation areas where routine inspection is required, remote viewing devices may be used, as needed. When this is not practicable, access to the high radiation areas are under the direct supervision of the unit health physics personnel in accordance with approved radiation work permits, when applicable.

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

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yards is regulated and controlled as described in Section 12.5.2. Plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Each room, corridor, and pipeway of the plant buildings is evaluated for potential radiation sources during normal operation, shutdown, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. These criteria were then used as the basis for the radiation shielding design. Radiation zone categories employed and their descriptions are given in Figure 12.3-2 (Sheet 1), and the specific zoning for each plant area is shown in Figure 12.3-2 (Sheets 1-6). All frequently accessed areas, i.e., corridors, were shielded for Zone A or Zone B access. Periodically dose levels in various areas will exceed their radiation zone levels from plant conditions or stored items. During plant operation and refueling conditions the Health Physics staff will evaluate area access, monitor entry into areas, and update posting and entry requirements in accordance with 10 CFR 20. Quality Control and Procurement Quality may update postings that are required to perform Radiography and metal examination.

The control of ingress or egress of plant operating personnel to controlled access areas and procedures employed to ensure that radiation levels and allowable working time are within the limits prescribed by 10 CFR 20 are described in Sections 12.5.2 and 12.5.3.

Any area accessible to individuals which could result in an individual receiving a dose equivalent in excess of 5 mrem in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, is posted with signs bearing the radiation symbol and the words, "CAUTION, RADIATION AREA." Access alert barriers (e.g. signs, chain, rope, door, etc.) are provided for radiation areas where practicable. Locations of these barriers are shown on Figure 12.3-2. Any area accessible to individuals which could result in an individual receiving a dose equivalent in excess of 100 mrem in one (1) hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates is posted with the radiation symbol and the words, "CAUTION, HIGH RADIATION AREA." High radiation areas are kept locked or barricaded if greater than 1000 mrem in one hour at 30 centimeters except during periods when access to the area is required in which case positive control is exercised over each individual entry. For these areas, in excess of 1000 mrem in one hour at 30 centimeters, that are located within large areas, such as containment, where no enclosure exists and where no enclosure can be reasonably constructed around the individual area, that area shall be barricaded, conspicuously posted and a flashing red light shall be activated as a warning device. Any area accessible to individuals which could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from the source or from any surface that the radiation penetrates shall be locked or barricaded and posted "GRAVE DANGER, VERY HIGH RADIATION AREA."

Whenever practicable, the measured radiation level and the location of the source is posted at the entry to radiation or high radiation areas.

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

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

12.3.2 SHIELDING

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

12.3.2.1 Design Objectives

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

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

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

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

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

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

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

12.3.2.2 General Shielding Design

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

The material used for most of the plant shielding is ordinary concrete with a minimum bulk density of 147 lb/ft3. Whenever poured-in-place concrete has been replaced by concrete blocks or other material, design assures protection on an equivalent shielding basis, as determined by the characteristics of the material selected (Ref. 1). Compliance of concrete radiation shield design with Regulatory Guide 1.69 is discussed in Appendix 3A. Water is used as the primary shield material for areas above the spent fuel transfer and storage areas.

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

12.3.2.2.1 Reactor Building Interior Shielding Design

During reactor operation, several areas inside the reactor building are High Radiation or Very High Radiation areas and normally inaccessible.

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

The reactor cavity shield design utilizes two types of shielding material contained in neutron shield segments hung from the support bars attached to the bottom of the seal plate of the permanent cavity seal ring (PCSR) at the RPV seal ledge. The bottom four inches of the shield is constructed of Reactor Experiments Type 207 material which is a flame resistant borated polyethylene. The upper ten inches is type 277 which is a refractory material that resembles concrete in its final form. This shield configuration possesses bulk shielding characteristics superior to the shielding provided by the analyzed 12 inch thick water shield. MORSE analyses have been performed which predict that limited personnel access may be allowed at the operating floor during power operation. The MORSE analysis was performed using a water bag design. Since the shielding properties were not reduced by replacing water bags with the configuration described above, the analysis is still considered valid.

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

After shutdown, most of the containment is accessible for limited periods of time, and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected, but not limited, to range from 0.5 to 1,000 mrem/hr, depending on the location inside the containment (excluding reactor cavity). These dose rates result from residual fission products, neutron-activated materials, and corrosion products in the reactor coolant system.

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

12.3.2.2.2 Auxiliary Building Shielding Design

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

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

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

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

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

12.3.2.2.3 Fuel Building Shielding Design

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

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

12.3.2.2.4 Radwaste Building Shielding Design

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

a. Liquid waste collection tanks and pumps

b. Liquid waste monitor tanks and pumps

c. Chemical drain tank and pump

d. Liquid waste, boron recycle, and secondary liquid waste evaporators

e. Liquid Radwaste Demineralizer Skid

f. Solid radwaste disposal and storage areas

g. Evaporator bottoms tanks

h. Liquid radwaste piping

i. Radwaste filters and demineralizers

j. Spent resin storage tank and pump

k. Gaseous radwaste surge tank, recombiners, and compressors

l. Gas decay tanks

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

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

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

12.3.2.2.5 Turbine Building Shielding Design

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

12.3.2.2.6 Control Room Shielding Design

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

12.3.2.2.7 Diesel Generator Building Shielding Design

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

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is designed for all plant buildings containing radiation sources so that anticipated radiation levels at the accessible outside surfaces of the buildings are maintained to meet Zone A levels. Plant yard areas which are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are within the Radiation Control Area boundary and closed off from areas accessible to the general public. The Radiation Control Area boundary fence is maintained less than or equal to 0.6 mrem/hr.

12.3.2.3 Shielding Calculational Methods

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

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

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall. The shielding thickness was designed such that the aggregate postulated radiation level from all contributing sources, at this point, would be attenuated to meet the criteria of the adjoining radiation zone specifications.

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

12.3.3 VENTILATION

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

12.3.3.1 Design Objectives

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

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

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

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

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

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

12.3.3.3 Design Guidelines

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

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

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

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

c. All-welded piping systems are employed on contaminated systems, to the maximum extent practicable, to reduce system leakage.

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

e. To minimize the amount of airborne radioactivity as a result of valve leakage, all valves 2-1/2 inches and larger in the radioactive systems are provided with graphite packing. Diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated.

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

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

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

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

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

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

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

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

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

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

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment

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

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

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

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

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

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

12.3.3.4 Design Description

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

a. Containment building (see Section 9.4.6)

b. Auxiliary building (see Section 9.4.3)

c. Fuel building (see Section 9.4.2)

d. Radwaste building (see Section 9.4.5)

e. Turbine building (see Section 9.4.4)

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

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

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

12.3.3.5 Air Cleaning System Design

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

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

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

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

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

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

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

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

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitoring

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

12.3.4.1.1 Design Bases

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

12.3.4.1.1.1 Safety Design Bases

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

12.3.4.1.1.2 Power Generation Design Bases

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

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

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

POWER GENERATION DESIGN BASIS FOUR - Certain monitors located near the fuel storage pool, new fuel storage vault, and cask handling area were originally installed to act as criticality alarm monitors in conformance with the requirements of 10 CFR 70, Regulatory Guides 8.5 and 8.12 and Standards ANSI/ANS-8.3-1979 and USAS N2.3-1967. The NRC issued an exemption to the requirements of 10CFR70.24 to WCNOC on June 24,1997. On November 12, 1998 the NRC issued 10CFR50.68, which provides eight criteria that may be followed in lieu of criticality monitoring per 10CFR70.24. One of these criteria require that radiation monitors are provided in storage areas when fuel is present to detect excessive radiation levels. These monitors will remain in place to provide prompt warning of high radiation. The monitors provide a hi-hi radiation alarm of 15 mrem/hr which will give prompt warning if high radiation occurs. These monitors are provided in accordance with GDC-63.

12.3.4.1.1.3 Codes and Standards

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

12.3.4.1.2 System Description

12.3.4.1.2.1 General Description

The ARMS consists of five-decade range GM tube detectors located throughout the plant to warn personnel of abnormal gamma radiation levels. The detector signals are transmitted to the control room over individual cables. The displays, both local and in the control room, are five-decade logarithmic ratemeters. The alarms are both audible and visual, and are located in the control room and near the local detectors. Two ARMS (Technical Support Center (TSC) SDRE0043 and Emergency Offsite Facility (EOF) SDRE0044) are stand-alone units and do not transmit a signal or alarms to the control room.

12.3.4.1.2.2 Criteria for Area Monitor Selection

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

RANGE - The ARMS has a five-decade range from 10-1 to 10+4 mrem/hr. Except the Post-Accident sample room radiation monitor which has a range from 10 to 105 mrem/hr. The ranges are made sufficiently wide to measure the radiation levels expected in the areas concerned. The system continues to read upscale if exposed to radiation levels above the maximum range.

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

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

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

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

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

For instruments located outside the containment:

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

For instruments located inside the containment:

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

EXPOSURE LIFE - Each monitor located inside the containment maintains its characteristics up to an integrated dose of 107 rads. Each monitor located outside the containment maintains its characteristics up to an integrated dose of 106 rads.

12.3.4.1.2.3 Alarms

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

12.3.4.1.2.4 Check Sources

Each monitor is provided with a check source, operated from the control room, which simulates a radiation level in the area for operational and gross calibration checks. The check source for most monitors can be operated from the control room. However, the TSC Monitor (SDRE0043) and the EOF Monitor (SDRE0044) must be operated locally.

12.3.4.1.2.5 Power Supplies

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

12.3.4.1.2.6 Calibration and Maintenance

The area radiation monitors are calibrated by the manufacturer, using a Cs-137 source. The manufacturer's calibration standards are traceable to National Institute of Standards and Technology primary calibration standard sources and are accurate to at least 5 percent. The source-detector geometry during this primary calibration is identical to the source-detector geometry in field calibrations. A secondary standard counted in reproducible geometry during the primary calibration is supplied with each monitoring system. The frequency of calibration is established in station procedures.

The count rate response of each monitor to the remotely positionable check source supplied with each monitor is recorded by the manufacturer after the primary calibration.

Check source response and counter background is maintained in accordance with station procedures. Following repairs or modifications, the monitors are recalibrated at the plant with the secondary radionuclide standard.

12.3.4.1.2.7 Sensitivities

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

12.3.4.1.2.8 Criteria for Location of Area Monitors

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

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

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

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

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

a. Main control room

b. Radwaste building corridors

c. Auxiliary building corridors

d. Fuel storage and handling area

e. Radwaste pipe tunnel

f. Railway access (fuel building)

g. Hot machine and hot instrument shops

h. Containment

i. Radwaste solidification area

j. Sampling rooms and laboratory

12.3.4.1.2.9 Setpoints

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

The setpoints for the individual area radiation monitors are provided in Table 12.3-2.

12.3.4.1.2.10 Safety Evaluation

The ARMS is designed to operate unattended for extended periods of time, detecting and measuring ambient gamma radiation. Ambient radiation dose rate at the detector is indicated locally at the detector and remotely in the main control room for most detectors. Most of these monitors cause an audible and visual alarm at the detector and in the main control room if the radiation levels exceed preset limits. However, the TSC Monitor (SDRE0043) and the EOF Monitor (SDRE0044) only provide local indication and alarm. All components are solid state, and the system is designed for high reliability.

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

12.3.4.2 <u>Airborne Radioactivity Monitoring</u>

Monitoring for the presence of airborne radioactivity inside the plant is necessary for the protection of plant personnel, in compliance with Regulatory Guide 8.2 and within the limits established by 10 CFR 20.

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

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

12.3.4.2.1 Design Bases

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

12.3.4.2.1.1 Safety Design Bases

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

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

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

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

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

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

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

12.3.4.2.1.2 Power Generation Design Bases

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

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

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

12.3.4.2.1.3 Codes and Standards

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

12.3.4.2.2 System Description

12.3.4.2.2.1 General Description

12.3.4.2.2.1.1 Data Collection

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

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

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

12.3.4.2.2.1.2 Selection Criteria for Airborne Monitors

12.3.4.2.2.1.2.1 Introduction

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

12.3.4.2.2.1.2.2 Sampling Criteria

The sampling system for the particulate/halogen/noble gas monitors is designed and installed in accordance with ANSI N13.1-1969, Guide to Sampling of Airborne Radioactive Materials. Systems whose sensitivity is dependent upon sample flow employ isokinetic nozzles and suitable control of the flow rate.

12.3.4.2.2.1.2.3 Detection Criteria

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

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

12.3.4.2.2.1.2.4 Instrumentation Criteria

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

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

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

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

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

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

12.3.4.2.2.1.3 Criteria for Airborne Radioactivity Monitor Locations

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

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

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

c. Exhaust ducts servicing an area containing processes which, in the event of major leakage, could result in concentrations within the plant approaching the limits established by 10 CFR 20 for plant workers are monitored.

d. Dilution from other exhaust ducts is considered when locating monitors in exhaust systems to ensure maximum coverage and still be able to detect 10 CFR 20 airborne radioactivity limits in the area with the lowest ventilation flow.

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

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

12.3.4.2.2.1.4 Alarms

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

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

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

12.3.4.2.2.1.5 Check Sources

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

12.3.4.2.2.1.6 Power Supplies

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

12.3.4.2.2.1.7 Calibration and Maintenance

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

The count rate response of each continuous monitor to remotely positionable check sources supplied with each monitor is recorded by the manufacturer after the primary calibration. Check sources response and counter background is maintained in accordance with station procedures. Following repairs or modifications, the monitors are recalibrated at the plant with the secondary radionuclide standards.

12.3.4.2.2.1.8 Sensitivities

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

The most restrictive isotope for each type of monitor is that isotope with the lowest worker derived air concentration (WDAC), as defined in Table 1, Column 3, of Appendix B to 10 CFR 20.1001 - 20.2402.

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

The sensitivities and alarm setpoints are given in Table 12.3-3. The High alarm points are based on the most restrictive isotopes which are expected to be present. The concentration levels are as defined in Table 1, Column 3, of Appendix B to 10 CFR 20.1001 - 20.2402 or Technical Specification limits, considering dilution.

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

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

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

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

12.3.4.2.2.1.9 Ranges and Setpoints

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

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

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

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

The setpoints are chosen to alert the operators to airborne radioactivity that might be present so that 10 CFR 20 limits on worker exposure or Technical Specification limits are not exceeded.

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

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

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

12.3.4.2.2.1.10 Expected System Parameters

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

12.3.4.2.2.2 Monitoring Systems

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

12.3.4.2.2.2.1 Access Control Area Ventilation Exhaust Radioactivity Monitor

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

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

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

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

12.3.4.2.2.2.2 Radwaste Building Ventilation Exhaust Radioactivity Monitor

The radwaste building ventilation exhaust radioactivity monitor, 0-GH-RE-22, continuously monitors for particulate radioactivity in the exhaust duct upstream of the radwaste exhaust filter adsorber. The sample is extracted from the duct through an isokinetic nozzle, in accordance with ANSI Standard N13.1-1969, to ensure that a representative sample is obtained. After passing through the fixed filter detector assembly and the pumping system, the sample is discharged back to the duct. The cartridge filter is removed periodically for laboratory isotopic analyses. Monitoring upstream of the filter adsorber provides the most rapid response to airborne radioactivity in the system.

The high and high-high alarms function to alert the operator to airborne particulate radioactivity in the radwaste building. Indication for this monitor is provided on the AiRMS CRT in the control room.

If required, grab samples are utilized to determine airborne radioactivity levels and iodine concentrations in specific areas to aid in the determination of the source of the release.

