SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 12
RADIATION PROTECTION

SECTIONS

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

12.1.1 Policy Considerations

It is the policy of FPL Energy Seabrook to maintain occupational radiation exposure during Seabrook Station operations as low as is reasonably achievable (ALARA). This policy and philosophy of keeping exposure to radiation ALARA is derived from the fact that ionizing radiation is biologically damaging and that the amount of damage is related to, among other things, the magnitude of the dose received.

The ALARA philosophy, as embodied in Regulatory Guide 8.8, Revision 3 and 8.10, Revision 1, is implemented by employing sound radiation protection practices and techniques as delineated in the Radiation Protection Manual and Health Physics procedures. The implementation of the ALARA philosophy includes all phases of station operations from startup to eventual decommissioning. What is "reasonably achievable," for exposure reduction, is a judgment which all Seabrook Station management personnel are required to make. However, it is also the responsibility of station employees to make judgments regarding their radiation exposure during the performance of their assigned tasks in a radiologically controlled area.

The basis for these judgments should include an assessment of the state of technology and the economics of improvements in relation to all of the benefits from these improvements.

12.1.1.1 Overall ALARA Policy Responsibilities

The Station Director has the overall responsibility and authority for implementing the ALARA philosophy. He delegates this responsibility and authority to the Health Physics Department Supervisor. The Health Physics Department Supervisor ensures that the ALARA philosophy receives proper attention and that adequate resources are made available. He also reviews the progress in the area of exposure reduction, ensures that corrective actions are taken when necessary, and provides overall direction and coordination of the ALARA policy. This may include the following:

a. Participation in design reviews for facilities and equipment that can effect potential radiation exposures

b. Identification of situations that have potential for causing significant exposures to radiation

c. Implementation of an exposure control program (ascertain which jobs should be closely controlled for exposure purposes)

d. Development of procedures and methods for keeping radiation exposures of station personnel ALARA

e. Review of selected job procedures to maintain exposures ALARA

f. Participation in the development and implementation of training programs related to work in radiation areas or involving radioactive materials

g. Establishment of a radiation exposure surveillance program to maintain data on exposures of station personnel by job function and/or specific jobs

h. Performance of trend analyses of station radiological data such as contamination and radiation levels

i. On-the-job inspection of selected tasks in progress to review effectiveness of the ALARA policy

j. Documentation of ALARA efforts as determined necessary

k. Supervising, training and qualifying the radiation protection staff of the station

l. Ensuring that adequate radiation protection coverage is provided for station personnel during all working hours.

A Health Physics professional assists the Health Physics Department Supervisor in performing and coordinating the above ALARA functions and activities. Additionally, health physics personnel have the responsibility for providing exposure reduction guidance to workers during routine support interfaces. Subsection 12.5.1 provides a description of the Health Physics Department organization and responsibilities with regard to radiation protection.

12.1.1.2 Direct ALARA Policy Responsibility

Direct responsibility for implementation of the ALARA philosophy rests with each member of the Seabrook Station management organization. As managers, they are directly responsible for a defined area of the overall station operation and understand this responsibility includes minimizing radiation exposures for both their own personal and other station personnel. In keeping with the overall station goals of providing maximum availability, highest efficiency and the best working environment possible, management will ensure that station personnel perform all radiologically controlled area tasks in accordance with general and/or job-specific procedures and the ALARA philosophy.

This is accomplished through careful consideration of the following general guidelines:

a. When required, develop clear, easily understood, job specific procedures

b. Ensure that work being performed by procedure is accomplished on an orderly and timely basis

c. Plan and schedule work to ensure that it does not interfere with other work in progress in the same area and that individuals not required for the work are not assigned

d. Ensure that workers are familiar with the work assignments, requirements, procedures, and locations

e. Modify methods, techniques and procedures on a timely basis to account for changes in technology, modified equipment and/or identification of inadequacies

f. Provide training to personnel as necessary to improve skills and minimize dependence on certain individuals for performance of specific tasks

g. Investigate and specify the use of exposure-reducing techniques wherever practical and reasonable

h. Coordinate activities and procedures with the Health Physics Department to insure adequate review.

12.1.1.3 Employee Responsibilities

Radiation workers, whether permanent or temporary, are informed of their responsibilities for maintaining ALARA exposure for both themselves and fellow workers. This includes the responsibilities to notify supervisors of procedural changes that should be considered to reduce exposure and to report to radiation protection personnel potential radiological hazards that may result in unnecessary exposure. Employees are expected to actively participate in training programs developed for their specific disciplines and assigned major exposure tasks.

12.1.1.4 Training

To ensure all personnel fully understand the necessity of minimizing their exposure as well as their fellow workers' exposure, a radiation safety training program is provided. The Manager of Nuclear Training is responsible for the conduct of the training program and will coordinate inputs by the Health Physics Department Supervisor and other department supervisors regarding content and concepts. As a minimum, radiation workers are given a basic course in radiation protection, station requirements and federal regulations to understand the requirements for entering a radiologically controlled area.

Additionally, detailed radiation protection instructions are provided to those individuals whose duties require working with radioactive materials, entering radiologically controlled areas (RCA), or directing the activities of personnel who work with radioactive materials or enter radiologically controlled areas. This instruction emphasizes exposure reduction techniques and the ALARA philosophy.

The above training programs and other training programs that are conducted at Seabrook Station are outlined in Section 13.2. All instruction, whether the subject matter pertains to radiation safety, plant systems or craft skills, is intended to result in a training program that promotes the ALARA philosophy through improved workmanship and reliability.

12.1.2 Design Considerations

This subsection deals with the station layout, equipment location, and equipment maintainability, as applicable to the ALARA concept. The objectives of the design considerations, which are consistent with Regulatory Guide 8.8, Section C.3, are to reduce the number of personnel needed to perform work in a radiation area associated with maintenance activities, reduce maintenance time, and minimize radiological conditions. The designers have employed experience from past designs and operating plants to help reduce exposure from components and work on components. Reviews have been conducted by personnel with experience in radiation protection at operating power plants.

12.1.2.1 Reduction of Work Force Exposure to Radiation

There are three basic design considerations which allow a reduction of the work force in a radiation area. The design considerations are:

a. Reliability

b. Location

c. Maintainability.

Each of the above items can be subdivided many times, but emphasis will be placed on generalized designs and considerations.

Emphasis has been given to the reliability of equipment. Increasing the reliability of equipment, by design, decreases the need for inspection, maintenance, and preventive maintenance. When redundancy is used to increase reliability, the maintenance duration and location may not change, but maintenance can usually be performed when radiological conditions are more conducive to ALARA.

Equipment location is used where practical to eliminate exposure by placing equipment in nonradiologically controlled areas. When equipment is placed in the radiologically controlled area, then, when practical, the equipment is located away or shielded from substantially higher radiation areas. Additional reductions in exposure are made possible by locating equipment so that a minimum number of personnel are necessary to perform maintenance; i.e., adequate working and removal space are provided.

The work force necessary to perform maintenance on equipment is dependent on the ease of repairing the equipment. Selection of some equipment and systems is partially based on the ease of repair.

12.1.2.2 Work Time Reduction in the RCA

The time necessary to perform tasks is reduced where practical. The time reduction is accomplished by advanced technology and good general designs, such as equipment location, ventilation, and remote tools.

The use of advanced technology allows inspection and repair work to be performed faster. For example, advanced technology is responsible for an increased speed in performing in-service inspection (ISI) of steam generators and steam generator tube plugging. Improved design of valves seals allows less and faster maintenance. There are many examples of the use of advanced technology in the Seabrook design helping to reduce the plant man-rem figure. The above two examples were chosen due to the high exposure rate associated with the work.

Good quality general designs have been used to reduce the man-rem expenditure. The reduction in man-rem exposure is due to parts of equipment fitting together properly without the need for modifications, along with other components and systems such as the ventilation system described in Subsection 12.3.3. This is extremely effective in reducing exposure due to removal and replacement of insulation for repair and ISI work.

12.1.2.3 Reduction of Radiological Conditions

The facility general design has been reviewed by experienced personnel to ensure that systems, equipment (described in Subsection 12.3.1), ventilation (described in Subsection 12.3.3), and shielding design (described in Subsection 12.3.2) provides acceptably low exposures. The following design features have been incorporated in the RCA where reasonably possible:

a. Shielding of pipe chases and equipment containing radioactive materials (permanent and provisions for temporary)

b. Placement of reach rods for remote control

c. Placement of switches for remote control of equipment

d. Use of low nickel and low cobalt alloys to reduce cobalt problems

e. Leak detectors to reduce the amount of leakage and contamination by rapid detection and, therefore, appropriate action

f. Remote control panels where appropriate, such as for rad-waste processing and fuel transferring

g. Design of pipes for low or no crud trap problems

h. Floor drains for moving radioactive liquids to the Waste Processing System

i. Penetrations are stepped, shielded, or are out of line of site with the source when possible

j. Radiation Monitoring System to detect changes in radiation levels (discussed in Subsection 12.3.4)

k. Limiting personnel access to areas by barricading and/or locking, (discussed in Subsection 12.5.3.3)

l. Provisions for flushing and purging of contaminated systems.

There are many methods of reducing the exposure magnitude which do not require permanent station modification. The following is a partial list of methods available to reduce the exposure rate to workers:

a. Prevention of contamination

b. Decontamination of equipment

c. Removal of components to lower radiation zone for work

d. Before transferring highly contaminated equipment the equipment should be decontaminated, or the contamination should be prevented from spreading by placing the equipment in a container.

e. Use of portable shielding

f. Provisions for air and water filtration.

12.1.2.4 ALARA Design Changes

Periodic review of plant design and equipment for ALARA considerations has resulted in the following changes:

a. Restricted access to the RHR vaults via ladders in the containment building spray heat exchanger room

b. Restricted access to the waste gas regenerative compressor room, waste gas dryer columns, and waste gas valve room in the Waste Processing Building

c. Rearrangement of the primary sample heat exchanger and sink to reduce shine

d. Rearrangement of the evaporator equipment to reduce exposure levels in adjacent walkways.

12.1.2.5 Management of Radiation Protection Design Review - Construction Phase

The Seabrook Station ALARA program for construction, design changes, and reviewing field run piping was the joint responsibility of Westinghouse Electric Corporation (Westinghouse), United Engineers and Constructors Inc., (UE&C), Yankee Atomic Electric Company - Nuclear Services Division (YAEC), and New Hampshire Yankee (NHY).

Westinghouse was responsible for the design, fabrication and delivery of the Nuclear Steam Supply System, related auxiliary systems and the nuclear fuel. Technical direction for the installation of this equipment and technical assistance throughout the preoperational testing, initial core loading and testing programs were further responsibilities of Westinghouse.

United Engineers and Constructors (UE&C) was responsible for the engineering, design and construction management of the station. Included in their scope were the supply and installation of the balance of plant systems, components, and structures so that a complete and integrated facility was assured.

The radiation design review performed by UE&C was the responsibility of the Chief Power Engineer. The Chief Power Engineer ensured that an overall design program was implemented to help maintain occupational radiation exposures As Low As Reasonably Achievable (ALARA) during operation of the facility. He provided appropriate guidance to Chief Discipline Engineers regarding ALARA design implementation and verified implementation.

The chief discipline engineers (electrical, mechanical, instrumentation, etc.) provided for incorporation of ALARA considerations. This was accomplished by providing guidance to engineers responsible for the design of Seabrook Station. The Chief Engineers or designers reviewed the various systems to ensure provided guidance was used.

The Quality Assurance Manual defined contractor responsibilities as follows:

> "Each contractor shall maintain design control measures as required by ANSI N45.2.11. These design measures shall be applied to areas such as the following: ... accessibility for in-service inspection, maintenance and repair..."

The YNSD Project Office established appropriate reviewers as determined by the Quality Assurance Manual and Section 17.1 of the Seabrook Station Updated FSAR. Project policies indicate primary and secondary reviewers of UE&C specifications; Westinghouse specifications; UE&C system descriptions;

Westinghouse system descriptions; Updated FSAR chapters, sections, and subsections, engineering changes and general arrangement drawings of the Containment, Fuel Storage and Primary Auxiliary Building. Project policies also indicated the type of documentation required for reviews. The documentation was in the form of Engineering Review Reports, memoranda or other reports.

a. Design Reviewers

YNSD Radiation Protection personnel did not always possess the necessary expertise to perform complete ALARA reviews. The YNSD Radiation Protection Group relied on engineers with the expertise to determine equipment compatibility, accessibility (ladders, platforms, laydown space), operability and maintenance history (low-maintenance). Individuals performing reviews were usually Engineer grade or higher (B.S. degree or equivalent and 3 years professional experience). Individuals within departments who performed the reviews were chosen based on their general knowledge of the system and equipment. The departments within YNSD who performed reviews are listed below with some of their responsibilities:

1. Plant Engineering Department (Instrumentation and Control Group, Electrical Engineering Group, Mechanical Engineering Group and Systems Engineering Group)

   - Supported the Project Office in general and detailed technical review and guidance for plant concept, design construction and licensing in the field of Fluid Systems, Instrumentation and Control, Electrical Engineering, Mechanical Engineering, Systems Engineering, Materials Engineering and Structural Engineering.

   - Coordinated electrical and control design between the architect-engineer and nuclear steam system supplier.

   - Reviewed conceptual design and detailed engineering of all assigned primary and secondary fluid systems, included types of components selected, modes of operation and physical arrangements.

   - Reviewed all electrical and control equipment specifications, logic and wiring diagrams. These reviews included transformers, motors and switchgear, plant control devices, nuclear instrumentation and reactor control and protection system equipment.

2. Nuclear Engineering Department (PWR Transient Analysis Group, Reactor Physics Group and LOCA Analysis Group)

   - Reviewed all reactor physics design work performed by the reactor supplier to assure adherence to the design criteria and the use of adequate methods and assumptions.

   - Reviewed and/or participated in the design of instrumentation for core monitoring.

   - Verified that operational requirements were given adequate consideration and were appropriately factored into the design.

   - Assisted in the development of Technical Specifications and station operating procedures for accident conditions.

   - Reviewed the various station anticipated transients and accidents to ensure conformance with all applicable criteria.

   - Reviewed and analyzed data from the station to verify the reactor physics design.

3. Fuel Management Department (Nuclear Materials Group, Economic Analysis Group and Core Components Group)

   - Reviewed and approved mechanical designs and specifications for nuclear fuel assemblies and components.

   - Reviewed specifications, procedures, purchase orders and drawings for proper definition of quality assurance requirements for nuclear fuel assemblies.

4. Environmental Engineering Department (Radiological Engineering Group, Radiation Protection Group, Environmental Sciences Group and Environmental Laboratory)

   - Established functional requirements of engineered safeguard systems and evaluation of their performance.

   - Participated in the design and review of solid, liquid and gaseous radioactive waste treatment systems.

- Participated in the establishment of system design requirements for plant process and area Radiation Monitoring Systems.

- Participated in the establishment of Station Radiological Equipment and Facilities.

12.1.2.6 Independent Reviews

The ongoing interactions between NHY, YAEC, Westinghouse and UE&C engineers and radiation protection personnel resulted in continuous cross-checking or "independent" reviews of each organization's design, construction and operational activities. Proposals to modify or establish designs received appropriate levels of review by these diverse organizations.