12.3.4.2.2.2.3 Waste Gas Decay Tank Area Ventilation Exhaust Radioactivity Monitor

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

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

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

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

12.3.4.2.2.2.4 Auxiliary Building Ventilation Exhaust Reactivity Monitor

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

The cartridge filter can be removed for laboratory isotopic analyses.

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

If required, grab samples can be utilized to determine airborne radioactivity levels and iodine concentrations in specific areas to aid in the determination of the source of the release.

12.3.4.2.2.2.5 Containment Atmosphere Radioactivity Monitors

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

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

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

12.3.4.2.2.2.6 Containment Purge System Radioactivity Monitors

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

The purge monitors are seismic Category I and completely redundant.

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

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

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

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

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

12.3.4.2.2.2.7 Control Room Ventilation Radioactivity Monitors

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

This seismic Category I system is completely redundant.

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

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

12.3.4.2.2.2.8 Fuel Building Ventilation Exhaust Radioactivity Monitors

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

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

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

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

12.3.4.2.2.3 Safety Evaluation

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

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

a. Containment atmosphere

b. Containment purge

c. Control room supply

d. Fuel building exhaust

e. Auxiliary building exhaust

f. Radwaste building exhaust

g. Access control area exhaust

h. Waste gas decay tank room exhaust

i. Portable monitor

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

The AiRMS is adequate and sufficient to ensure personnel protection from airborne radioactivity. The system provides indication to the operator that airborne radioactivity exists in several possible areas. The location of the airborne radioactivity can then be further identified by using portable air samplers to collect general area air samples.

The combination of the AiRMS, in conjunction with administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns throughout the

plant, plant equipment layout, and restricted radiation Zone E areas, is sufficient to ensure that airborne radioactivity levels are safe in terms of the required duration of personnel access throughout all areas of the plant.

A general review of these concepts follows:

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

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

c. Prior to entry for work in airborne, or potentially airborne areas, Health Physics will take appropriate air samples. Authorization must be obtained before entry. Prior to entry, a high-volume portable air sampler may be used by the health physics group to collect a representative air sample. Gaseous, iodine, and particulate activity of the area will be analyzed before entry as applicable.

d. Health physics programs are discussed in Section 12.5.

12.3.5 REFERENCES

1. R. L. Walker and M. Grotenhuis, A Summary of Shielding Constants for Concrete, ANL-6443 (November 1961).

2. C. M. Lederer, et. al., Table of Isotopes, Lawrence Radiation Laboratory, University of California (March 1968).

3. R. G. Jaeger, et. al., Engineering Compendium on Radiation Shielding, Shielding Fundamentals and Methods, I (1968).

4. G. W. Goldstein, X-Ray Attenuation Coefficients from 10 keV to 100 MeV, National Bureau of Standards Circular 583 (issued April 10, 1957).

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

6. J. F. Kircher and R. E. Bowman, Effects of Radiation on Materials and Components (March 1964).

7. T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., New York (1956).

8. C. R. Tipton, Jr., Reactor Handbook, Vol. I, Materials, second edition (1962).

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

10. E. P. Blizzard and L. S. Abbott, Reactor Handbook Vol. 111, Part B, Shielding, second edition (1962).

11. N. M. Schaeffer, Reactor Shielding for Nuclear Engineers, TID-25951 (1967).

12. D. S. Duncan and A. B. Spear, Grace I -An IBM 704-709 Program Design for Computing Gamma Ray Attenuation and Heating in Reactor Shields, Atomics International (June 1959).

13. D. S. Duncan and A. B. Spear, Grace II - An IBM 709 Program for Computing Gamma Ray Attenuation and Heating in Cylindrical and Spherical Geometries, Atomics International (November 1959).

14. W. W. Engle, Jr., A User's Manual for ANISN: A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering, Union Carbide Corporation, Report No. K-1693 (1967).

15. E. D. Arnold and B. F. Maskewitz, SDC, A Shielding Design Code for Fuel Handling Facilities, ORNL-3041 (March 1966).

16. Richard E. Malenfant, "QAD, A Series of Point-Kernel General-Purpose Shielding Programs," Los Alamos Scientific Laboratory, LA 3573 (October 1966).

17. D. A. Klopp, NAP - Multigroup Time-Dependent Neutron Activation Prediction Code, IITRI-A6088-21 (January 1966).

18. E. A. Straker, P. N. Stevens, D. C. Irving, and V. R. Crain, MORSE - A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code, ORNL-4585 (September 1970).

19. W. A. Rhoades and F. R. Mynatt, The DOT III Two-Dimensional Discrete Ordinates Transport Code, ORNL-TM-4280 (1973).

20. R. E. Malenfant, "G3 A General Purpose Gamma-Ray Scattering Program," Los Alamos Scientific Laboratory, LA 5176 (June 1973).

21. M. J. Bell, "ORIGEN - The ORNL Generation and Depletion Code," Oak Ridge National Laboratory, ORNL-4628 (May 1973).

22 Letter from J. C. Stone, USNRC, to O. L. Maynard, WCNOC dated June 24, 1997, Request For Exemption From 10 CFR 70.24 Criticality Monitoring Requirements - Wolf Creek Generating Station (TAC NO. M89161).

TABLE 12.3-1

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

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

WOLF CREEK

TABLE 12.3-2

AREA RADIATION MONITORS

Instrument Number	Location	Radiation Zone	Range (mrem/hr)	Hi Alarm (mrem/hr)	Hi-Hi Alarm (mrem/hr)
O-SD-RE-1	Radwaste Building Corridor, Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-2	Radwaste Building Corridor, Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-3	Radwaste Building Corridor, Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-4	Radwaste Building Corridor, Ground Floor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-5	Radwaste Building Corridor, Ground Floor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-6	Solid Radwaste Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-7	Truck Space	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-8	Sample Laboratory	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-9	RW Bldg Valve Room Corridor	C	$0.1 - 10^4$	15 (1)	100 (2)
O-SD-RE-10	RW Bldg Valve Room Corridor	C	$0.1 - 10^4$	15 (1)	100 (2)
O-SD-RE-11	RW Bldg HVAC Filter Unit	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-12	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-13	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-14	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-15	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-16	Aux Bldg Corridor Basement	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-17	Pipe Tunnel & Personnel Access	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-18	Aux Bldg Ground Floor Corridor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-19	Aux Bldg Ground Floor Corridor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-20	Aux Bldg Valve Room Corridor Ground Floor	C	$0.1 - 10^4$	15 (1)	100 (2)
O-SD-RE-21	Aux Bldg Valve Room Corridor Ground Floor	C	$0.1 - 10^4$	15 (1)	100 (2)
O-SD-RE-22	Aux Bldg Corridor Ground Floor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-23	Aux Bldg Corridor Ground Floor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-24	RC Sample Room	C	$0.1 - 10^4$	15 (1)	100 (2)
O-SD-RE-25	Filter Unit Aux Bldg	A	$0.1 - 10^4$	0.5 (1)	2.5 (2)
O-SD-RE-26	RHR Heat Exchanger Outside	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-27	Ctmt Purge Filter Unit	C	$0.1 - 10^4$	15 (1)	100 (2)
O-SD-RE-28	Personnel Hatch	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-29	Decontamination Room	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-30	Hot Instrument Shop	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-31	Hot Laboratory	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-32	Control Bldg Corridor	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-33	Control Room	A	$0.1 - 10^4$	0.5 (1)	2.5 (2)
O-SD-RE-34	Cask Handling Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-35	New Fuel Storage Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-36	New Fuel Storage Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-37	Fuel Storage Pool Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-38	Fuel Storge Pool Area	B	$0.1 - 10^4$	2.5 (1)	15 (2)
O-SD-RE-39	Seal Table Area	E	$0.1 - 10^4$	1000 (4)	10000 (5)
O-SD-RE-40	Personnel Access Hatch Area	E	$0.1 - 10^4$	1000 (4)	10000 (5)
O-SD-RE-41	Manipulator Bridge Crane	E	$0.1 - 10^4$	30 (6)	10000
O-SD-RE-42	Containment Building	E	$0.1 - 10^4$	1000 (4)	10000 (5)
O-SD-RE-43	Technical Support Center	A	$0.1 - 10^4$	0.5 (1)	2.5 (2)
O-SD-RE-44	Emergency Offsite Facility	A	$0.1 - 10^4$	0.5 (1)	2.5 (2)
O-SD-RE-47	Pass Sampling Room	E	$10 - 10^5$	1000 (4)	10000 (5)

(1) High alarm set for maximum radiation level for the radiation zone for that area.
(2) High-high alarm set for the maximum radiation level for the radiation zone above the one for that area.
(3) Deleted
(4) High alarm based on highest expected radiation level for the area during full power operation.
(5) High-high alarm set one order of magnitude above high alarm to indicate extremely high radiation level.
(6) High alarm is established for the protection of operators on the bridge crane and is set above background radiation levels.

TABLE 12.3-3

INPLANT AIRBORNE RADIOACTIVITY MONITORS

Monitor	Type (continuous)	Range (μCi/cc)	MDC (1) (μCi/cc)	Controlling Isotope	Hi Alarm (μCi/cc)	Hi-Hi Alarm (μCi/cc)	Total Ventilation Flow (cfm)	Subcompartment Flow Rate (cfm)	Dilution Factor	Minimum Required Sensitivity (μCi/cc)	Monitor Control Function
O-GT-RE-31 O-GT-RE-32 Containment atmosphere monitors	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1 X 10^{-11} 1 X 10^{-10} 2 X 10^{-7}	Cs-137 I-131 Xe-133	1 x 10^{-8}(8) 2 x 10^{-8}(8) 2.06 x 10^{-4}(11)	1 x 10^{-7}(7) 2 x 10^{-7}(7) 2.06 x 10^{-3}(12)	420,000 420,000 420,000	NA NA NA	NA NA NA	1 X 10^{-7} (6) 2 X 10^{-7} (6) 1 X 10^{-3} (6)	See Table 11.5-3 for process control functions & alarms for personnel protection.
O-GT-RE-22 O-GT-RE-33 Containment purge system monitors	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1 X 10^{-11} 1 X 10^{-10} 2 X 10^{-7}	Cs-137 I-131 Xe-133	1 x 10^{-8}(8) 2 x 10^{-8}(8) (10)	1 x 10^{-7}(7) 2 x 10^{-7}(7) (10)	20,000/4,000 20,000/4,000 20,000/4,000	NA NA NA	NA NA NA	1 X 10^{-7} (6) 2 X 10^{-7} (6) 1 X 10^{-3} (6)	See Table 11.5-3 for process control functions & alarms for personnel protection.
O-GG-RE-27 O-GG-RE-28 Fuel building exhaust monitors (2)	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1 X 10^{-11} 1 X 10^{-10} 2 X 10^{-7}	Cs-137 I-131 Xe-133	1 X 10^{-8}(8) 2 X 10^{-8}(8) 1.62 X 10^{-4}(13)	1 X 10^{-7}(7) 2 X 10^{-7}(7) 1.62 X 10^{-3}(14)	20,000 20,000 20,000	NA NA NA	NA NA NA	1 X 10^{-7} (6) 2 X 10^{-7} (6) 1 X 10^{-3} (6)	See Table 11.5-3 for process control functions & alarms for personnel protection.
O-GK-RE-04 O-GK-RE-05 Control room air supply monitors	Particulate (3) Iodine (4) Gaseous (3)	10-12 to 10-7 10-11 to 10-6 10-7 to 10-2	1 X 10^{-11} 1 X 10^{-10} 2 X 10^{-7}	Cs-137 I-131 Xe-133	1 X 10^{-8}(8) 2 X 10^{-8}(8) 1.35 X 10^{-4}(15)	1 X 10^{-7}(7) 2 X 10^{-7}(7) 1.35 X 10^{-3}(16)	1950 1950 1950	NA NA NA	NA NA NA	1 X 10^{-7} (6) 2 X 10^{-7} (6) 1 X 10^{-3} (6)	See Table 11.5-3 for process control functions & alarms for personnel protection.
O-GL-RE-60 Auxiliary bldg. ventilation exhaust monitor	Particulate (3)	10^{-12} to 10^{-7}	1 X 10^{-11}	Cs-137	1 X 10^{-8} (8)	1 X 10^{-7} (17)	12,000	100	8X10^{-3}(5)	1 X 10^{-7} (6),(9)	Alarms
O-GH-RE-22 Radwaste building exhaust	Particulate (3)	10^{-12} to 10^{-7}	1 X 10^{-11}	Cs-137	1 X 10^{-8} (18)	1 X 10^{-7} (17)	12,000	50	4X10-3(5)	4 X 10^{-10} (6),(9)	Alarms
O-GK-RE-41 Access control area ventilation exhaust monitor	Particulate (3)	10^{-12} to 10^{-7}	1 X 10^{-11}	Cs-137	1 X 10^{-8} (18)	1 X 10^{-7} (17)	6,000	100	1.67x10^{-2}(5)	1 x10^{-7} (6),(9)	Alarms

TABLE 12.3-3 (Sheet 2)

Monitor	Type (continuous)	Range (μCi/cc)	MDC (1) (μCi/cc)	Controlling Isotope	Hi Alarm (μCi/cc)	Hi-Hi Alarm (μCi/cc)	Total Ventilation Flow (cfm)	Sumbompartment Flow Rate (cfm)	Dilution Factor	Minimum Required Sensitivity (μCi/cc)	Monitor Control Function
O-GH-RE-23 Waste gas decay tank area ventilation exhaust monitor	Gaseous (3)	10^{-7} to 10-2	2 X 10^{-7}	Kr-85 Xe-133	5 X 10^{-5} (8)	5 X 10^{-4} (7)	500	250	0.5 (5)	5 X 10^{-4} (6),(9)	Alarms
Portable monitor	Particulate (3) Iodine (4) Gaseous (3)	10^{-12} to 10^{-7} 10^{-11} to 10^{-6} 10^{-7} to 10^{-2}	1 X 10^{-11} 1 X 10^{-10} 2 X 10^{-7}	Cs-137 I-131 Kr-85	NA NA NA	NA NA NA	NA NA NA	NA NA NA	NA NA NA		Alarms

Sample Flow for each channel is 3 cfm.
(1) MDC = minimum detectable concentration.
(2) When fuel is in the building
(3) Beta scintillation detector.
(4) Gamma scintillation detector.

(5) Dilution factor = $\frac{\text{SUBCOMPARTMENTAL FLOW IN CFM}}{\text{Total flow in cfm}}$

(6) Minimum required sensitivity of monitor in Ci/cc at 10 DAC-hrs for the controlling isotope = dilution factor X 10 DAC.

(7) 10 DAC X dilution factor or monitor maximum, whichever is less.
(8) DAC X dilution factor or one tenth of Hi-Hi Alarmm, whichever is less.
(9) Grab samples are analyzed in the laboratory, and iodine concentrations are calculated, using previously established ratios.
(10) Hi alarm and Hi-Hi alarm setpoints are governed by the WCGS Offsite Dose Calculation Manual. The alarm points are variable dependent upon the isotope mixture. upon the isotope mixture. See WCGS Technical Specifications.
(11) Equivalent to 0.9 mR/hr submersion dose rate (may increase per Tech Spec Table 3.3-6)
(12) Equivalent to 9 mR/hr submersion dose rate (may increase per Tech Spec Table 3.3-6)
(13) Equivalent to 0.4 mR/hr submersion dose rate
(14) Equivalent to 4 mR/hr submersion dose rate
(15) Equivalent to 0.2 mR/hr submersion dose rate
(16) Equivalent to 2 mR/hr submersion dose rate
(17) DAC(Q) X dilution Factor where DAC(Q) is the Annual Concentration permitted. (2000 hours X the controlling Isotope DAC).
(18) $\frac{\text{DAC(Q)}}{10}$ X dilution Factor.