Additionally, these organizations recognized that professional contractor organizations are available for use, as necessary, to provide assistance in specialized areas of radiation protection.

Contractor organizations have been used to provide specialized evaluations in such cases as the analysis of the proposed removable shielding for the reactor vessel annulus and the assessment of the radiological impact of the spent resin transfer system design. Such special evaluations significantly contributed to design review efforts directed toward ensuring that occupational exposures are maintained ALARA.

12.1.2.7 Field Reviews

The later example, cited in Subsection 12.1.2.6, is an instance of the onsite ALARA reviews conducted during the construction phase under the auspices of the Seabrook Station Health Physics Department. These evaluations were performed on systems, components and areas in accordance with station ALARA policies.

These evaluations were used to identify potential beneficial modifications, as well as to provide background information for use during operations. Expertise and assistance was obtained, as necessary, from other applicable station departments.

Coordination of major actions and the final decisions were the responsibility of appropriate station and corporate management. YNSD and, when necessary, UE&C were party to these station activities.

Day-to-day aspects of this ALARA effort were conducted by station Health Physics Technicians supervised by Health Physics Working Foremen under the cognizance of health physics supervisors. A Health Physicist - ALARA also participated in the daily efforts as well as coordinating long-term ALARA-related activities. Minimum experience and educational qualifications for Health Physics Department personnel are described in Subsection 12.5.1.

The onsite ALARA field review program, in concert with ALARA design considerations addressed by Westinghouse, UE&C, YNSD and selected professional contractors, as discussed in previous sections, is used by Seabrook management to ensure that the construction and operation of Seabrook Station results in occupational radiation exposures that are as low as reasonably achievable.

12.1.3 Operational Considerations

Operational considerations concern the station personnel efforts of ensuring the application of the ALARA philosophy to both day-to-day routines, and periodic tasks such as special maintenance, in-service inspections and refueling. In all cases, these efforts are consistent with the established radiation protection program.

12.1.3.1 General Operational Concept

As described in Subsection 12.1.1, the responsibilities for maintaining exposures ALARA does not rest with just one or several individuals, but rests with all personnel involved with planning, supervising or implementing work in radiological areas. The health physics department is charged with the overall responsibility and likewise, will be the most influential and informed department for specifying and ensuring the use of exposure-reduction techniques and methods by station personnel wherever and whenever practicable.

When a task is scheduled to be performed on any system that contains, collects, stores or transports radioactive liquids, gases and/or solids, the specific current and/or anticipated radiological conditions involved with that task will be determined. Health Physics factors the task complexity, difficulty, actual and/or projected radiological conditions, and a knowledge of plant systems into the specification of radiation protection and control requirements for the performance of the job. In addition, health physics provides input into the job planning and preparations so the job can be performed in a radiologically safe manner and with the least practical exposure. Where possible, the total exposure expenditure for the job (collective dose of all those personnel planned to be involved) is estimated. Such estimates permit manpower planning by management, but more importantly, they aid in the identification of those jobs that would likely result in a substantial total exposure and accordingly, where a significant dose reduction could be realized. Means to reduce exposure which could then be employed include, among others, the following:

a. Preoperational briefing by health physics personnel

b. Development of special procedures

c. Specific training on a mock-up of a component, equipment or structure

d. Development and use of special tools

e. Use of localized ventilation

f. Decontamination of equipment/areas

g. Draining, flushing or filling of pipes or components

h. Use of contamination containing devices (i.e., liquid collection, drop cloths, containments or tents)

i. Removal of equipment to areas of lower dose rates

j. Prefabrication of complicated equipment or structure

k. Radiological surveillance during a job to identify changing conditions

l. Provision for periodic supervision

m. Contracting specialist services.

In most instances, the collective exposure from using any of the above techniques or others and from the actual performance of a job must be considered to ensure actions are feasible and will not result in an increase of total exposure (although this may be necessary in special circumstances). Additionally, post-operation debriefings may be used, as appropriate, to identify what problems were encountered and to determine how the job techniques may be modified to reduce exposure in the future.

12.1.3.2 Procedure Development

Each department supervisor is responsible for developing department procedures that ensure compliance with applicable regulations. These procedures, developed as discussed in Section 13.5, also provide instructions to personnel for the performance of routine and special tasks that may include work in a radiologically controlled area of the station. These tasks include, among others, refueling operations, radioactive waste handling, in-service inspections, process sampling and surveillance, instrument calibrations and maintenance. Radiation protection supervisory personnel review procedures involving work with radioactive materials or work to be performed in radiation or high radiation areas, as defined in the Radiation Protection Manual.

Jobs with a significant radiological risk (e.g., above a projected dose threshold) undergo a documented planning and review process. This planning includes a review of procedures and techniques, work site provisions, worker preparation and training, and radiological considerations which can minimize the radiological risk to the worker. Personnel with relevant expertise are given prime responsibility for planning such jobs and documenting their preparations, and Health Physics supervisory personnel oversee and approve the final preparations.

12.1.4 Decommissioning Considerations

Seabrook Station relies on preparations in several areas to ensure that occupational radiation exposures are maintained as low as reasonably achievable (ALARA) during plant decommissioning.

As discussed in the following subsection, Seabrook incorporates design features that offer significant exposure reduction during decommissioning. These design features are merely the basis for the performance of an ALARA-oriented radiation protection program during plant operations, as well as effective ALARA preparations during decommissioning planning.

An important aspect of the decommissioning procedure is the use of specific ALARA practices tailored to deal with the particular decommissioning method employed. Delineation of these specific ALARA practices (including engineering design modifications) takes place during decommissioning planning, after information concerning the specific decommissioning method becomes available. Consistent with the guidance provided by 10 CFR 20 and Regulatory Guide 8.8, Revision 3, the specific practices implemented will be based on "as assessment of the state of technology and economic considerations" prevalent at the time of decommissioning. The state of technology and the economics that will prevail 40 years in the future are unknown factors and, therefore, performance of a cost benefit analysis is precluded at this time.

The commitment to formulate and implement the ALARA philosophy during decommissioning includes an acknowledgement and understanding that the process of preparing for eventual decommissioning with occupational exposure as low as reasonably achievable is ongoing in nature.

12.1.4.1 Design Features Contributing to ALARA during Decommissioning

Many basic ALARA design features incorporated into Seabrook Station for operations, maintenance and refueling will enhance exposure reduction during those phases and also during decommissioning, regardless of the specific decommissioning procedure. In effect, this is a generic, ALARA approach to operations and decommissioning.

Specific design features that will be used to maintain occupational radiation exposures ALARA include:

a. A separate building exists for waste processing and disposal that ensures availability of waste processing facilities while other systems and components are being maintained or dismantled.

b. The main hoist of the polar crane has been derated to 302 tons. The crane must be refurbished before it is capable of removing the reactor vessel or the steam generators. There is adequate capacity as is to lift the pressurizer. These components can be removed with minimal displacement of permanent concrete shielding to afford its maximum effectiveness.

c. Seabrook containments are equipped with 27-foot diameter equipment hatches that facilitate removal of large equipment intact.