TABLE 12.3-4

POWER SUPPLIES FOR AREA AND IN-PLANT AIRBORNE MONITORS

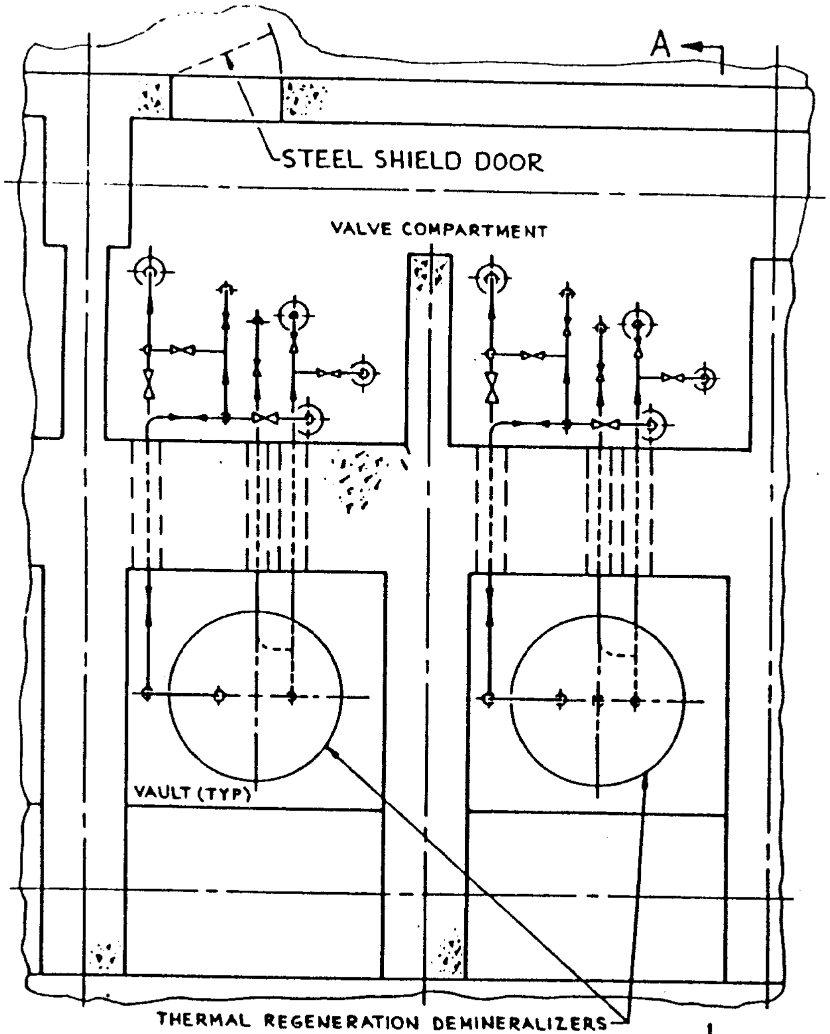
Area Radiation Monitors

Monitor Number	Normal Power Supply	Restored After Loss of Offsite Power	Remarks
0-SD-RE-1 0-SD-RE-2 0-SD-RE-3 0-SD-RE-4 0-SD-RE-5 0-SD-RE-6 0-SD-RE-7 0-SD-RE-8 0-SD-RE-9 0-SD-RE-10 0-SD-RE-11 0-SD-RE-12 0-SD-RE-13 0-SD-RE-14 0-SD-RE-15 0-SD-RE-16 0-SD-RE-17 0-SD-RE-18 0-SD-RE-19 0-SD-RE-20 0-SD-RE-21 0-SD-RE-22 0-SD-RE-23 0-SD-RE-24 0-SD-RE-25 0-SD-RE-26 0-SD-RE-27 0-SD-RE-28 0-SD-RE-29 0-SD-RE-30 0-SD-RE-31 0-SD-RE-32 0-SD-RE-33 0-SD-RE-34 0-SD-RE-35 0-SD-RE-36 0-SD-RE-37 0-SD-RE-38 0-SD-RE-39 0-SD-RE-40 0-SD-RE-41 0-SD-RE-42 0-SD-RE-47	Supplied by regulated instrumentation ac power which is supplied from the diesel generators on loss of offsite power.	Yes	
0-SD-RE-43	Supplied by TSC emergency generator on loss of offsite power.	Yes	
0-SD-RE-44	Supplied by EOF emergency generator on loss of offsite power	Yes	

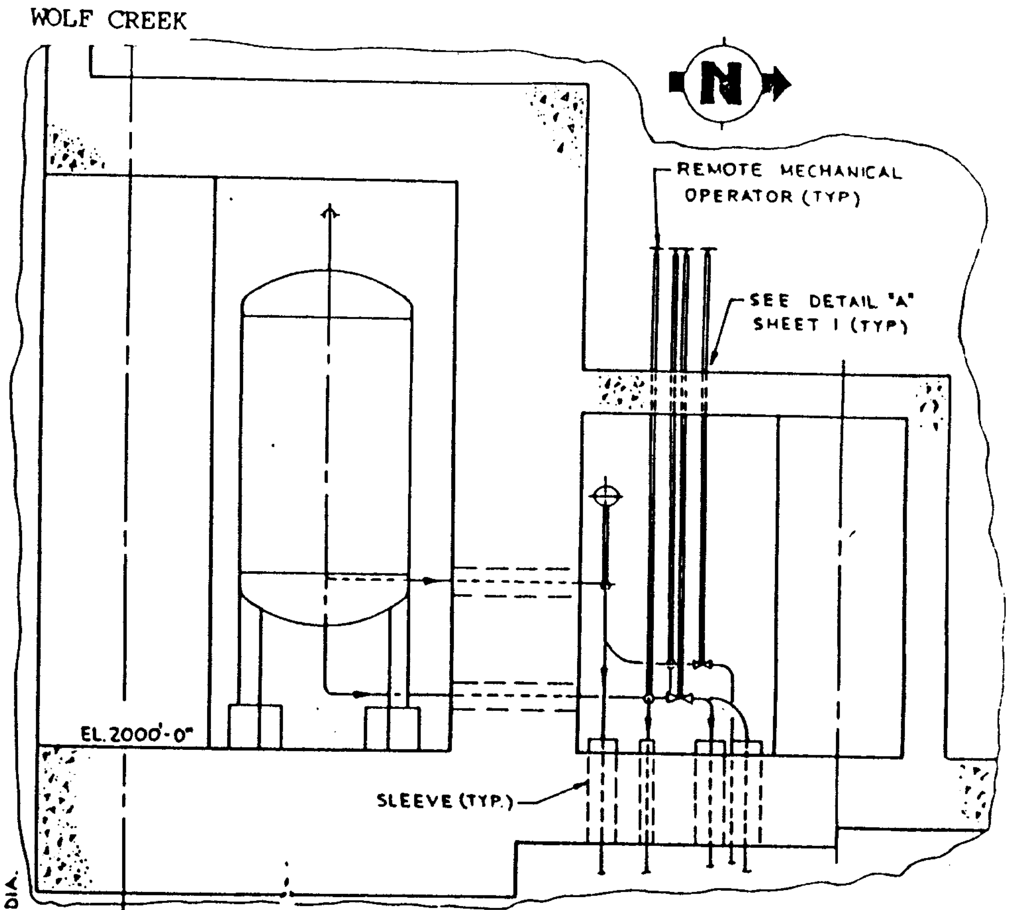
TABLE 12.3-4 (Sheet 2)

In-Plant Airborne Radioactivity Monitors (Class IE)

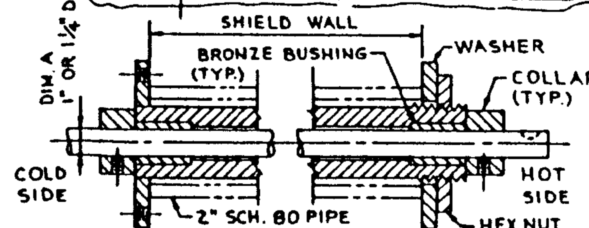
Monitor Number	Normal Power Supply	Restored After Loss of Offsite Power	Remarks
Containment atmosphere O-GT-RE-31 O-GT-RE-32	Class IE MCCs	Yes	
Containment purge system O-GT-RE-22 O-GT-RE-33	Class IE MCCs	Yes	
Fuel building exhaust O-GG-RE-27 O-GG-RE-28	Class IE MCCs	Yes	
Control room air supply O-GK-RE-04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



PLAN VIEW AUX. BLDG.
DEMINERALIZER AREA EL. 2000'-0"



SECTION "A-A"

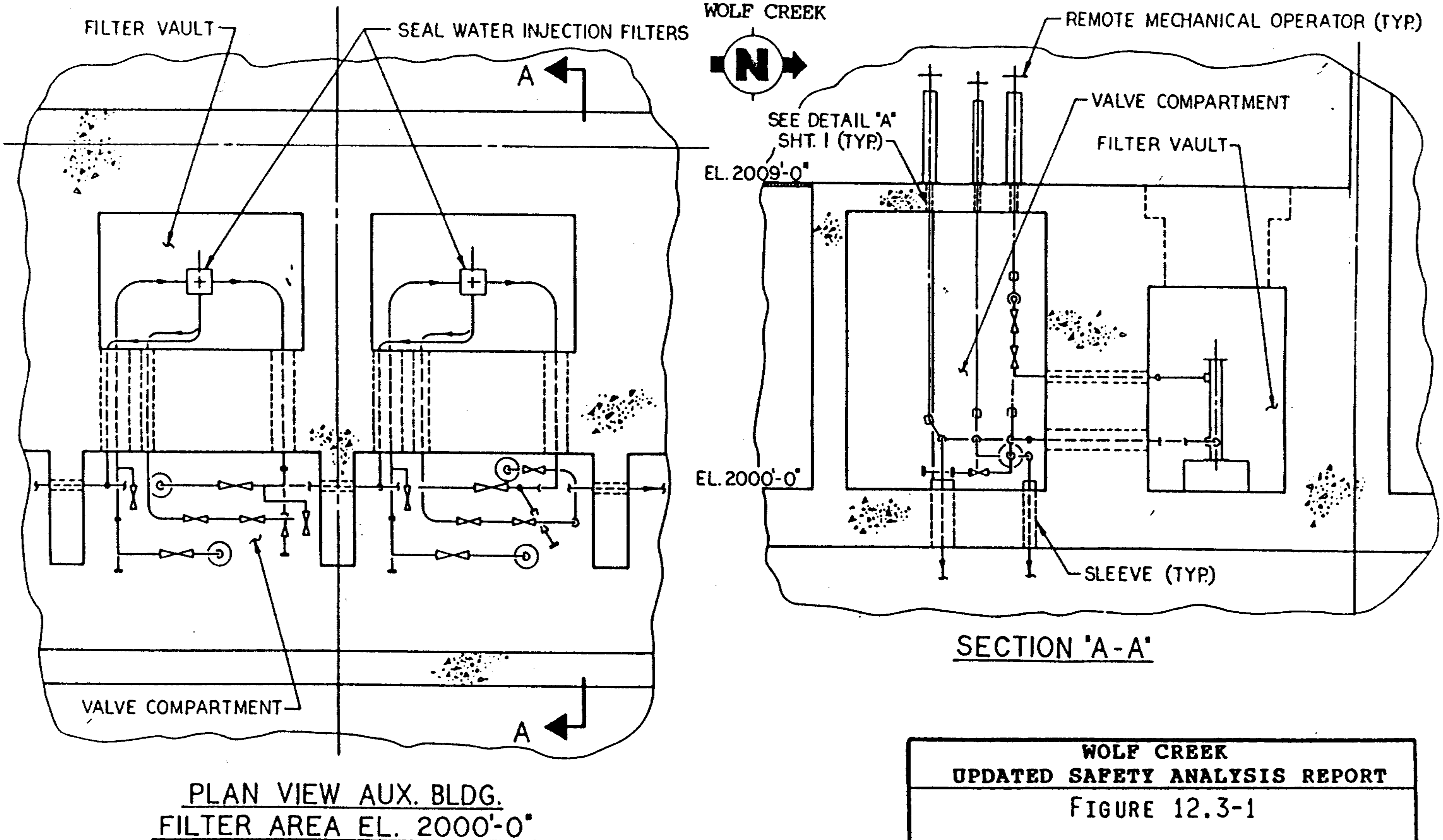


SKETCH "A"
TYPICAL SHAFT SLEEVE FOR RADIOACTIVE SERVICE

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.3-1

TYPICAL VALVE COMPARTMENT ARRANGEMENT (SHEET 1)



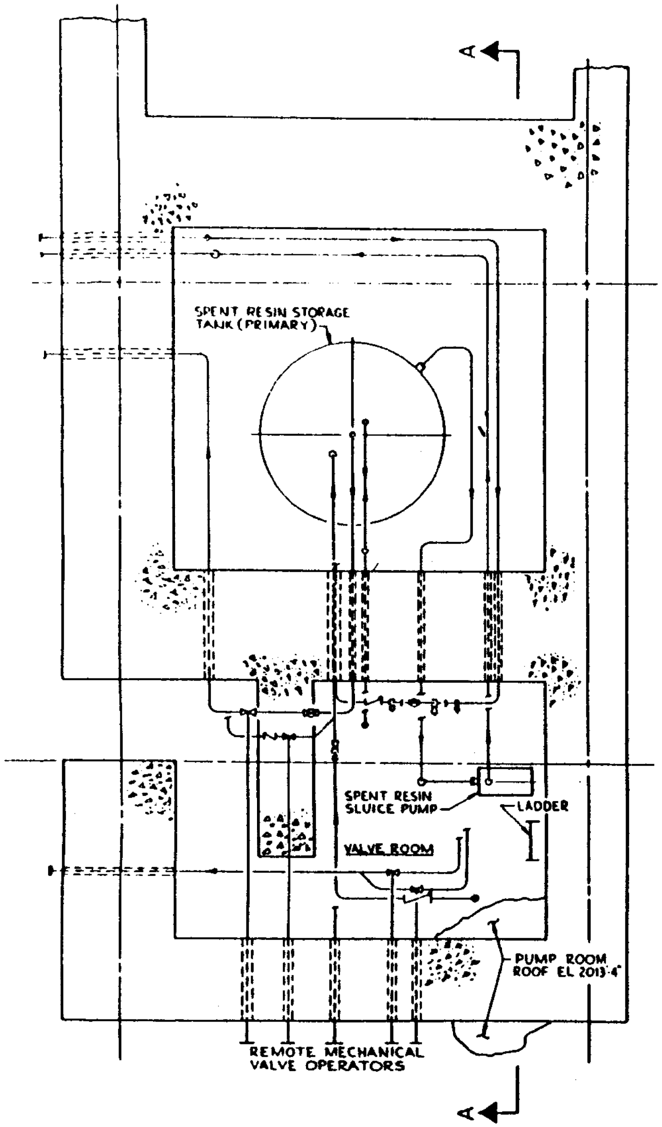
Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