Generic design features used to maintain occupational radiation exposure ALARA include:

a. Location of Equipment

As stated in Updated FSAR Subsection 12.1.2.1, paragraph (4), "Equipment location is used, where practical, to eliminate exposure by placing equipment in nonradiation control areas." This philosophy is embodied in the segregation of areas with radioactive systems and components. The design philosophy "to minimize the extent of areas housing radioactive equipment and piping through efficient arrangement of equipment and systems" as stated in Updated FSAR Subsection 12.3.1b.

b. Equipment Accessibility and Removability

Updated FSAR Subsection 12.1.2.1 indicates that equipment is designed and located to maximize accessibility to facilitate rapid, efficient work. Equipment is also designed and placed to enhance removal operations and thus, minimize exposure time.

c. Plant Layout

Updated FSAR Subsection 12.3.1.3a explains that "plant layout includes optimal location of radioactive components."

d. Flush/Drain Connections

The provision of flush and drain connections on many systems and components enables extensive chemical decontamination prior to operating phase maintenance and later, decommissioning.

e. Corrosion Control

Internal accumulation of radioactive material is limited through effective corrosion-control methods. Careful selection of plant materials and an aggressive chemistry control program greatly reduce source terms that must be dealt with during operating phase maintenance and decommissioning.

12.2 RADIATION SOURCES

The primary source of neutron and gamma radiation is the reactor core and Primary Coolant System. The secondary sources are contained in, or emanate from, the auxiliary systems and components not included as part of the Nuclear Steam Supply System. Radioactivity in these systems generally originates from:

- Fission products that have escaped into the coolant and are either deposited or carried into other systems, and
- Corrosion products activated while passing through the reactor core.

The basis for calculation of radiation source terms for shielding purposes for the various plant systems for normal operation is the concentrations of fission and corrosion products in the reactor coolant during operation at one percent failed fuel. The bases for calculation of radioisotope concentrations in the reactor coolant, including assumed failed fuel percentage and the applicability of ANSI N237 and Regulatory Guide 1.112, are discussed in Subsection 11.1.1. Source terms for accident conditions are addressed in Chapter 15.

The radiation sources and associated input parameters, assumptions, and methodology described in Section 12.2 represent those used to establish original shielding design. The radiation sources utilized for the design of the primary shield were developed by the reactor vendor and are based on a standard four loop Pressurized Water Reactor. The radiation sources in the spent fuel and the N-16 source were also developed by the reactor vendor, and are based on a reactor power of 3565 MWt. The radiation sources in the primary coolant system and the auxiliary systems are based on a reactor power of 3654 MWt, a one-year fuel cycle, and 1 percent fuel element defects.

The impact on plant shielding requirements was evaluated for an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle. This represents a minor change from the original design basis. The conservative analytical techniques used to establish the original shielding requirements, and the Station Technical Specification which will restrict the reactor coolant activity to levels significantly less than the 1 percent fuel defects, ensure that operation at the licensed core power level will have no significant impact on shielding requirements and safe plant operation.

With the exception of Table 12.2-11, Table 12.2-27 and Table 12.2-29, data contained in Table 12.2-1, Table 12.2-2, Table 12.2-3, Table 12.2-4, Table 12.2-5, Table 12.2-6, Table 12.2-7, Table 12.2-8, Table 12.2-9, Table 12.2-10, Table 12.2-11, Table 12.2-12, Table 12.2-13, Table 12.2-14, Table 12.2-15, Table 12.2-16, Table 12.2-17, Table 12.2-18, Table 12.2-19, Table 12.2-20, Table 12.2-21, Table 12.2-22, Table 12.2-23, Table 12.2-24, Table 12.2-25, Table 12.2-26, Table 12.2-27, Table 12.2-28, Table 12.2-29, Table 12.2-30, Table 12.2-31, Table 12.2-32, Table 12.2-33, Table 12.2-34, Table 12.2-35, Table 12.2-36, and Table 12.2-37 are independent of one or two unit operation. The data contained in Table 12.2-11, Table 12.2-27, and Table 12.2-29 are based on a two-unit operation with the relevant assumptions provided in the appropriate sections.

12.2.1 Contained Sources

The sources contained in the equipment of the Primary, Auxiliary and Radioactive Waste Management Systems and all other major sources of radiation during normal operation are described in this chapter. The components of these systems are represented by cylinders approximating their actual geometry in the computer code used to calculate shielding requirements. The location of all equipment containing radioactive sources is shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15 and Figure 12.3-16.

12.2.1.1 Source Terms for Reactor Core and Spent Fuel Pool

a. Source Terms for the Reactor Core

The major contribution of neutron radiation levels external to the reactor biological shield is due to fast neutron leakage from the reactor cavity during power operation. Reactor shielding design is based on a reactor vendor-supplied source term, the result of a quadrature analysis using the DOT III-W code. The source term consists of fifty-three angular fluxes at 216 mesh intervals for thirteen neutron energy groups at the reactor vessel surface. Neutron and gamma flux spectrums are given in Table 12.2-1 and Table 12.2-2.

b. Source Terms for Spent Fuel Pool Shielding

Shielding design source terms for the high density spent fuel pool are based on the "worst-case" of a full core of 193 fuel assemblies with 100 hours decay, and are given in Table 12.2-3.

12.2.1.2 Source Terms for Spent Fuel Transfer

Shielding for spent fuel transfer is based on a source term supplied by the reactor vendor, shown in Table 12.2-3.

The source strength given represents an average fuel assembly with four days decay after an irradiation time of 3.1 years. The fuel assembly is approximately 152 inches long and 8.5 inches square in cross section.

The bounding source term used to evaluate minimum water depth requirements to maintain dose rates below 2.5 mrem during fuel movement in the spent fuel pool is shown in Table 12.2-3. The source strength given represents a peak fuel assembly with 80 hour decay after an irradiation time of 69,000 Mwd/Mtu. The fuel is approximately 152 inches long and 8.5 inches square in cross section.

12.2.1.3 Sources in the Reactor Coolant System (RCS)

a. Reactor Coolant System Nitrogen-16

The N-16 activity of the coolant is the controlling radiation source in the design of the RCS secondary shielding, and is given in Table 12.2-4 as a function of transport time in a reactor coolant loop.

b. Deposited Corrosion Products

The most significant radiation sources encountered during normal maintenance and inspection of most plant equipment (pumps, heat exchangers, tanks, valves, and other out-of-core primary equipment) are deposits from the reactor coolant, such as activated corrosion products and some fission products.

The activity of the deposits is predominantly due to Co-58 and Co-60. It is estimated that 50 to 90 percent of personnel radiation exposure can be attributed to Co-58 and Co-60, each of which contributes about equally to the exposure.

Cobalt in the Reactor Coolant System is minimized by limiting the cobalt content of materials in contact with the reactor coolant. Low cobalt materials are specified for the steam generator, pressurizer, reactor coolant pump, reactor coolant loop piping, and reactor core internal structures. In addition, chemical treatment and analysis techniques are used to minimize cobalt buildup. At plant operating conditions, the primary coolant chemistry is designed to inhibit the corrosion of materials in contact with reactor coolant. At refueling shutdowns, during the cooldown period, hydrogen peroxide (H_2O_2) can be added to the Reactor Coolant System to oxidize and solubilize Co-58 and Co-60. The solubilized cobalt is removed, by demineralization, prior to lifting the reactor head. These techniques minimize the concentration of Co-58 and Co-60 in the refueling water and reduce personnel exposures during the refueling operation.

12.2.1.4 Sources in the Chemical and Volume Control System (CVCS)

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

The most important ion exchangers in the CVCS, from a shielding standpoint, are the mixed-bed and the cation-bed demineralizers. The mixed-bed demineralizers are normally in continuous use. They remove fission products in cation and anion form, and are very effective in removing corrosion products. The cation-bed demineralizer is used intermittently to remove lithium for pH control, and is very effective in removing the monovalent cations cesium and rubidium.

Noble gases are removed from the letdown purification flow by the letdown degasifier and are processed by the Radioactive Gas Waste System. During periods when the letdown degasifier is in service with oxygenated letdown, residual fission gases may be discharged to the plant vent via the aerated vent header.