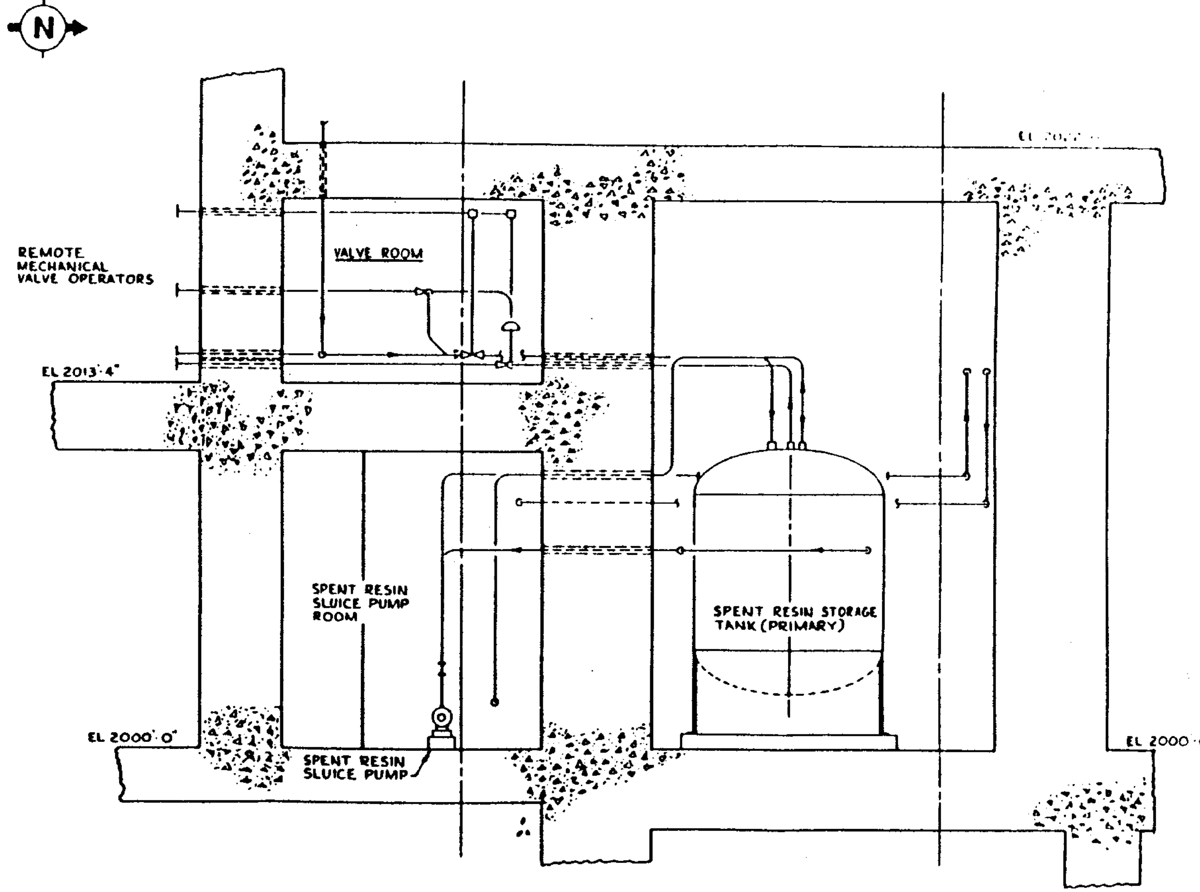
FIGURE 12.3-1

TYPICAL VALVE COMPARTMENT ARRANGEMENT
(SHEET 2)

WOLF CREEK



PLAN VIEW RADWASTE BLDG.
SPENT RESIN STORAGE AREA EL. 2000'-0"
(PRIMARY)



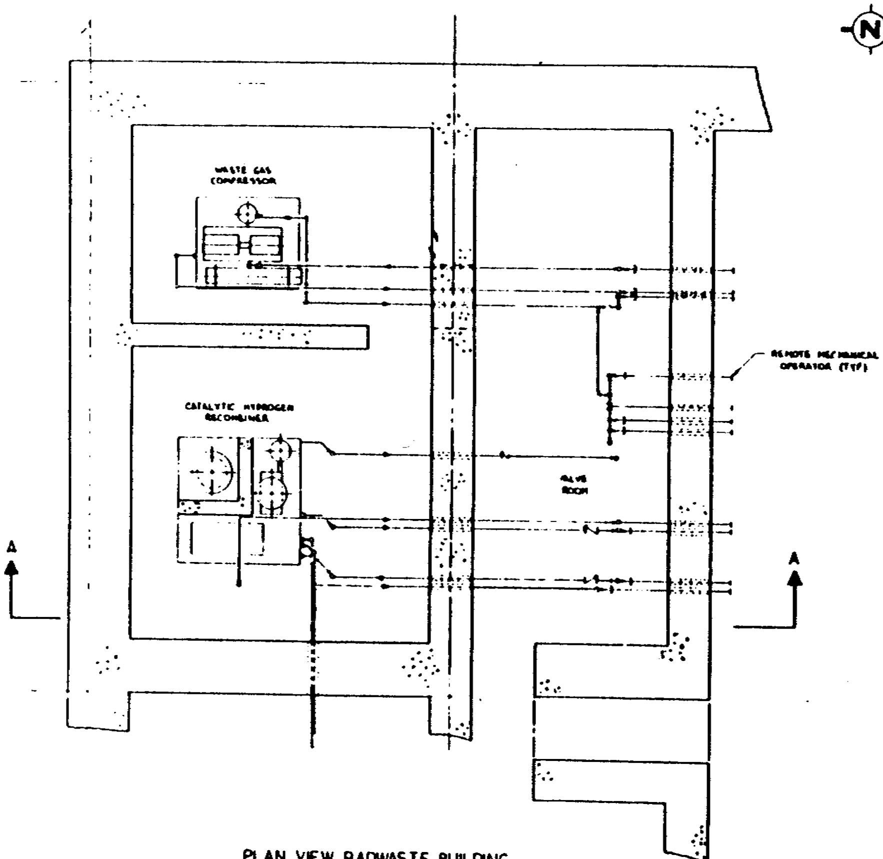
SECTION A-A

Rev. 0

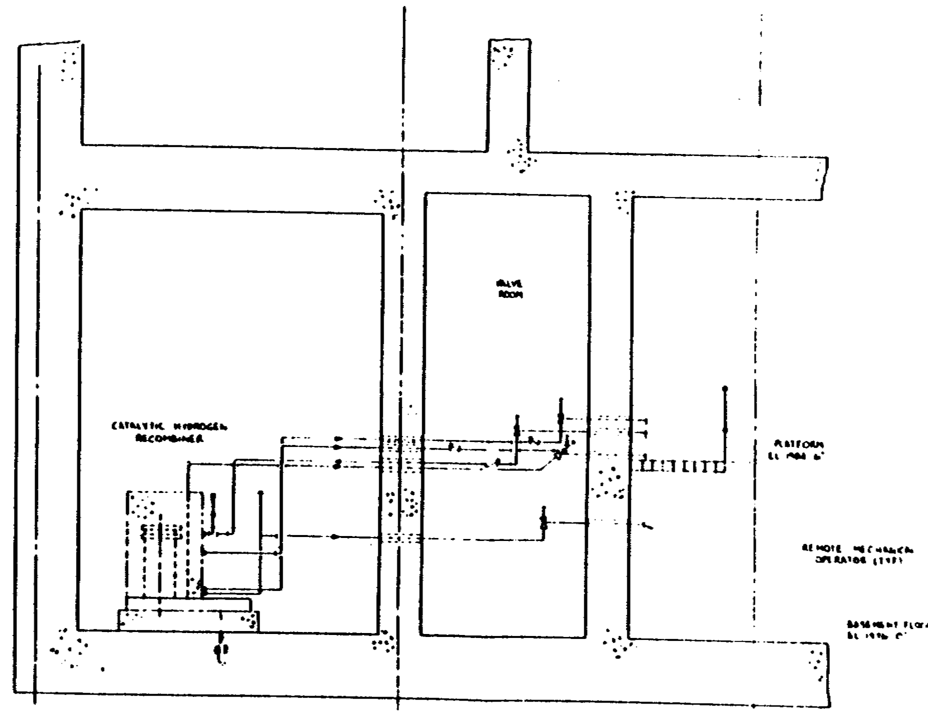
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.3-1

TYPICAL VALVE COMPARTMENT ARRANGEMENT
(SHEET 3)



PLAN VIEW RADWASTE BUILDING
WASTE GAS COMP & HYDROGEN RECOMBINER AREA EL 1976'-0"



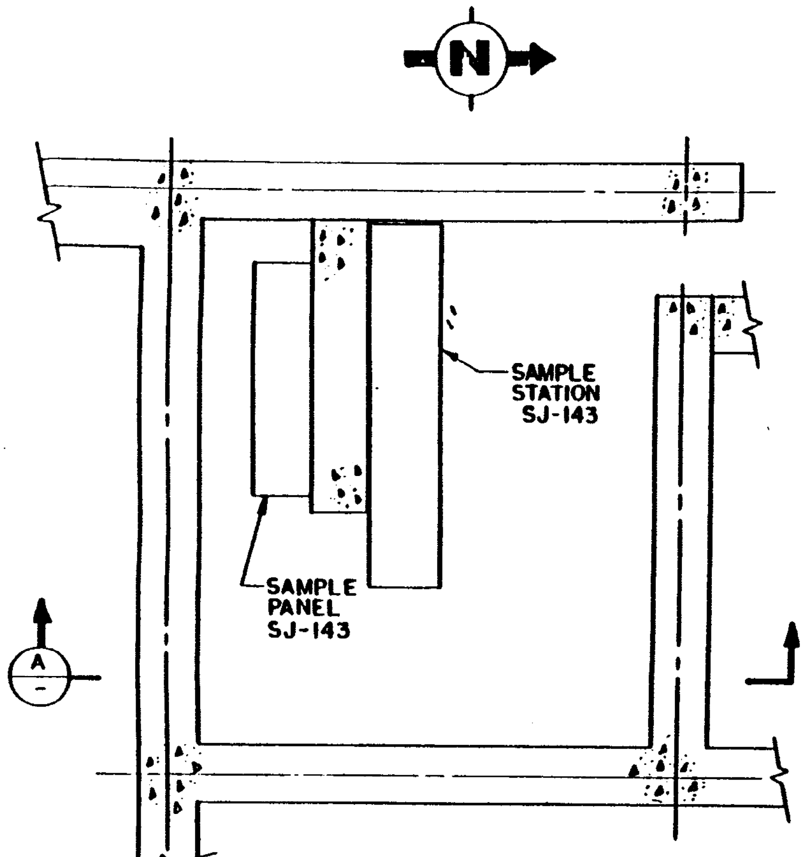
SECTION A-A

Rev. 0

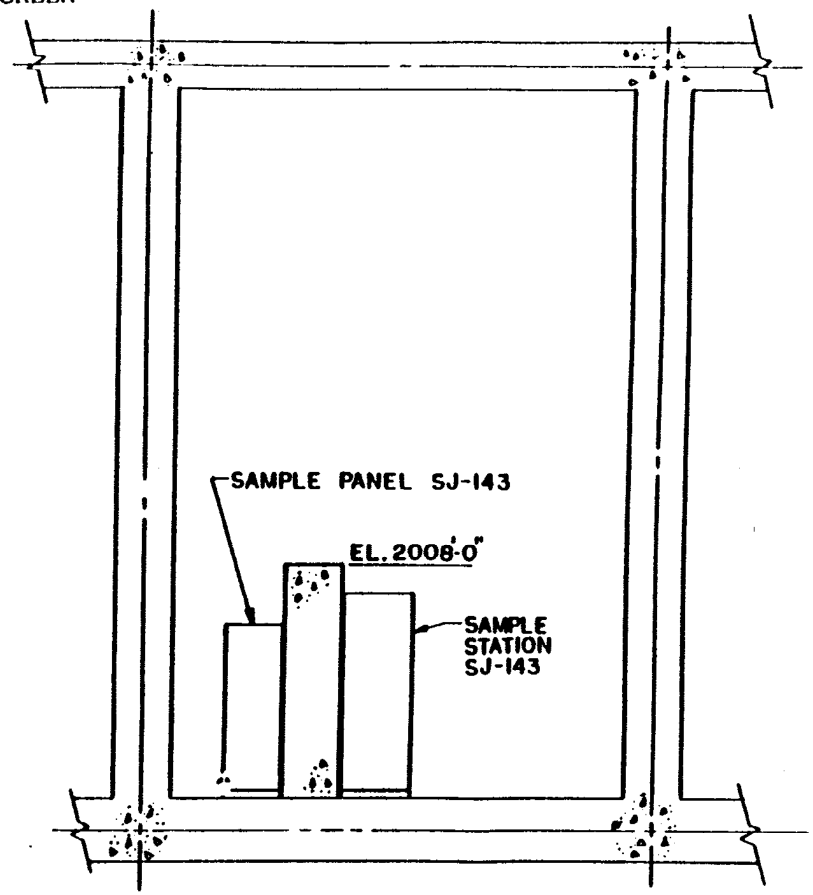
WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.3-1

TYPICAL VALVE COMPARTMENTTY
ARRANGEMENT
(SHEET 4)



PLAN VIEW-AUX. BUILDING
SAMPLING ROOM EL. 2000'-0



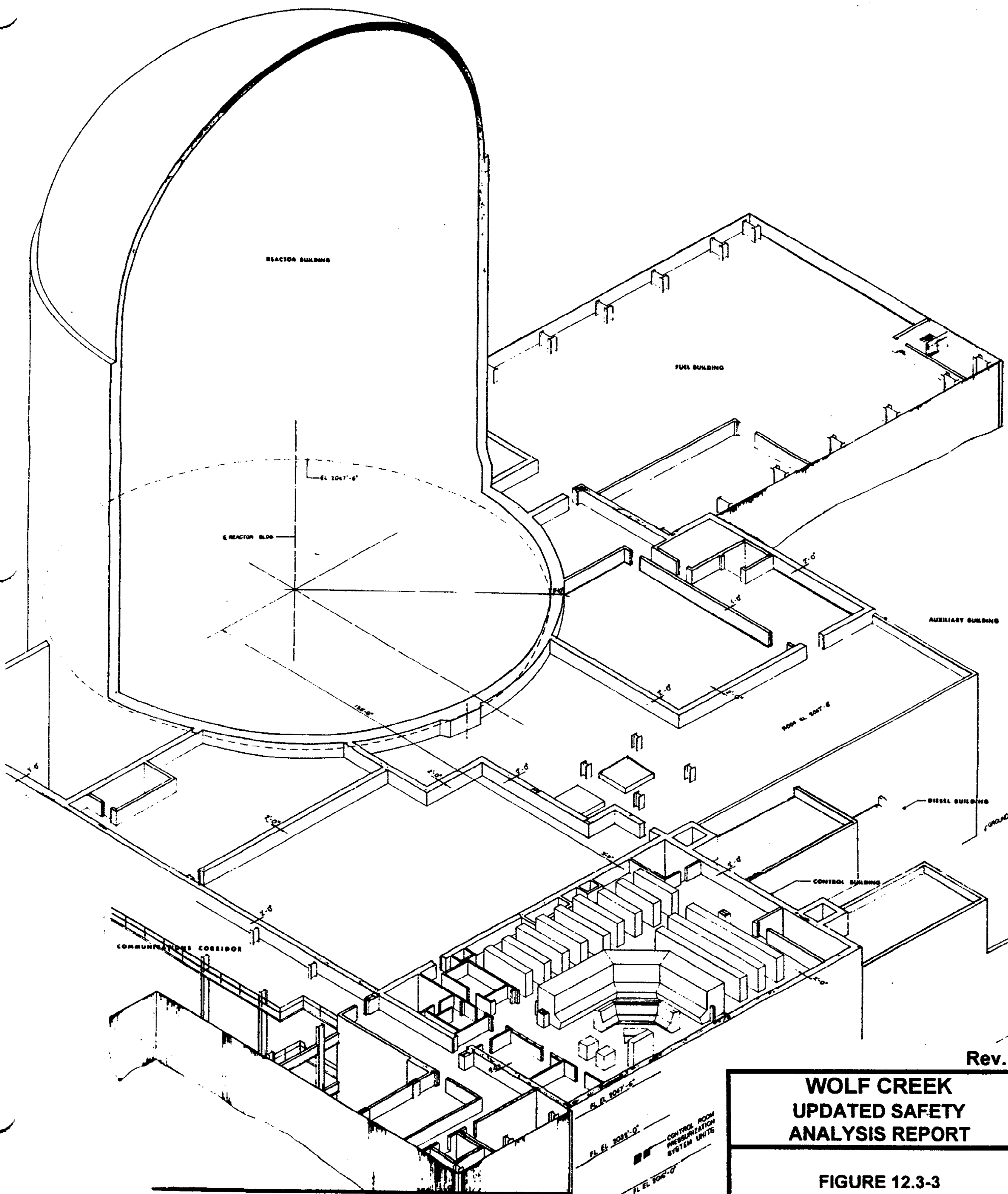
SECTION A

Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.3-1

TYPICAL VALVE COMPARTMENT ARRANGEMENT (SHEET 5)



Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.3-3

CONTROL ROOM ISOMETRIC

12.4 DOSE ASSESSMENT

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

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

12.4.1 EXPOSURES WITHIN THE PLANT

12.4.1.1 Direct Radiation Dose Estimates

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

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

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

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

d. Anticipated manhour occupancy requirements for WCGS are given in Table 12.4-12.

e. Table 12.4-13 provides an estimate of the distribution of the annual man-rem according to work function.