The boron thermal regeneration demineralizers are used to regulate the boron concentration in the reactor coolant water during load follow operations to remove boron from the coolant as the nuclear fuel is depleted. These demineralizers collect any remaining radioactive anions, such as iodine and bromine, which may have passed through the mixed bed demineralizer.

The regenerative and excess letdown heat exchangers are located in the Containment Building. They provide the initial cooling for the reactor coolant letdown. Their radiation sources include Nitrogen-16.

The letdown heat exchanger provides secondary cooling for the reactor coolant letdown flow prior to its entering the demineralizers. The seal water heat exchanger cools water from several sources, including the reactor coolant discharged from the excess letdown reheat heat exchanger. During boron release, the tube side of the letdown reheat heat exchanger heats the letdown water before the water enters the boron thermal regeneration (BTR) demineralizers.

The geometry of the various components is described in Table 12.2-6. Locations of the equipment in this system are shown in Figure 12.3-1, Figure 12.3-5, Figure 12.3-6 and Figure 12.3-7.

The source term for each component in this system has been developed individually, with regard to its function and relationship to the rest of the system. The bases and assumptions for the calculation of the source terms for the CVCS equipment follow:

a. Regenerative Heat Exchanger

This component contains reactor coolant concentrations of fission and corrosion products as well as N-16 activity. The shielding source term is given in Table 12.2-5 and was used for both tube and shell sides of the heat exchanger.

b. Excess Letdown Heat Exchanger

The source term used for shielding this heat exchanger (tube side) is the same as that used for the regenerative heat exchanger. The shell side contains cooling water.

c. Letdown Heat Exchanger

The tube side of this heat exchanger contains reactor coolant concentrations of radioisotopes given in Table 11.1-1. The shell side contains cooling water.

d. Mixed Bed Demineralizers and

Cation Bed Demineralizer

The source terms for these components, given in Table 12.2-5, are calculated through buildup and decay of fission and corrosion products from the reactor coolant over a one-year period. Anion and cation removal efficiency of 100 percent is assumed.

e. Letdown Degasifier Regenerative Heat Exchanger

The source term for the tube side is demineralized reactor coolant, while the shell side is the same less the noble gases. Isotopic source strengths are given in Table 12.2-5.

f. Letdown Degasifier Preheater

The shielding for this component is based on the tube side containing demineralized reactor coolant and the shell side containing cooling water.

g. Letdown Degasifier

This component is represented by three different source terms: a vapor region, a liquid-vapor region, and a liquid region. The liquid region is demineralized reactor coolant less the noble gases.

The vapor region is assumed to be occupied with noble gases and carried-over iodine. The concentrations are based on an input flow of 120 gpm, a gas removal of 0.25 scfm and partition factors of 1 for noble gases and 0.0075 for iodines. The source term for the liquid-vapor region is assumed to be demineralized reactor coolant including noble gases.

h. Letdown Degasifier Recirculation Pump and Letdown Degasifier Trim Cooler

The source term for these components is the same as that for the degasifier liquid region.

i. Volume Control Tank

The source terms for this tank are based on input from the excess letdown stream which is pure reactor coolant isotopic concentrations. The liquid region source term is assumed to be reactor coolant less the noble gases. The equilibrium concentrations for the vapor region are calculated using an input flow rate of 120 gpm and a purge rate of 0.7 scfm. The volume control tank stripping efficiency is assumed to be 0.4.

j. Charging Pumps

The source term for these pumps is the same as that for the liquid region of the volume control tank.

k. Seal Water Heat Exchanger

The tube side of this heat exchanger has reactor coolant concentrations of isotopes. The shell side contains cooling water.

l. Moderating Heat Exchanger

The fluid on the tube side of this component contains demineralized reactor coolant. The shell side has the same fluid except for the iodine concentration. Iodine is assumed to be released by the boron thermal regeneration demineralizers after an accumulation period of 12 hours.

m. Letdown Chiller Heat Exchanger

This component contains demineralized reactor coolant (tube side) and cooling water (shell side).

n. Letdown Reheat Heat Exchanger

This heat exchanger contains fluid with reactor coolant concentrations on the tube side and demineralized reactor coolant on the shell side.

o. Thermal Regeneration Demineralizer

The input to these demineralizers is reactor coolant which has been processed by the mixed bed demineralizers. The thermal regeneration demineralizers are anion ion exchangers which accumulate iodines.

The buildup of iodine is based on a flow rate of 120 gpm of fluid in which 10 percent of the reactor coolant iodines pass through the mixed bed demineralizers.

Accumulation time is assumed to be 12 hours. Demineralizer efficiency is taken to be 100 percent.

p. System Liquid Filters

Shielding for the reactor coolant filter and the demineralizer prefilter is based on a contact dose rate of 500 rem/hr. All other filters in this system are shielded for a contact dose rate of 100 rem/hr.

12.2.1.5 Sources in the Residual Heat Removal System

Shielding for the components of the Residual Heat Removal System is based on shutdown source terms shown in Table 12.2-7. These source terms are the reactor coolant isotope concentrations with a four-hour delay during which the Chemical and Volume Control Purification System operates.

The components of this system are represented in the computer shielding code as cylinders approximating the actual geometries. The dimensions of these cylinders are given in Table 12.2-8.

The locations of the components are shown in Figure 12.3-4.

12.2.1.6 Sources in the Steam Generator Blowdown System

The input to the Steam Generator Blowdown System consists of steam generator secondary side blowdown at a rate of 75 gpm. Concentrations of isotopes on the secondary side are determined by a primary-to-secondary leak rate of 500 gpd. Isotopic composition of system input fluid is given in Table 12.2-9. Geometry of the equipment in this system is contained in Table 12.2-10. The location of the system components is given in Figure 12.3-7 and Figure 12.3-12.

The basis for the isotopic composition of the input to the steam generator blowdown evaporator trains is for an anticipated operational occurrence involving a primary-to-secondary leak rate of 0.5 gpm for a duration of 90 days. The location of the evaporator trains is shown in Figure 12.3-12.

a. Blowdown Flash Tank

The source concentration of the liquid is 10/7 of the system input fluid concentrations. The isotopic concentration in the steam is calculated using a carryover fraction of 0.05 and a specific volume for saturated steam at 60 psia.

b. Flash Tank Bottoms Cooler

The source strength of this component is the same as the liquid phase for the flash tank.

c. Flash Steam Condenser and Cooler

The source strength of this component is based on the carryover fraction from the flash tank and is shown in Table 12.2-9.

d. Flash Tank Distillate Pumps

The source term for these pumps is the same as that for the condenser/cooler.

e. Blowdown Evaporator

The inventory of the bottoms fluid is a 90-day accumulation of the fission and activation products from a 0.5 gpm reactor coolant leak. Specific activity is based on 1300 gallons of bottoms.

f. Bottoms Pump and Bottoms Cooler

Specific source term for these components is the same as the evaporator bottoms.

g. Evaporator Distillate Condenser, Distillate Accumulator and Distillate Pump

The source term for evaporator distillate is calculated using decontamination factors of 10^0 for noble gases, 10^2 for iodine and 10^3 for other fission and activation products.

h. Demineralizers

The Steam Generator Blowdown System demineralizers are not utilized when the secondary side of the plant has significant contamination as noted for the conditions above. For incidental levels of contamination on the secondary side, the Blowdown Demineralizer System may be utilized as long as the general area radiation levels adjacent to the demineralizer vessels are maintained within the Zone II radiation limits.

12.2.1.7 Sources in the Boron Recovery System

The input to this system has been processed by the letdown purification system and is, therefore, demineralized and degasified. The isotopic source terms for this fluid are given in Table 12.2-11.