The precision of the man-rem estimate is of secondary importance. That estimate's relationship to actual man-rem doses received during subsequent plant operation will depend primarily on operating experience and maintenance and repair problems encountered rather than on design projections, however precise.The maximum and expected average dose rates in the plant radiation areas are given below:

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

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

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

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

Assuming a work year of 2080 hours, the total estimated annual occupancy time for an individual in different work functions in various radiation zones is as follows:

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

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

c. Inservice inspection: The total occupancy time for an individual involved in this work function in various radiation zones is expected to be 320 hrs/yr (Table 12.4-12) at the rate of 40 hrs/wk for 8 wks/yr. The annual average dose for an individual involved in this work function is approximately 0.5 rem.

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

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

f. Radwaste processing: The radwaste operations are performed by personnel involved in work functions such as routine operation and surveillance, health physics, chemistry, and routine maintenance. The total times expended by personnel in various radiation levels for the category are given in Table 12.4-1. The expected occupancy time for an individual involved in radwaste processing is about 2080 hrs/yr. The annual average dose per individual in this category is estimated to be about 3 rem.

12.4.1.2 Airborne Radioactivity Dose Estimates

12.4.1.2.1 Exposures Due to Airborne Radioactivity

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

12.4.1.2.2 Model for Calculating Exposures Due to Airborne Radioactive Sources

Thyroid and inhalation doses are calculated using the following equation:

$$D_O = \sum_i C_I \, (BR).t.DF_{OI}$$

where

C_I = Airborne concentration of the ith nuclide in pci/cm^3

and

BR = Breathing rate for occupational worker in cm^3/sec = 347

t = Time duration of inhalation of radioactivity contaminated air in seconds

DF_{OI} = Dose factor for adult for organ 0 (thyroid

or lung) via inhalation in mrem/pCi inhaled for the

i^{th} isotope (These dose factors are taken from Regulatory Guide 1.109, Rev. 1)

D_O = Dose in millirems to organ 0 due to inhalation

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

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

$$D_O = (DR)_O \; 10^{-3}.h$$

where

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

and

h = Annual occupancy in man-hours/yr
D_0 = Annual exposure in man-rem/yr

12.4.1.3 Illustrative Examples of Dose Assessment

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

a. Operation and Surveillance

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

b. Routine Maintenance

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

c. Radwaste Processing

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

d. Refueling

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

e. Inservice Inspection

Inservice inspections at operating PWRs have taken much of the refueling outage, requiring from 3 to 38 men depending on the inspections scheduled, at up to 2500 mrem/day. Total exposures have ranged from 10 to 24 man-rem. Steam generator eddy current testing, as required by Regulatory Guide 1.83, is perhaps the largest single cause of exposure for ISI. Table 12.4-1 provides a listing of various ISI functions and their associated man-rem from operating plant data.

f. Special Maintenance

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

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

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

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

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

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

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

12.4.2 EXPOSURES AT LOCATIONS OUTSIDE PLANT STRUCTURES

12.4.2.1 Direct Radiation Dose Estimates

Direct radiation outside plant structures from the containment and the auxiliary, radwaste, and turbine buildings is negligible, compared to that from outside storage tanks. The principal sources of radioactivity not stored in the plant structures are the reactor makeup water storage tank, the refueling water storage tank, and the condensate storage tank. These tanks, shall be limited to the conditions outlined in Section 16.11.1.1. The dose rate at the nearest site boundary for the sectors in which these tanks are exposed has been calculated to be of the order of 10-5 mrem/yr. Using the projected populations for the year 2020 in the exposed sectors to a distance of 50 miles throughout the year results in a negligible population exposure of less than 10-3 man-rem.

12.4.2.2 Exposures Due to Airborne Radioactivity

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

12.4.3 REFERENCES

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

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

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

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

TABLE 12.4-1

ILLUSTRATIVE EXAMPLES OF DOSE ASSESSMENT

Operation and Surveillance:

Description of Task	No. of Days [(1)]	No. of Personnel [(2)]	Mrem/man-day	No. of man-rem
Fuel bldg.	Refueling	29	10-113	6
Aux. bldg.	Entire yr	5-9	9-186	2
Aux. bldg. equip. test	18	3	7-65	0.2
Ctmt. (initial survey after SD)	1-3	5-17		0.1-0.5
Total for oper. & surv.				13-30

Routine Maintenance

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

TABLE 12.4-1 (Sheet 2)

Description of Task	No. of Days(1)	No. of Personnel(2)	Mrem/man-day	No. of man-rem
Reactor coolant pump motor work	35	18-30		6
Uncouple/couple reactor coolant pumps	22	15-19		0.8
Install/remove reactor coolant pump scaffolding	12	7-10		0.3
Repair ctmt. sump pumps & level indicators	9-75	9-30		0.4-4
Repair pressurizer relief valves	5-31	4-9		0.2-0.3
Replace excore detectors	23-42	15-30		1-8
Polar crane maintenance	9-78	9-22		0.3-3
Ctmt. pressure test & valve repair	60	26-32		2
Remove & clean SI system strainers	3	14-25		0.6
Check & repair snubbers	3-43	6-21		0.1-2
Remove equip. hatch	5-10	2-9		0.002-0.1
Replace equip. hatch	2-3	3-7		0.03-0.04
Radwaste Processing				
Waste drum loading	Refueling	7	62-280	6
Loading spent resin casks	Refueling	10	78-760	4
Dry waste drum handling	Refueling	3-59		0.2-4
Fuel storage pool filter replacement	2	4		0 6
Transferring resin	1	4-5		0.1
Refueling				
Install and remove reactor cavity seal ring	7-16	7-30		0.4-0.9
Remove transfer tube flange	12-17	2-10		0.3-0.4

TABLE 12.4-1 (Sheet 3)

Description of Task	No. of Days[1]	No. of Personnel[2]	Mrem/man-day	No. of man-rem
Install & remove upper guide structure lifting rig	3-31	6-25		0.2-3
Rig underwater lights	2-18	4-13		0.02-0.3
Flood refueling canal	4	5-9		0.3
Fuel handling (ctmt.)	14-16	49-92		3-7
Fuel handling (fuel bldg.)	5-28	25-77		0.5-1
Incore instrumentation removal	3-36	13-24		2-6
Remove & replace missile shield	6-12	9-24		0.2-0.3
Remove & replace cable & ductwork to reactor head	11-13	9-27		0.9-3
Remove tool access flanges & uncouple control rods	10-13	9-39		0.8-9
Install bullet noses	1-9	6-15		0.7-2
Remove bullet noses	4-5	2-4		0.1-0.4
Remove & replace head insulation	7-10	4-5		0.3-0.4
RPV studs - remove, replace, & clean	10-19	26-40		3-13
Install & remove alignment pins	2-11	3-12		0.1-1
Install lift rig & remove RPV head	2-10	11-37		0.6-3
Replace RPV head	1-3	6-18		0.4-7
Decon. RPV head & vessel flanges & replace "0" rings	7-14	10-24		3-5
Remove & replace holddown ring	4-6	14-31		3-8
Clean stud plugs & holes and install & remove plugs	4-11	12-26		1-6
Clean incore instrument flanges, studs, & nuts	9-28	12-32		2-6
RPV head gasket replacement	6-32	19-35		1-4

TABLE 12.4-1 (Sheet 4)

Description of Task	No. of Days(1)	No. of Personnel(2)	Mrem/man-day	No. of man-rem
Drain, fill, and vent reactor coolant system	2-9	5-10		0.1-0.7
Install handrails around refueling canal	14	6-8		0.1
Total for refueling	15-30	8-61	32-347	13-66
Inservice Inspection				
Steam generator tubes	7-51	13-38	29-2500	4-22
Spent fuel sipping	20	17	14-242	3
Spent fuel inspection	57	22-30		1-2
Remove & replace primary manway covers	4-14	12-37		3-7
Remove & replace secondary man- way covers	7-8	4-16		0.4-0.8
Steam generator secondary side	2-4	11-17		0.2-2
Install primary loop dams	2	3		1
Clean steam generator manway studs	6	4-12		0.6
Total for ISI				10-24
Special Maintenance				
Replace upper guide structure on control rods	69	78	17-785	13
Sludge lancing of steam generator secondary side	5	16	19-105	2
Steam generator tube plugging	7-30	5-10		20-200

TABLE 12.4-1 (Sheet 5)

NOTES:

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

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

TABLE 12.4-2

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

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

NOTES:

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

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

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

TABLE 12.4-3

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

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

NOTES:

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

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

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

TABLE 12.4-4

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

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

NOTES:

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

TABLE 12.4-4 (Sheet 2)

2. Multiunit plants have been excluded for the most part unless commercial operation dates are close together (example, Surry 1 & 2), or unless the additional units are not scheduled for commercial operation for approximately 2 years or more. (Example, Arkansas 1 & 2, Three Mile Island 1 & 2, St. Lucie, Davis-Besse 1, 2, & 3, Farley 1 & 2, and Salem 1 & 2).

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

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

TABLE 12.4-5

DISTRIBUTION OF THE NUMBER OF PERSONNEL
(>100 MILLIREM/YR)
ACCORDING TO WORK FUNCTION

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

See notes at end of Table 12.4-7.

TABLE 12.4-6

DISTRIBUTION OF PERSONNEL (>100 MILLIREM/YR) ACCORDING TO EMPLOYEE CATEGORY

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

See notes at end of Table 12.4-7.

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

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

Notes for Tables 12.4-5, 6 and 7:

1. PWRs and BWRs operating at all power levels were considered in compiling these tables.

2. Percentages for 1974 and 1975 are based on approximately 39 percent and 50 percent of the total exposures reported in the appropriate year for light water reactors which had been in commercial operation for at least one full year, as of 12/31/74 and 12/31/75, respectively.

3. Percentages for 1976 and 1977 are on only those facilities which have been in commercial operation for at least one full year, as of 12/31/76 and 12/31/77, respectively.

4. Distributions of personnel receiving an annual exposure of greater than 100 millirems, according to either work function or employee category, are not available for 1976 or 1977.

TABLE 12.4-8

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

	Westinghouse		C-E		B & W		
Approx. Yr. of Operation	No. of Reactors	Man-Rem Yr-Unit	No. of Reactors	Man-Rem Yr-Unit	No. of Reactors	Man-Rem Yr-Unit	Comments
1	7	149	5	117	2	47	
2	8	243	4	522	3	211	CE: Palisades - 1,131 $\frac{\text{man-rem}}{\text{yr}}$
3	7	608	3	419	3	335	W: Ginna - 1,032 $\frac{\text{man-rem}}{\text{yr}}$ CE: Palisades produces very little power.
4	6	741	3	229	-	-	W: Surry 1 & 2 - 3,165 $\frac{\text{man-rem}}{\text{yr}}$
5	6	876	2	471	-	-	W: Ginna 1 - 1,224 $\frac{\text{man-rem}}{\text{yr}}$ Robinson 2 - 1,142 $\frac{\text{man-rem}}{\text{yr}}$ Surry 1 & 2 - 2,307 $\frac{\text{man-rem}}{\text{yr}}$
6	4	553	1	100	-	-	-----------
7	4	341	-	-	-	-	-----------
8	3	443	-	-	-	-	-----------
9	2	665	-	-	-	-	W: San Onofre - 800 $\frac{\text{man-rem}}{\text{yr}}$
10	2	745					

Notes - On Table 12.4-8:

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

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

TABLE 12.4-9

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

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

TABLE 12.4-9 (Sheet 2)

Maine Yankee	5	790	68	237	Percentages of full power produced were low in 1973 (52 percent) and 1974 (55 percent)
Fort Calhoun 1	4	457	63	244	1975- Only 55 percent of full power was produced.

Notes on Table 12.4-9:

1. Only single unit PWRs have been considered.
2. Full power data based on information given in "Commercial Nuclear Power Plants" by NUS Corp., 1978, Ed. 10.
3. Robinson full power is based on a summertime value.
4. Only plants which have been in commercial operation for equal to or greater than 4 years were considered.