Input flow rate is assumed to be 341 lb/hr per unit during normal operation and 200 gpm for a unit during shutdown. System equipment geometries are described in Table 12.2-12.

Location of equipment is shown in Figure 12.3-8, Figure 12.3-9, and Figure 12.3-10.

a. Primary Drain Tank Demineralizers

Source term for this component buildup considers normal bleed rate (0.68 gpm/unit) for a period of one year (7008 hrs. at 80 percent capacity factor) followed by a refueling shutdown. The isotopic content is given in Table 12.2-11. Ion removal efficiency is assumed to be zero for iodines and unity for all other isotopes.

b. Boron Waste Storage Tanks

Maximum anticipated inventory for a boron waste storage tank occurs with a full tank following a shutdown. The fluid has been processed by the cesium removal ion exchanger for which the following D.F.s were assumed; 10^0 for iodines and 10^1 for all other isotopes. The source strength for this tank is shown in Table 12.2-11.

c. Recovery Evaporator Feed Pump

Isotopic source strength for these pumps is assumed to be the same as the input to the boron waste storage tank.

d. Recovery Evaporator

Bottoms concentrations of isotopes are calculated on the basis of processing one full boron waste storage tank at a feed flow rate of 25 gpm. Distillate concentrations are based on assumed D.F.s of 10^3 for iodine and 10^2 for other solids.

e. Recovery Evaporator Bottoms Pump, Recovery Evaporator Bottoms Cooler, Recovery Evaporator Reboiler Pump and Recovery Evaporator Reboiler

The shielding for these components is based on the source term for the evaporator bottoms concentration. This source term is shown in Table 12.2-11.

f. Recovery Evaporator Distillate Condenser, Recovery Evaporator Distillate Accumulator, Recovery Evaporator Distillate Pump, Recovery Evaporator Distillate Cooler and Recovery Test Tank Pump

The shielding for these components is based on the distillate source term described in Table 12.2-11.

g. Recovery Test Tank

Source term calculations for this component are based on an input flow rate of 25 gpm of distillate condensate for buildup and decay through a 12-hour filling period. Activities for this component are presented in Table 12.2-11.

h. Recovery Demineralizer

Source term calculation for this component considers a one-year buildup from recovery test tank input and a 100 percent efficiency of removal for all isotopes. The maximum inventory of activity is given in Table 12.2-11.

i. Recovery System Filters

Shielding for all filters in this system is based on 100 rem/hr. contact dose rate.

12.2.1.8 Sources in the Primary Drain System

The input to this subsystem of the Boron Recovery System is the drainage from various primary side equipment. The isotopic composition and strength is assumed to be untreated reactor coolant including all noble gases and iodine. This source term is given in Table 1.1-1. The geometry of each component of the system is described in Table 12.2-14. The location of the equipment in this system is shown in Figure 12.3-8, Figure 12.3-9, and Figure 12.3-10.

a. Primary Drain Tank (PDT), Primary Drain Tank Transfer Pump, Primary Drain Regenerative Heat Exchanger and Primary Drain Tank Degasifier Preheater

The isotopic concentrations in the source terms for these components are assumed to be the same as the reactor coolant as given in Table 11.1-1.

b. Primary Drain Tank Degasifier

Shielding for this degasifier is based on two different source terms, one for the intermediate liquid-steam region and the lower all liquid region and the second term for the upper gas region. The liquid and liquid-steam source term is assumed to be reactor coolant (undemineralized) less the noble gases. The upper region source term is taken to be noble gas and iodine. Isotope concentrations for this component are given in Table 12.2-13.

c. PDT Degasifier Recirculation Pump and PDT Degasifier Trim Cooler

The source term for these components is the same as the liquid source term for the degasifier.

d. PDT Degasifier Prefilter

Shielding for this filter is based on 100 rem/hr contact dose rate.

12.2.1.9 Sources in the Spent Resin Sluicing System

The Spent Resin Sluicing System collects the spent resin from all the demineralizers and ion exchangers of the nuclear plant. The geometry of the components of this system is described in Table 12.2-16. The location of this equipment is shown in Figure 12.3-8 and Figure 12.3-9.

a. Spent Resin Sluice Tank and Spent Resin Transfer Pump

The source term for this equipment is based on the accumulation of the fission and activation products from the reactor coolant inventory over a period of one year. Isotopic source strength is given in Table 12.2-15. Those isotopes with half-life of less than one day are neglected unless produced from a long-lived isotope.

b. Spent Resin Sluice Pump

This pump is protected from processing the spent resin by a strainer in the spent resin sluice tank. However, the source term for this pump assumes 100 ppm of spent resin escapes the strainer, giving a source term which is 10^{-4} that of the spent resin tank.

12.2.1.10 Sources in the Spent Fuel Pool Cleanup System

The design basis source term for the equipment in this system was derived assuming mixing of the water in the spent fuel pool with the water in the containment refueling pool during fuel transfer. Shielding calculations use cylindrical approximations of equipment geometries. Dimensions for the components in this system are given in Table 12.2-18. The location of the equipment in the Spent Fuel Pool Cleanup System is shown in Figure 12.3-5.

a. Spent Fuel Pool Demineralizer

Shielding for this demineralizer is based on the peak activity level calculated for accumulation and decay of isotopes following initiation of the cleanup loop. This calculation includes a four-day cleanup period of the reactor coolant followed by dilution with refueling water before mixing of reactor cavity and spent fuel pool water. No credit was taken for the diluting effect of the spent fuel pool water. Spent fuel pool cleanup flow rate is 120 gpm. Demineralizer efficiency is assumed to be 100 percent for all isotopes. The design source term for the spent fuel pool demineralizer is given in Table 12.2-17.

b. Spent Fuel Pool Demineralizer Prefilter and Spent Fuel Pool Demineralizer Post-filter

Shielding for these filters is based on 100 rem/hr contact dose at change-out.

12.2.1.11 Sources in the Miscellaneous Chemical Drain System

The input to this system is primarily from the drains in the chemistry lab. The shielding source term was developed from expected sample volumes and frequencies. Composition of the input is shown in Table 12.2-19. Geometry of system components is given in Table 12.2-20. Location of equipment in this system is shown in Figure 12.3-8 and Figure 12.3-17.

a. Chemical Drain Tank

The shielding source term for this tank is based on the system input concentrations at an average input flow rate of 103 gpd. Consideration is given to buildup and decay during the filling of this tank. The design inventory of a full chemical drain tank is given in Table 12.2-19.

b. Chemical Drain Transfer Pump

This pump is used to transfer the contents of the chemical drain tank. Shielding is therefore based on the same source term.

c. Chemical Drain Treatment Tank

The major source of fluid for this tank is the chemical drain tank. The source term is based on buildup and decay during the filling of this tank at a rate of 103 gallons per day.

12.2.1.12 Source Terms for Water Storage Tanks

The isotopic inventory of the water storage tanks is given in Table 12.2-21 and Table 12.2-22. The location of these tanks is shown in Figure 12.3-10. The physical geometry of these components is described in Table 12.2-23.

a. Reactor Makeup Water Storage Tank

The inventory of this tank is based on 1 percent fuel clad defects and reactor coolant recycled by the Boron Recovery System.

b. Refueling Water Storage Tank

The inventory given in Table 12.2-22 is based on the refueling water storage tank refilled with cavity flood water immediately following a refueling. The reactor coolant radionuclide concentrations prior to the refueling preparation correspond to 1 percent failed fuel and the duration of the cavity flood condition is assumed to be 20 days. No credit for other decay is assumed.

12.2.1.13 Source Terms for Liquid Waste System

The source terms used in shielding calculations for the equipment in the Liquid Waste System are shown in Table 12.2-24. Each component is represented by a cylinder approximating its actual geometry. The dimensions of the cylinders are shown in Table 12.2-25. The location of this equipment is shown in Figure 12.3-8, Figure 12.3-9, and Figure 12.3-10.

a. Floor Drain Tank

The concentrations of radionuclides that form the shielding source term for the floor drain tank are given in Table 12.2-24. The source term is developed by buildup and decay calculations using an input activity of 0.075 reactor coolant at a flow rate of 120 gpm. The floor drain tank has a capacity of 10,000 gallons.

b. Floor Drain Tank Pumps

Since the floor drain tank pumps serve to recirculate the water contained in the floor drain tanks, they may, at times, contain some reactor coolant without full dilution. Therefore the shielding is based on an assumed activity of 0.1 reactor coolant.

c. Liquid Waste Evaporator

The source term for shielding the liquid waste evaporator bottoms is calculated with consideration of buildup and decay using the following parameters:

1. The input activity is the same as the floor drain tank concentrations.
2. Feed flow rate is 25 gpm.
3. Concentration time is 100 hours.

The source term for the evaporator distillate is based on a feed flow rate of 25 gpm and decontamination factors of 10^3 for iodine, 10^4 for other fission and activation products and 10^0 for noble gases.

d. Liquid Waste Evaporator Distillate Condenser

The liquid waste evaporator distillate condenser contains both vapor phase distillate and liquid phase condensate. The distillate source term is the same as that for the liquid waste evaporator, while the condensate does not contain the noble gases.

e. Liquid Waste Evaporator Distillate Accumulator, Pump and Cooler

The source term for these components is the same as that for the condensate in the liquid waste evaporator distillate condenser.

f. Waste Test Tanks

The source term for the waste test tanks (WTT) are evaluated by calculating the buildup and decay of isotopes, given the input activity of evaporator distillate condensate at a rate of 25 gpm. Each WTT has a capacity of 25,000 gallons.

g. Waste Test Tank Pumps

The WTT pumps are used at times for WTT recirculation. These pumps may then contain liquid with the activity of the WTT feed, which is the same as the liquid waste evaporator distillate condenser condensate.

h. Waste Demineralizer

Calculation of the source term for the waste demineralizer considers buildup and decay of radioisotopes over a one-year service period. The input activity is the same as the WTT concentration at a flow rate of 2,750 gpd. This demineralizer is assumed to have a 100 percent efficiency for removal of solids.

i. Liquid Waste Evaporator Reboiler, Reboiler Pump, Bottoms Pump and Bottoms Cooler

The source terms for this equipment are the same as that for the liquid waste evaporator bottoms described in Subsection 12.2.1.13c.

j. Liquid Waste System Filters

All filter cartridges will be changed when the dose rate reaches 100 rem/hr on contact. This value forms the basis for shielding calculations.

k. Skid-Mounted Waste Liquid Processing System

The skid-mounted system is designed and changeout criteria will maintain the equipment room area outside the shield wall at ≤ a zone III Radiation Area (≤ 15 mr/hr).

12.2.1.14 Source Terms for Radioactive Gaseous Waste System (RGWS)

There are three major sources of activity which serve as input to the Radioactive Gas Waste System: the letdown degasifier (Units 1 and 2) and the primary drain tank degasifier. Each degasifier produces effluent at a rate of 0.25 scfm; therefore the shielding source terms for the system are based on a gas flow rate of 0.75 scfm and are shown in Table 12.2-26 and Table 12.2-27. The RGWS also processes gas from the hydrogenated vent headers; however, this gas would dilute the concentrations of radionuclides and has not been included in the system input. The geometry of RGWS components is described in Table 12.2-28. Equipment locations are shown in Figure 12.3-12 and Figure 12.3-13.

a. Waste Gas Chillers

The concentrations in the waste gas chillers are the same as the system input. The inventory of isotopes for these components is given in Table 12.2-27.

b. Waste Gas Dryers

The specific isotopic shielding source term for these components is assumed to be the same as that for the waste gas chillers, i.e., taking no credit for iodine removal by the iodine guard beds. Each dryer contains 29 percent aluminosilicate by volume. The calculated isotopic inventory is presented in Table 12.2-27.

c. Iodine Guard Beds

The equilibrium accumulation of iodines at a gas flow of 0.75 scfm is used for the shielding source term for these components. The isotopic inventory used for shielding calculations is presented in Table 12.2-27.

d. Carbon Delay Beds

The source term for these components is based on a delay per bed of 12 days for Xenon and 17 hours for Krypton at a flow rate of 0.75 scfm. The shielding inventory for each of the beds is given in Table 12.2-27.

e. Hydrogen Surge Tank

Shielding for this tank is based on the inventory of isotopes given in Table 12.2-27. This source term assumes that the tank is filled with gas at the specific activity calculated for the outlet of the fifth carbon delay bed at tank design pressure.

f. Regenerative Compressor

Shielding for this component is based on the same specific source term as for the waste gas dryers, given in Table 12.2-27.

g. HEPA Filter

The source term for this filter was calculated on the assumption that one percent of the inventory in the last carbon delay bed is transported with charcoal fines to the filter cartridge. No credit is taken for decay of isotopes after leaving the carbon delay bed. The shielding design inventory is given in Table 12.2-27.

12.2.1.15 Source Terms for Solid Waste Management System

The source terms used for shielding calculations for the equipment in the Solid Waste Management System are listed in Table 12.2-29; the shielding geometry and dimensions for this equipment are presented in Table 12.2-30. The locations of this equipment are shown in Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, and Figure 12.3-12.

a. Waste Concentrates Tank, Waste Concentrates Transfer Pump

The waste concentrates tank has a working capacity of 6000 gallons, and can hold roughly four batches of evaporator bottoms. The shielding source terms are based upon two batches of bottoms from the boron recovery evaporator, and one batch each from the liquid waste evaporator and the steam generator blowdown evaporator. No radioactive decay during transit has been considered.

b. Waste Feed Tanks and Waste Feed Recirculation Pumps

The input stream to these tanks contains about 12 percent solids by weight; after preparation/processing in the tanks, the output stream contains about 10.4 percent solids by weight. The shielding source terms for this equipment are about 0.87 of those for the waste concentrates tank described in a. above.

c. Spent Resin Transfer Pump, Spent Resin Hopper, Spent Resin Recirculation Pump, and Resin Centrifuge Metering Pump

The shielding source terms for this equipment are the same as those for the spent resin sluice tank described in Subsection 12.2.1.9.

d. Spent Resin Centrifuge

The input resin slurry contains about 15 percent of resin by weight; after the dewatering process, the remaining contents at the discharge from the centrifuge are about 50 percent resin by weight (with no transport water). Therefore, the shielding source terms for the spent resin centrifuge are 3.3 times those for the spent resin hopper described in c. above.

e. Waste Crystallizer/Evaporator, Crystallizer Recirculation Pump, and Crystallizer Drain Pump

The input waste concentrates contains about 10.4 percent solids by weight; after crystallizer/evaporation, the remaining slurry is about 35-50 percent total dissolved solids by weight. Therefore, the shielding source terms for this equipment are 2.9 times those for the waste feed tanks described in b.

f. Concentrates Bottom Tank, Concentrates Bottom Tank Recirculation Pump, Waste Metering Pump, and Alternate Station Concentrates Feed Pump

The shielding source terms for this equipment are the same as those for the waste crystallizer/evaporator described in e. above.

g. Entrainment Separator, Crystallizer Condenser (Shell Side), Crystallizer Distillate Tank, Crystallizer Distillate Pumps, and Crystallizer Subcooler (Shell Side)

This equipment is not expected to contain significant radioactivity; the fluids are expected to contain about 1 ppm of solids by weight. The shielding source terms for the equipment are about 8.3×10^{-6} of that for the waste concentrates tank described in a.

h. Spent Resin Dewatering Pump

This equipment is not expected to carry significant radioactivity. For conservatism, and consistency with the guidance given by NUREG-0017, the shielding source terms for this equipment are taken to be the same as those for the resin sluice pump described in Subsection 12.2.1.9.

i. Crystallizer Reflux Pot and Crystallizer Reflux Pump

This equipment is not expected to contain significant radioactivity. The shielding source terms for this equipment is taken to be the same as those for the crystallizer distillate tank described in g.

j. Extruder

In the extruder, spent resins or crystallizer bottoms are mixed with the asphalt binder in a one-to-one ratio by weight. The shielding source terms for this equipment are 0.5 times those for the spent resin centrifuge described in d.