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

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

Notes on Table 12.4-10:

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

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

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

TABLE 12.4-11

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

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

NOTE:

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

TABLE 12.4-12

ESTIMATES OF OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA DOSES TO PLANT PERSONNEL

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

TABLE 12.4-12 (Sheet 2)

Operation	Zone	Percentage of Occupancy	Hrs/yr	Hourly Dose Rate (Rems/Hr)	Yearly Dose Rate (Rems/Yr)	No. of men	Annual Exposures (Man-Rem/yr-Unit)
Refueling	A	30	48	1×10^{-4}	0.004	6	0.22
	B	40	64	5×10^{-4}	0.03	6	1.47
	C	24	39	2.5×10^{-3}	0.09	6	4.48
	D	4	6	1.5×10^{-2}	0.09	6	4.14
	E	2	3	1×10^{-1}	0.30	6	13.8
Total		100	160		0.51		24.1
Inservice Inspection	A	55	176	1×10^{-4}	0.02	46	0.81
(ISI)	B	36	115	5×10^{-4}	0.06	46	2.64
	C	7	23	2.5×10^{-3}	0.06	46	2.61
	D	1	3.5	1.5×10^{-2}	0.05	46	2.30
	E	1	3	2×10^{-1}	0.67	46	30.67
Total		100	320		0.86		39.03
Special Maintenance	A	75	240	1×10^{-4}	0.024	200	4.8
	B	20	64	5×10^{-4}	0.03	200	6.4
	C	3	10	2.5×10^{-3}	0.025	200	5
	D	1	3	1.5×10^{-2}	0.04	200	9
	E	1	3	2×10^{-1}	0.6	200	120
Total		100	320		0.71		145

TABLE 12.4-13

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

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

TABLE 12.4-14

ANNUAL OCCUPANCY IN PLANT AREAS CONTAINING AIRBORNE RADIOACTIVITY

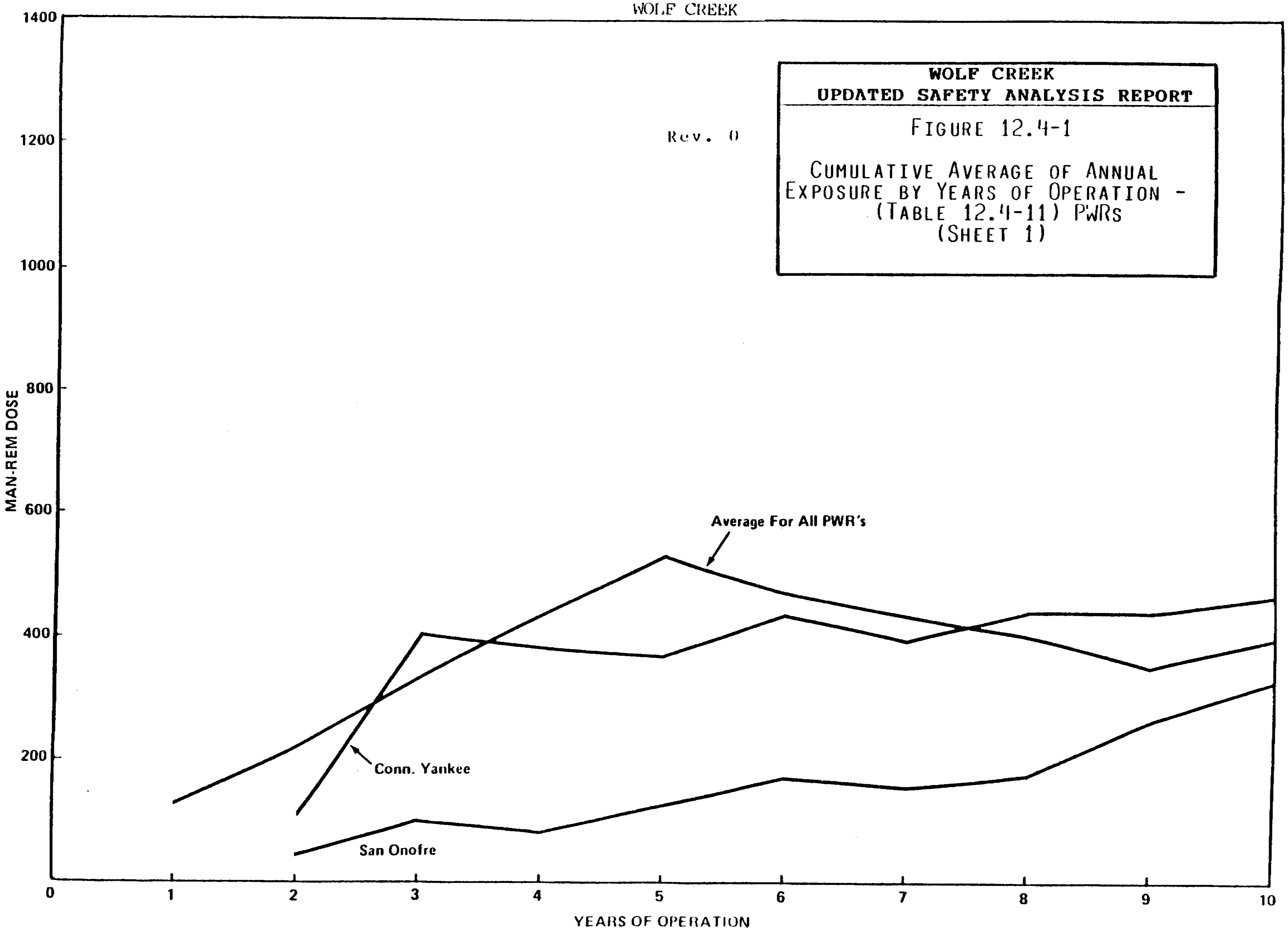
Building	hr/yr per man	No. men	man-hours yr
Containment -power	250	3	750
Containment -refueling	62.5	32	2,000
Fuel building -power	250	2	500
Fuel building -refueling	125	14	1,750

TABLE 12.4-15

DOSES TO PLANT PERSONNEL CAUSED BY AIRBORNE RADIOACTIVITY

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

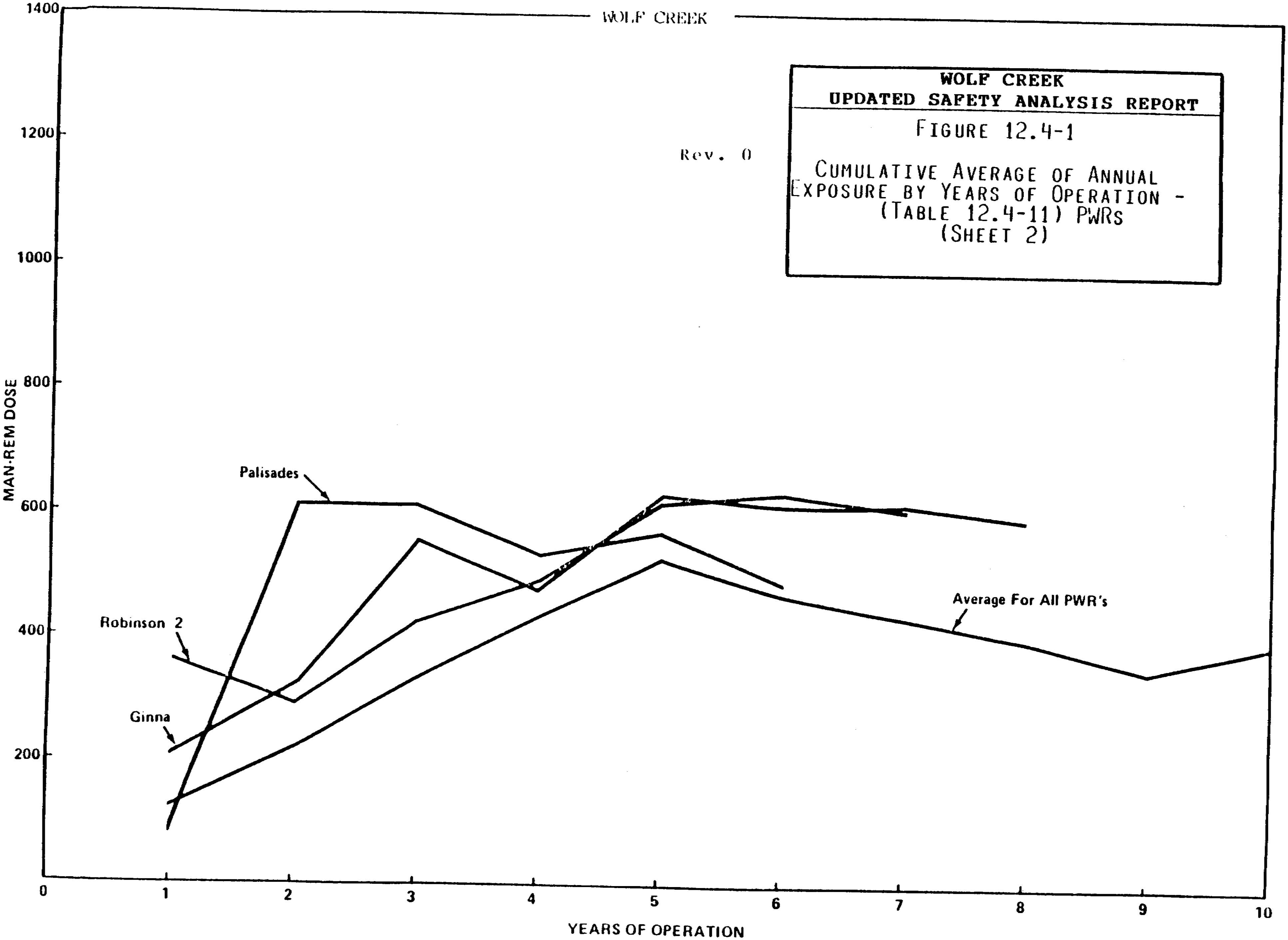
N = Negligible



WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.4-1

CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION - (TABLE 12.4-11) PWRs
(SHEET 1)

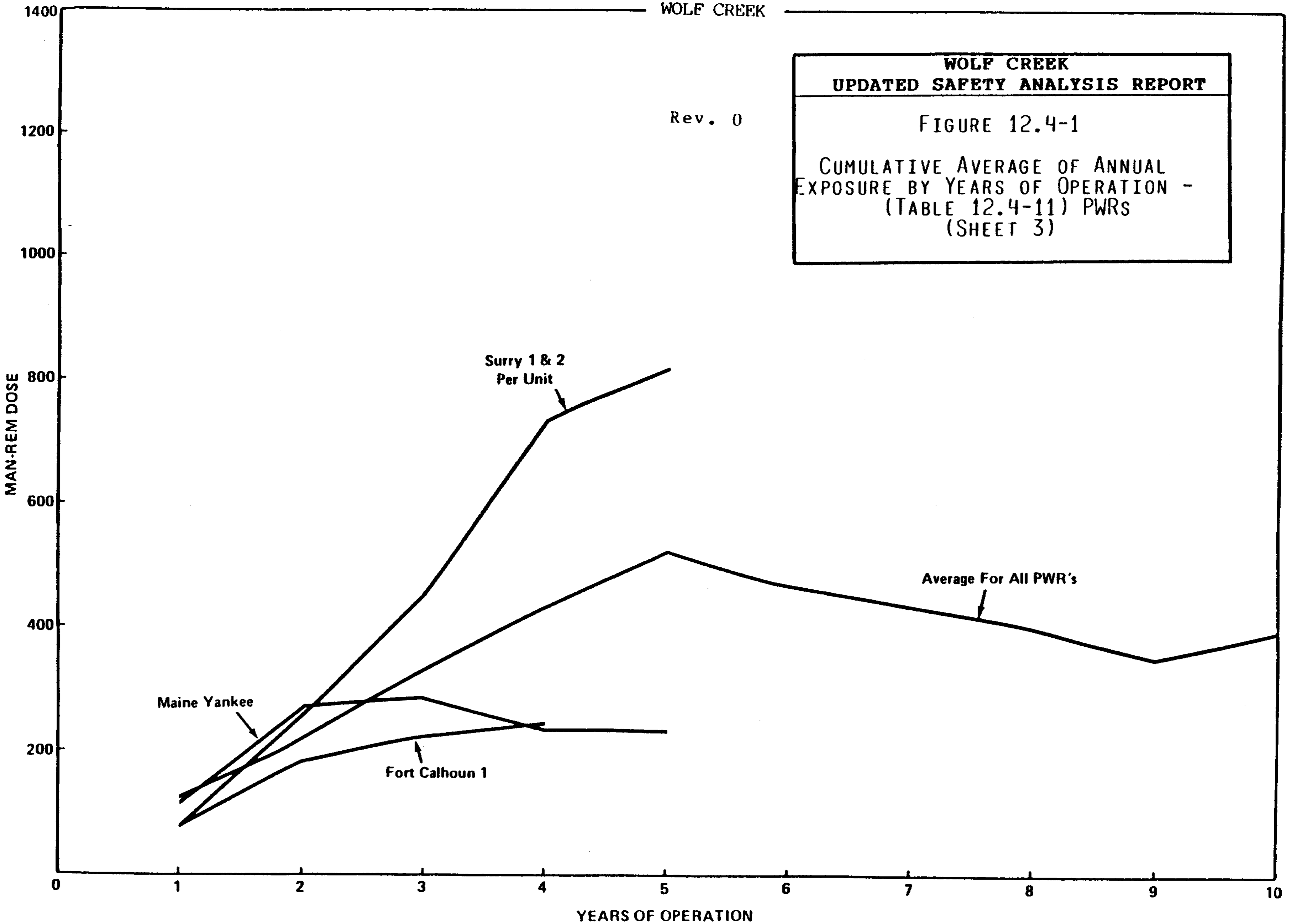


Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.4-1

CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION - (TABLE 12.4-11) PWRs
(SHEET 2)

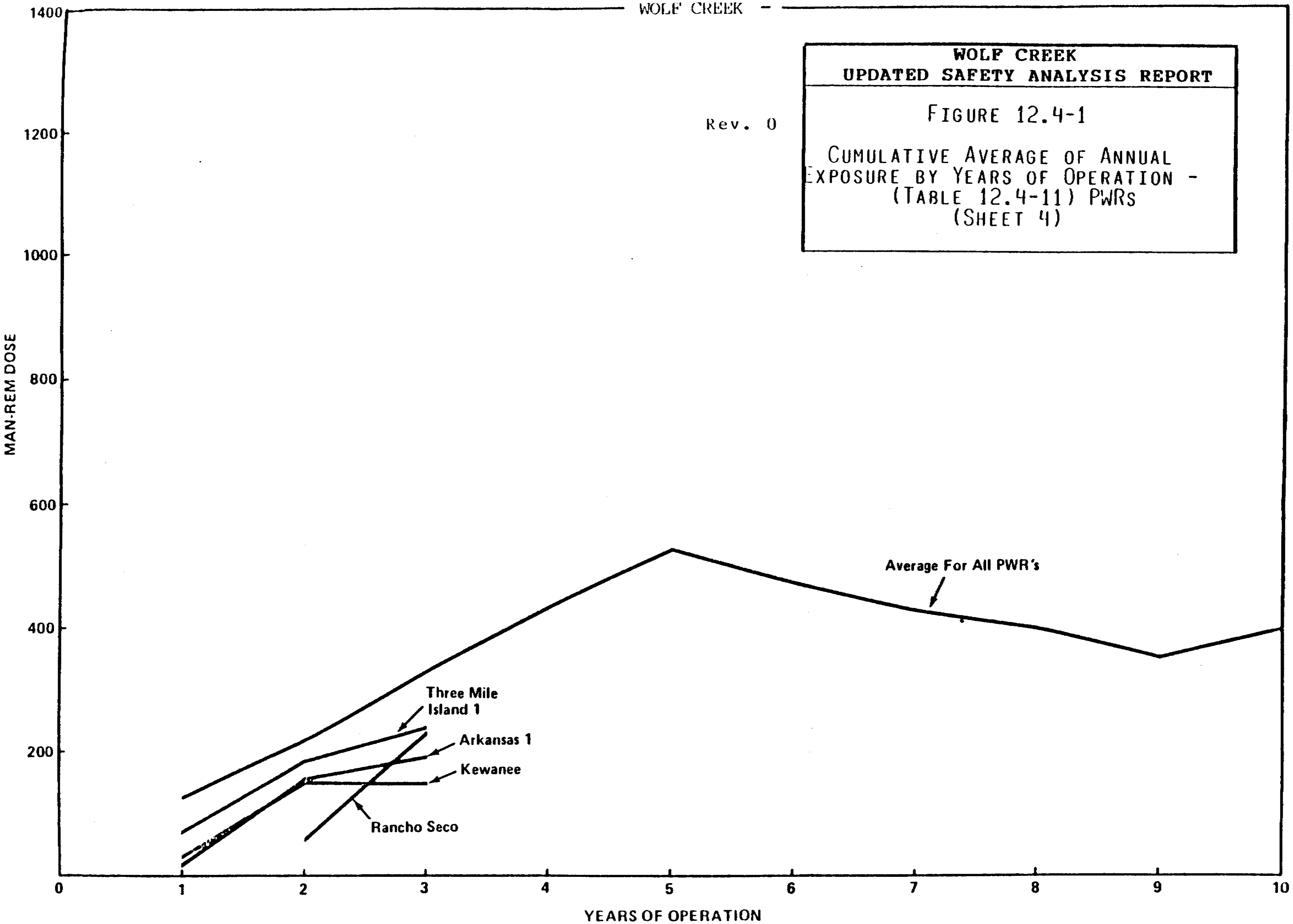


Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.4-1

CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION - (TABLE 12.4-11) PWRS (SHEET 3)



Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.4-1

CUMULATIVE AVERAGE OF ANNUAL EXPOSURE BY YEARS OF OPERATION - (TABLE 12.4-11) PWRs (SHEET 4)

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

The WCGS Health Physics Program is established to provide an effective means of radiation protection for station personnel, visitors and the general public. The program consists of: A management philosophy which supports radiation protection and ALARA concepts, a site organizational structure, (See Chapter 13.0), qualified personnel to direct and implement the Health Physics Program, written procedures outlining acceptable radiation protection practices and the appropriate equipment and facilities necessary to support a comprehensive Health Physics effort. The program is developed and implemented through the applicable sections of the Code of Federal Regulations, Regulatory Guides and ANSI standards. (See Table 12.1-1 for the applicable criteria).

The Health Physics Section is responsible for providing technical support to the WCGS Health Physics Program. This section is headed by the Superintendent Chemistry/Radiation Protection who in turn reports directly to the Manager Chemistry/Radiation Protection. At the time of commercial operation at least one individual within the section met the qualifications of ANSI N18.1-1971, Regulatory Guide 1.8, 8.2, 8.8 and 8.10.

The resume of the Manager Chemistry/Radiation Protection is provided in Section 13.1. The qualifications of the Superintendent Chemistry/Radiation Protection, who fulfills the requirements as the Radiation Protection Manager, are also included in Chapter 13.1.

The major responsibilities of this section include:

1) Providing technical support to WCGS in the area of Health Physics, radwaste, and emergency programs during normal plant operation, outages and unanticipated events.

2) Assisting in long range planning for WCGS based on new and proposed regulatory changes.

3) Performance of ALARA design reviews/cost benefit analysis, exposure data trending, ALARA committee activities and technical reviews.

4) coordinating with all other station departments to provide health physics coverage for all activities that involve radiation or radioactive materials.

5) assuring that radiation protection related training provided to personnel is technically accurate and properly describes Station Health Physics practices.

6) enacting the site ALARA program.

7) providing a personnel radiation monitoring and health physics records management program.

8) providing radiation surveys and appropriate posting of station areas, posting of radiation zones, supplying follow up monitoring and maintaining records.

9) maintaining Health Physics equipment functional and calibrated.

10) providing equipment and a respiratory protection program for the station personnel.

11) assuring proper shipment and receipt of radioactive material.

Procedures are developed with special attention given to maintaining personnel exposures to ALARA levels. Work in a radiation area for activities such as maintenance, inspection, refueling, and nonroutine operations is appropriately planned prior to the initiation of work so as to minimize exposures to as low as practicable levels. Where circumstances allow, specific exposure reduction techniques may be utilized as part of the procedures for work in radiation areas. Examples of these techniques include:

a. Minimizing source strength and contamination levels by flushing equipment prior to performing maintenance.

b. Minimizing radiation levels in work areas by the use of temporary shielding.

c. Using remote handling equipment and other special tools.

d. Minimizing discomfort of workers so that efficiency is increased and less time is spent in radiation areas.

Each individual is responsible for obeying the applicable site radiation protection procedures. A thorough, demonstrable knowledge of these is required for unescorted individuals prior to entry in the RCA. Individuals are further charged with following the guidance of the Health Physics staff during their indoctrination, and normal work tasks and any activity noted to be in violation of procedures is reported to the appropriate supervisor for resolution.

12.5.2 EQUIPMENT, INSTRUMENTATION AND FACILITIES

12.5.2.1 Health Physics Equipment and Instrumentation

Laboratory, fixed and portable equipment, and instrumentation are selected on the basis of: job requirements, usefulness, and characteristics such as sensitivity, response time, type of radiation detected, accuracy, dependability, and lifetime. Factors considered before selection include previous experience, recommendations and comments from other utilities, and guidelines in applicable ANSI Standards and Regulatory Guides. Laboratory equipment for Health Physics is stored in the Health Physics offices and labs, the Hot Laboratory and the Counting Room. Details on laboratory counting equipment are given in Table 12.5-1. Some equipment and instrumentation in these areas is shared jointly by Health Physics and Radiochemistry and includes multi-channel analyzers, scintillation counters, fume hoods, and other ancillary equipment and supplies needed to perform adequate Health Physics and Radiochemistry sample preparation and analysis.

A list of portable instrumentation with respect to instrument type, location, type of detector, range, accuracy and quantity is given in Table 12.5-2.

Calibration of Health Physics portable equipment and instrumentation is performed by qualified trained personnel using written procedures and using standards that are traceable to the National Institute of Standards and Technology. The frequency of calibration is established in station procedures.

A record of the calibration is produced for each calibration and is kept on file at the station. Instruments that are calibrated have a current calibration sticker attached and are stored separately from instruments that are out of service.

Portable instrumentation for routine use is stored in the RCA and at various locations throughout the Site in emergency kits, lockers, etc., as designated by Health Physics to meet survey instrumentation requirements.

The Respiratory Protection Program is the responsibility of the Manager Chemistry/Radiation Protection and complies with 10 CFR 20 Subpart H.

Selection and use of respiratory equipment, i.e., self contained breathing apparatus, airline equipment, full face and hood respirators, chemical cartridges, etc., is according to regulations established in 30 CFR, Part 11 and includes only equipment approved by the NRC or the National Institute for Occupational Safety and Health's Equipment Certification Manual. Respirator storage is at the entrance to the RCA, in emergency cabinets, and at locations designated by Health Physics. Adequate quantities and types of respiratory equipment is available to support peak respiratory demands.

Airborne Radioactivity Monitoring is normally performed by several methods. Permanently installed particulate iodine and gas monitors are described in Section 12.3.4.2. Mobile continuous airborne monitors (CAMS) are available in the Radiological Controlled Area to provide airborne monitoring for routine operations/ maintenance or abnormal occurrences. Portable high and low volume "grab" type air samples are used to supplement or substitute for the CAMS as an additional method of determining airborne concentrations. Calibration of the monitors use National Institute of Standards and Technology traceable sources.

Forty-two Remote Area Radiation Monitors are located throughout the plant in locations that optimize their use and meet the recommendations of Regulatory Guide 8.12. These monitors are also calibrated according to acceptable methods using National Institute of Standards and Technology traceable sources. A further description of these monitors is provided in Section 12.3.4.1.

Personnel who require entry to the Radiation Controlled Area are normally monitored by the use of Beta-Gamma Thermoluminescent Dosimeter badges. Neutron exposures are evaluated by neutron measurement and stay time, neutron to Gamma ratios and stay time, or neutron sensitive personnel monitoring devices. Readout of TLD badges is through the use of the manufacturer's standard TLD readout equipment. TLD's are processed on a routine basis (Quarterly) or when an individual's exposure status is in question.

Pocket ion chambers (PICs) and/or electronic dosimeters, for determining Gamma exposure are issued to personnel who are entering the Radiological Controlled Area. Selection, use, care and testing of PICs and electronic dosimeters follow the guidelines of ANSI N13.5 1972 and Regulatory Guides 8.4 and 8.14. Extremity badges or other additional dosimetry are issued to personnel on an individual basis as determined by Health Physics. Personnel dosimetry records are maintained using guidelines established in ANSI N13.6 1966 (R1972) and Regulatory Guide 8.7 and meet the requirements of 10 CFR 20.401.

12.5.2.2 Health Physics Facilities

Figures 12.5-1, 12.5-2 and 12.5-3 show Health Physics Facilities located in the Walter P. Chrysler Support Complex, Control, and Radwaste Buildings, respectively.

The main Access Control Facility is located in the Control Building at standard plant elevation 1984 feet. During routine working hours and outages, personnel designated to sign personnel into and out of the RCA are located at the RCA entrance near the Health Physics Office. Personnel requiring assistance to enter the RCA on the off-shift hours, weekends or holidays can contact the Shift Health Physics Technician. Normal entrance to the RCA is via a corridor outside the Health Physics Office. Normal exit from the RCA is through Personnel Contamination Monitors. During major outages, an auxiliary access may be established for the access/egress of additional personnel. Separate male and female toilets are located directly across from the Health Physics Office and are equipped with sinks for noncontaminated washing. Clean shower facilities are located in the female toilet area and the male locker room. The men and women's locker rooms are located next to the respective toilet areas and are normally used for donning of modesty garments, and storage of street clothes. Adjacent to the ingress corridor is the decontamination area. This area is equipped with a decontamination shower and sinks used for personnel decontamination. Decontamination supplies are normally stored in this room. Drains from the hot shower and sinks in these rooms are connected to the Liquid Radioactive Waste System.

Located next to the ingress and egress corridors is a Laundry Room equipped with dryers and washers that can be used to launder modesty garments and contaminated personal clothing. Across from the laundry is the Health Physics Count Room. See Figure 12.5-2.

The Hot Laboratory and Counting Rooms are adjacent to the Health Physics Office. Equipment and instrumentation stored and used therein are described in Section 12.5.2.1. The Hot Laboratory and Counting Room sinks and floor drains are connected to the radwaste system and the fume hoods and room ventilation system are connected to the access control area ventilation system.

A tool and equipment decontamination room is located in the Hot Machine Shop on elevation 2000'. Ultrasonic cleaners and steam spray booths are located in the Decontamination Room and may be used to remove radioactive material from items before maintenance or repair. Depending on contamination levels, air sampling and Respiratory Protection may be used during these decontamination activities. The drains in this room are connected to the Liquid Radioactive Waste System.

Anti-contamination clothing will be maintained in sufficient quantity to support plant work activities. Protective clothing is stored in plant locations designated by Health Physics. Protective clothing requirements are specified by Health Physics and may include:

1) Coveralls - Cotton and Disposable
2) Laboratory Coats
3) Caps and Hoods
4) Rubber and Plastic Shoe Covers
5) Cotton, Plastic and Rubber Gloves
6) Plastic Suits

The Walter P. Chrysler Support Complex (See Figure 12.5-1) contains the HP instrument repair and calibration facility, the TLD processing area, Whole Body Counter, the HP Records Office, and Respiratory facility. Portable Instrument calibration is performed with sources traceable to National Institute of Standards and Technology.

The Bioassay Program follows the guidelines of Regulatory Guide 8.9. Whole body counting is used as required to determine the effectiveness of the Respiratory Protection Program and to assess the internal exposure of individuals who are involved in activities that have the potential for inhalation, ingestion or absorption of radioactive material. Excreta analyses may also be used to verify the uptake of radioactive material. Whole body counting and bioassay analysis is performed by Health Physics or contract personnel.

An additional Sample Room is located in the Radwaste Building at elevation 2000 feet and is for obtaining samples of the Radwaste Systems for analysis. Sample analysis and counting equipment may be located in this area. For details see Figure 12.5-3.

Conformance to Regulatory Guide 8.2 is addressed in Section 12.3.4.1. The means by which the recommendations of Regulatory Guide 8.8 are implemented are discussed at length throughout Chapter 12.0. Regulatory Guide 1.97 is discussed in Table 7.5-4.

12.5.3 PROCEDURES

ALARA Regulatory Guides 8.8 and 8.10 are an integral part of Health Physics procedures and policy developed by the WCGS staff. The use of qualified and experienced personnel in developing and implementing procedures is a tool used to keep exposures ALARA. Detailed written procedures, including applicable instructions and precautions will conform to 10 CFR 20.

The type, frequency, and location, of radiation surveys are outlined in procedures and are conducted in a manner that assures that exposure is ALARA. Survey requirements are delineated in Health Physics procedures and consist of combinations of radiation level, contamination level, and atmospheric particulate and/or radioiodine concentrations. Contamination surveys normally use the "smear" or "swipe" test and are taken at locations that are dependent on factors such as location, occupancy factor, and potential radiological hazard. Decontamination, using acceptable methods and techniques may be performed on areas and equipment to reduce personnel exposure and contamination levels. Areas that cannot be cleaned using these decontamination practices are posted and barricaded per Health Physics procedures. Entry and exit of these areas are controlled through the use of the RWP System. A posting of radiological conditions outside the Health Physics Office at the Controlled Access Area entrance point is available for the information of personnel entering the RCA. Placards, acceptable posting methods and physical controls are outlined in Health Physics procedures that specify proper methods that are in compliance with 10 CFR 20 and Regulatory Guide 8.38, section 1.5. The Health Physics Exposure Records System established is in accordance with federal regulations and follows guidelines established in Regulatory Guide 8.7.

Administrative procedures are employed to effectively control employee exposures and maintain station doses ALARA and will follow the guidance of Regulatory Guide 8.2.

The RWP System is used to specify personnel who may enter the RCA and prescribe the required clothing, dosimetry, respiratory equipment, special instructions, descriptions and information that is relevant to providing proper radiological surveillance and control. The RWP System is outlined in detail in the Health Physics procedures.

Airborne Radioactivity is normally controlled through the use of engineering controls, i.e., the installed Ventilation Systems, HEPA and Activated Charcoal Filters. The use of respiratory protection, decontamination, glove boxes, tents, etc., may be used to further reduce the possibility of personnel exposure to airborne activity in excess of 10 CFR 20 limits.

The Respiratory Protection Program is outlined in the Health Physics procedures, meets the requirements of 10 CFR 20 Subpart H and follows the guidance of Regulatory Guide 8.15. To insure that the Respiratory Protection Program is functioning properly, a method of determining internal exposure, such as whole body counting or bioassay, as discussed in Section 12.5.2, is established.

A Radiation Worker Training Program is developed and implemented for instruction of personnel who have unescorted access to the RCA. The training involved is approved by the Manager Chemistry/Radiation Protection through a Superintendent Chemistry/Radiation Protection and includes information needed to perform work in the RCA. Training includes instruction on station rules and practices; state, local and federal regulations; the basics of radiological health; biological effects of radiation; and ALARA concepts and philosophies. Station personnel receive retraining annually. Posting of notices, instructions and reports to the plant workers is in accordance with 10 CFR Part 19. Personnel who work at the site, whose duties do not require the handling of radioactive material or that they enter RCA,are instructed as to why they may not enter such areas. Plant workers receive prenatal radiation exposure instructions as recommended in Regulatory Guide 8.13.

Personnel monitoring, including internal and external, with the associated record keeping system, is discussed in Section 12.5.2.

Conformance to Regulatory Guides 8.9 and 8.14 is discussed in Section 12.5.2. Regulatory Guide 1.8 is discussed in Section 12.1.3 and Chapter 13.0. Compliance with Regulatory Guide 1.16 is discussed in Appendix 3A, and report content is also discussed in WCGS Technical Specifications. Regulatory Guide 1.33 is discussed in Appendix 3A and Section 13.4. Appendix 3A provides compliance with Regulatory Guide 1.39.

TABLE 12.5-1

HEALTH PHYSICS AND LAB EQUIPMENT

Instrument	Location	Type Radiation Detector	Detected	Estimated Quantity	Remarks
Computer Based Gamma Spectroscopy System	Counting Room	IGE	Gamma	Seven	Primary system, Health Physics, Effluent Samples
Gas Proportional Counter	Counting Room HP Area	-	Alpha, Beta, Gamma	Two	Used for counting Smears, Air Samples and Radio-chemistry Samples
Liquid Scintillation	Counting Room	-	Beta	One	For Tritium Determinations
TLD Reader System	Health Physics Lab	-	Beta, Gamma	One	Personnel Dosimetry and Environmentals (See Note)
Pocket Dosimeter Charger	Health Physics Office	-	-	Ten	Used for reading Pocket Dosimeter
Condenser R Meter or Digital Dosimeter	Health Physics Office or Lab	-	Gamma	Three	Primary Standard for Calibrations

Note: Health Physics will maintain the “Estimated Quantity” of TLD Reader System(s) when TLD’s are being processed on site.

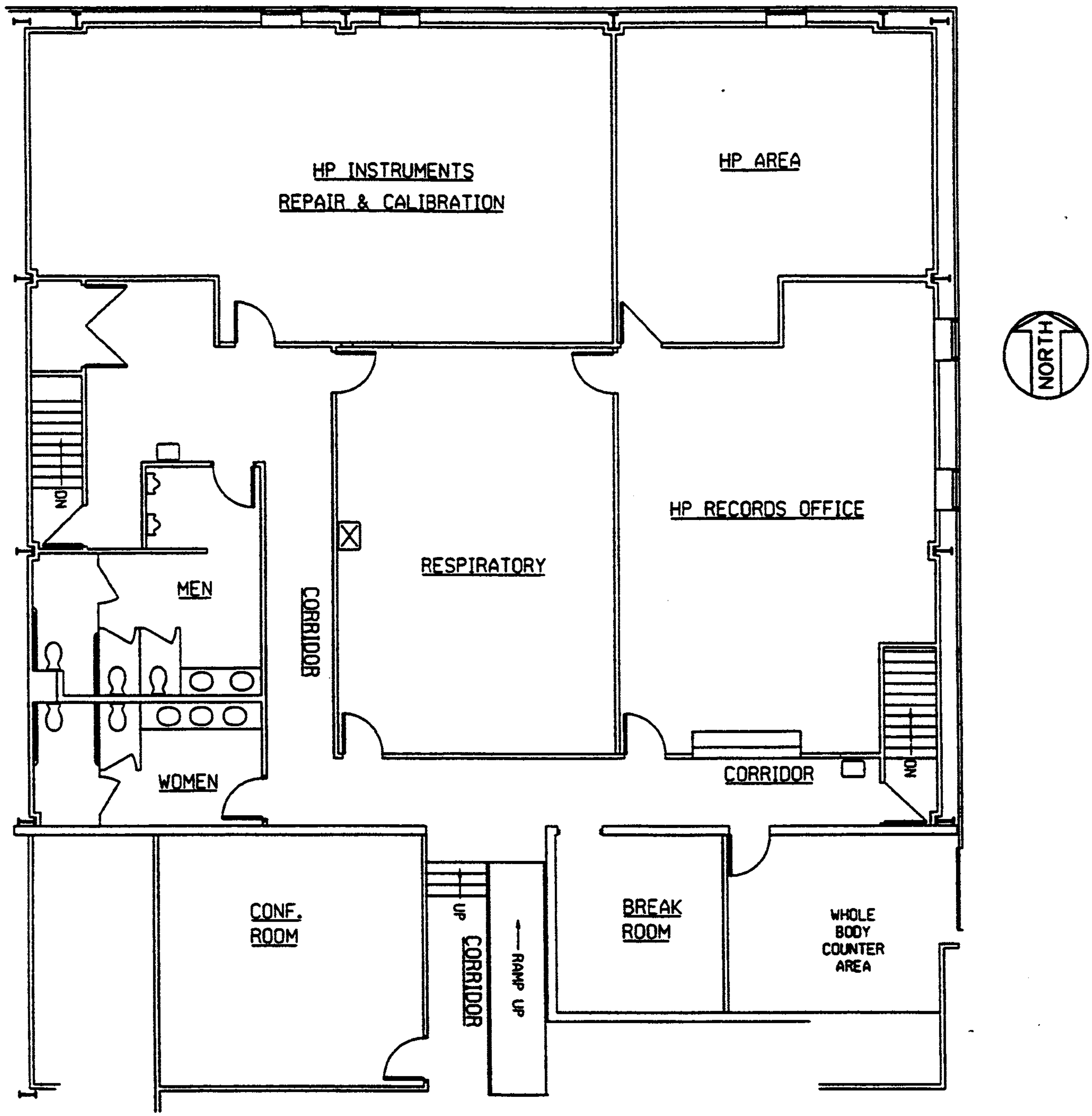
TABLE 12.5-2

PORTABLE HEALTH PHYSICS EQUIPMENT

Instrument	Radiation	Energy Range	Range	Accuracy	Type of Monitoring	Number	Location	Remarks
G-M Survey Meter	Beta, Gamma	- NA-	0-500,000 CPM	± 20% Full Scale	Contamination	10	Health Physics Office	With standard or pancake probe
G-M Survey Meter	Beta, Gamma	0.31-1.33 Mev	0-1 R/HR	± 20% Full Scale	Working Area Radiation	15	Health Physics Office	With Internal Detector
G-M Survey Meter	Beta, Gamma	0.1-3.0 Mev	0-1000 R/HR	± 20% Full Scale	Working Area Radiation	15	Health Physics Office	With Telescoping Probe
Survey Meter	Beta, Gamma	0.1-1.2 Mev	0-20,000 R/HR	± 20% Full Scale	Working Area Radiation	5	Health Physics Office	With extending Probe or Cable
Survey Meter	Beta, Gamma	0.8-2 Mev	0-5 R/HR	± 20% Full Scale	Working Area Radiation	15	Health Physics Office	Ionization Chamber with Beta Shield
Survey Meter	Beta, Gamma	0.8-2 Mev	0-50 R/HR	± 20% Full Scale	Working Area Radiation	10	Health Physics Office	Ionization Chamber with Beta Shield
Neutron REM	Neutron	Thermal-10 Mev	0-2 REM/hr	± 20% Full Scale	Working Area Radiation	3	Health Physics Office	
Neutron REM	Neutron	Thermal-10 Mev	0-100 REM/hr	± 20% Full Scale	Working Area Radiation	3	Health Physics Office	
Scintillation Counter	Alpha	-NA-	0-50,000 CPM	± 20% Full Scale	Contamination	4	Health Physics Office	Alpha Scintillation Crystal with a mylar window
Portal Monitor	Beta, Gamma	80 Kev-1.2 Mev	Variable range Switch	± 20% Full Scale	Personnel Contamination	3	Personnel exits at security and from controlled access	Equipped with audible alarm, Gas proportional or scintillation
Mobile Continuous Airborne Monitor	Beta, Gamma	-NA-	$10\text{-}10^5$ CPM	± 20% Full Scale	Air	5	As directed by Superintendent Chemistry/ Radiation Protection	Equipped with audible alarm

TABLE 12.5-2 (Sheet 2 of 2)

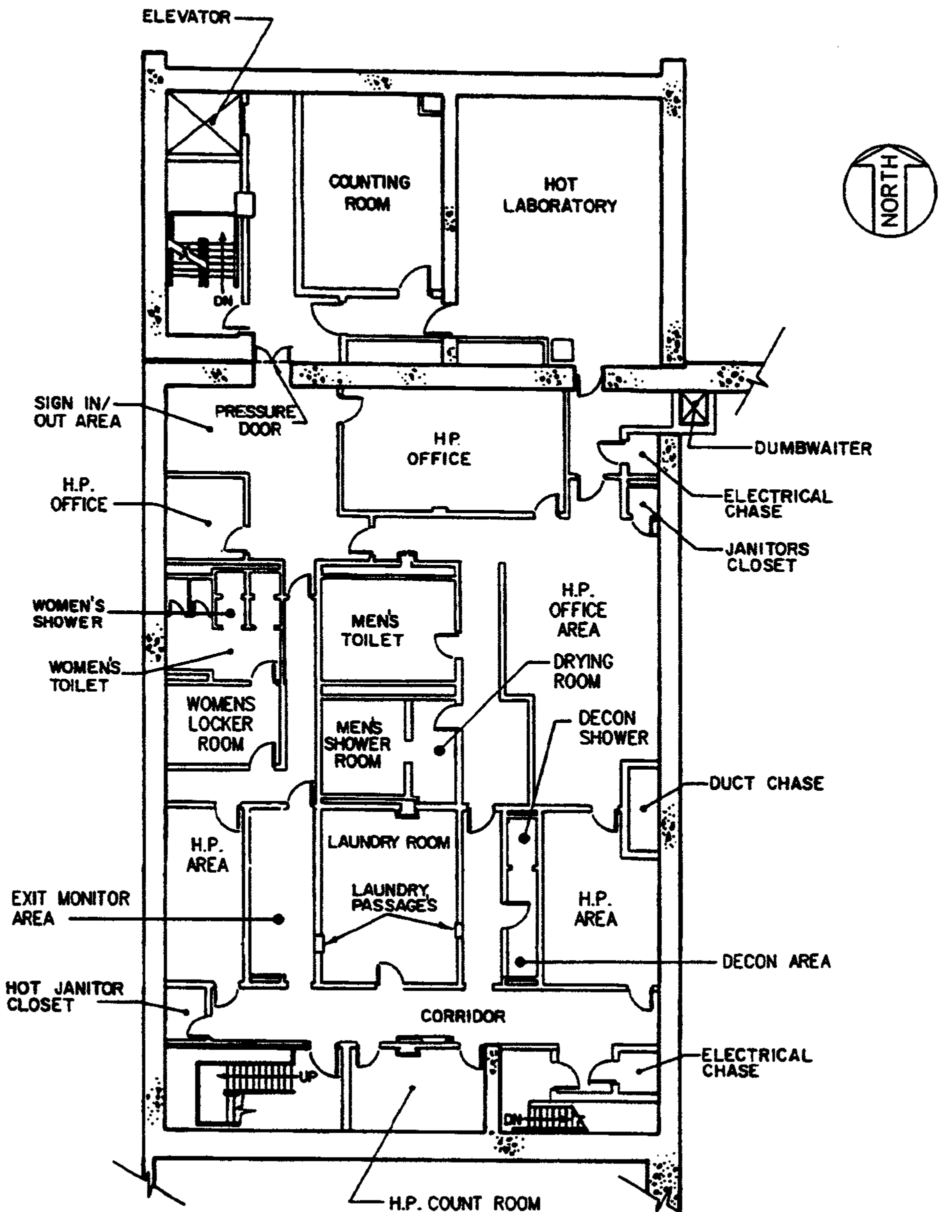
Instrument	Radiation	Energy Range	Range	Accuracy	Type of Monitoring	Number	Location	Remarks
Individual Personnel Monitor	Gamma	60 Kev -1.25 Mev	0-9990 mR	± 20% (10R/Hr)	Individual Exposure	20	Health Physics Office	Audible alarm response propor-tional to dose rate or total dose
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-200 mR	± 15% Full Scale	Individual Exposure	300	Health Physics Office	Integrating, direct reading
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-500 mR	± 15% Full Scale	Individual Exposure	1000	Health Physics Office and Emergency Cabinets	Integrating, direct reading
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-1 R	± 15% Full Scale	Individual Exposure	100	Health Physics Office	Integrating, direct reading
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-5 R	± 15% Full Scale	Individual Exposure	75	Health Physics Office and Emergency Cabinets	Integrating, direct reading
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-10 R	± 15% Full Scale	Individual Exposure	10	Health Physics Office	Integrating, direct reading
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-50 R	± 15% Full Scale	Individual Exposure	25	Health Physics Office	Integrating, direct reading
Pocket Dosimeter	Gamma	80 Kev 1.2 Mev	0-200 R	± 15% Full Scale	Individual Exposure	30	Health Physics Office and Emergency Cabinets	Integrating, direct reading
Electronic Dosimeters	Gamma	55 Kev 6 Mev	0-999 R	± 15%	Individual Exposure	100	Health Physics Office	Integrating, direct reading



NORTH END OF WALTER P. CHRYSLER SUPPORT COMPLEX
SECOND LEVEL

Rev. 13

WOLF CREEK UPDATED SAFETY ANALYSIS REPORT
FIGURE 12.5-1 HEALTH PHYSICS AREA IN THE WALTER P. CHRYSLER SUPPORT COMPLEX



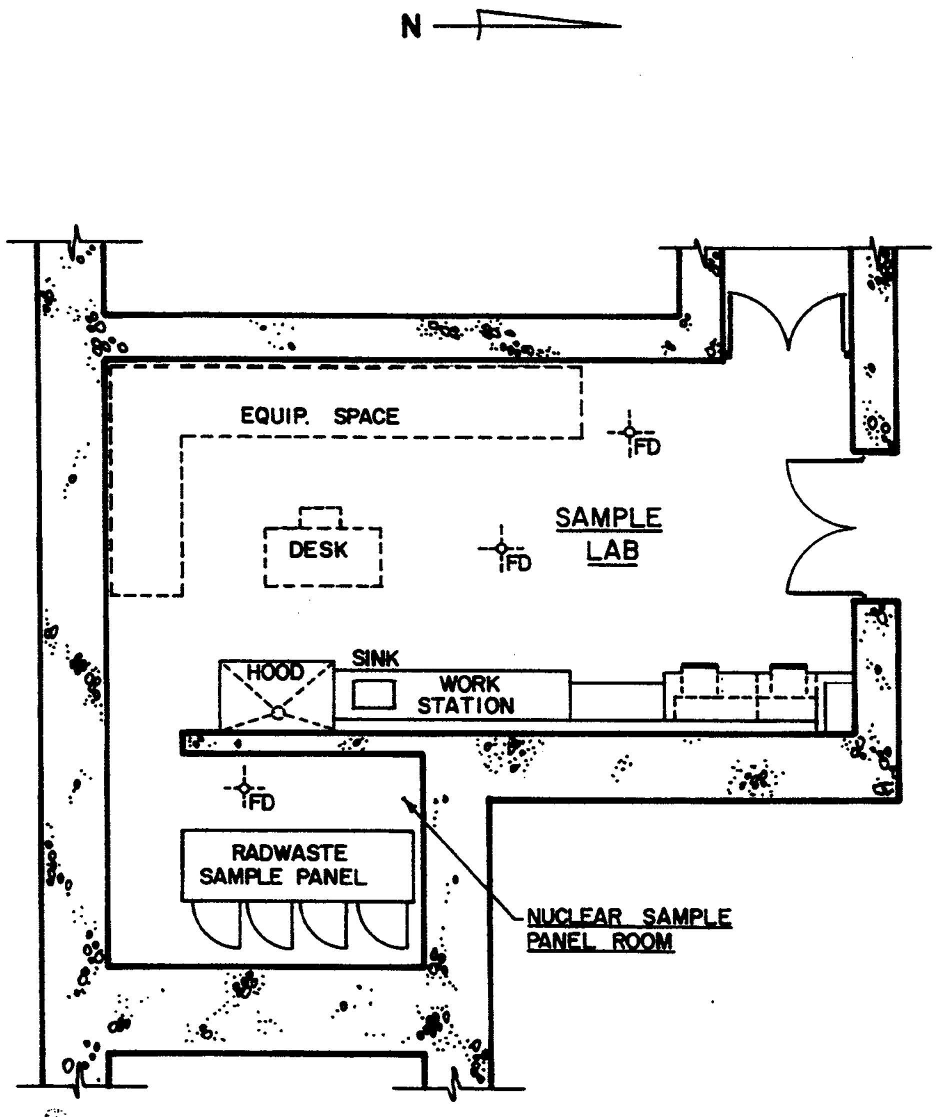
CONTROL BUILDING AT ELEVATION 1984

REV. 10

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.5-2

HEALTH PHYSICS AREA IN THE CONTROL BUILDING



RADWASTE BUILDING AT ELEVATION 2000

Rev. 0

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 12.5-3

SAMPLE LAB FACILITIES IN THE RADWASTE BUILDING