12.2.2 Airborne Radioactive Material Sources

The major source of airborne contamination during normal operation is leakage of radioactive fluid from equipment and valves. Other sources from anticipated occurrences include opening of sealed equipment and evaporation during fuel handling. Accident sources such as DBA, fuel handling accident, and radwaste system failures are discussed in Chapter 15.

12.2.2.1 Design Basis

The ventilation system was designed to maintain the normal airborne radioactivity concentration to levels below the applicable occupational concentration values listed in 10 CFR 20 for air. On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

Air flow is directed from areas of high occupancy and low airborne radioactive concentrations to areas of increasing contamination and lower occupancy requirements. Design basis maximum leakage rates to various buildings and compartments were chosen in accordance with NUREG-0017, as referenced by Regulatory Guide 1.112. Specific isotopic activity of the fluid leakage is based on 0.25 percent fuel clad defects, as discussed in Subsection 11.1.1.

12.2.2.2 Leakage Sources

a. Containment

The design basis maximum daily leakage to the Containment, as recommended in NUREG-0017, is 1 percent of the reactor coolant noble gas inventory and 0.001 percent of the iodine inventory. Reactor coolant isotopic concentrations for 0.25 percent failed fuel are listed in Table 11.1-1. The containment ventilation purge systems are designed to reduce airborne concentrations to acceptable levels within 20 hours after shutdown. Specific airborne concentrations in the Containment at shutdown and following purging are presented in Table 12.2-31 and Table 12.2-32.

b. Turbine Building

Calculating maximum airborne contamination in the Turbine Building assumes the leakage of 1700 pounds per hour of steam with the specific activities described in Table 11.1-4. Building volume, ventilation rates for summer and winter, and airborne isotopic concentrations are listed in Table 12.2-33 and Table 12.2-34.

c. Auxiliary Buildings

Several areas in the Primary Auxiliary Building have a concentration of piping and valves with a potential for significant contributions to airborne contamination. None of these areas requires continuous occupancy. The design basis maximum leakage rate to the Primary Auxiliary Building, per NUREG-0017, is 20 gpd. The source for these areas is reactor coolant with 0.25 percent fuel clad defects, as presented in Table 11.1-1. Airborne concentrations are presented in Table 12.2-35 for the average contaminated area.

12.2.2.3 Movement of Spent Fuel

The most significant contribution to airborne radioactive contamination during the movement of spent fuel is from tritium released by evaporation from the surface of the spent fuel pool and the refueling canal. The design concentration of tritium in the spent fuel pool and the refueling canal is 0.63 microcuries per milliliter. Tritium concentration in the water is controlled by release; in liquid form through the Boron Recovery System and in gaseous form via the ventilation system and the plant vent.

Airborne concentration is controlled by the ventilation system. Design air flow rates and tritium concentration are presented in Table 12.2-36 and Table 12.2-37. Evaporation rates of 270 lb/hr for the refueling canal (140°F) and 378 lb/hr for the spent fuel pool (175°F), which were used in the design calculations, assumed an air temperature of 50°F and a relative humidity of 20 percent.

12.2.2.4 Reactor Vessel Head Removal

The potential for release of airborne contamination due to reactor vessel head removal is significantly affected, and to a large degree controlled, by operating procedures and timing following shutdown. First, the reactor coolant is diluted with water from the reactor makeup water storage tank and/or the boric acid tanks via the Chemical and Volume Control System during cooldown. Then, there is a time lag between shutdown and vessel head lift (minimum 4 days) which allows decay. Letdown purification and degasification also continue during this time interval. Finally, the reactor coolant in the vessel is diluted by water from the refueling water storage tank by way of the Residual Heat Removal System. If necessary, portable fans can be used to mix the air in the vessel head area with the general containment air circulation to take advantage of the 31,000 cfm refueling purge flow rate.

12.2.2.5 Relief Valve Venting

Relief valves with significant potential for contribution to airborne contamination do not vent to building atmosphere. The relief valves on tanks and piping containing nondegasified fluids discharge to the primary drain tank. The primary drain tank, in turn, vents to the hydrogenated vent header for processing through the Radioactive Gas Waste System prior to environmental release.

12.2.2.6 Calculation Models and Parameters

Radionuclide input rates to building atmospheres were determined using assumed leakage rates of radioactive fluid to in-plant areas and corresponding specific activities and partition factors.

Calculations of the various airborne isotopic concentrations of interest were performed using the UE&C computer code HDOSE. The code determines isotopic concentrations from specific radionuclide input and removal processes, and allows simulation of specified, pre-determined time behavior for such processes.

In performing these calculations, credit was taken for the following removal mechanisms:

a. Natural radioactive decay

b. Recirculation - air is taken from building atmosphere, filtered, and returned to the building

c. Purge - outside air is drawn into the building and interior air is expelled to the outside.

(Recirculation and purge proceed with a mixing efficiency of 70 percent.)

Also taken into account was the formation of airborne radionuclides from:

a. The decay of airborne parent species

b. Trapped filterable parent species which decay to unfilterable daughters (e.g., iodine to xenon).

The rate equation used in HDOSE to describe the activity of isotope "j" in a compartment is given by:

$$\frac{dNc,j}{dt} = R_j + \sum_{k=1,2} B_{k,j}\,\lambda_k\,N_{c,k} + gj\sum_{k=1,2} B_{k,j}\,\lambda_k\,N_{r,k} - \left(\lambda_j + L_r\,e_j + L_e\right) N_{c,j}$$

where:

Nc, j = activity of isotope "j" in the compartment

R_j = release rate of isotope "j" into the compartment (Ci/hr)

$_j$ = radiological decay constant of isotope "j"

$_k$ = radiological decay constant of isotope "k"

$B_{k,j}$ = branching ratio of decay from isotope "k" to isotope "j"
(k=1 indicates parent of "j", k=2 indicates grandparent)

$N_{r,k}$ = accumulation of nuclide "k" on recirculation filter

g_j = 1 for gaseous isotope "j"
0 for nongaseous isotope "j"

e_j = recirculation cleanup filter efficiency for isotope "j"

L_e = compartment leak rate or purge rate

L_r = filtered recirculation flow rate.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The radiation protection philosophy for the design of Seabrook Station is to restrict radiation exposure to plant personnel and the general public to within the limits of 10 CFR 20 and 10 CFR 50, while ensuring high flexibility and availability within the power generation and safety objectives of plant operation.

This philosophy can be summarized in several basic design goals intended to minimize exposures:

a. To minimize, to the extent possible, the production of radioactive isotopes, e.g., the steps taken to reduce crud production described in Subsection 12.2.1.3.

b. To minimize the extent of areas housing radioactive equipment and piping through efficient arrangement of equipment and systems.

c. To shield the normally occupied areas from radiation.

d. To minimize exposures within high radiation areas by, first, controlling access to those areas, and secondly, through design of systems and equipment for reliability and ease of maintenance.

The plant is designed to permit periodic online equipment inspection and maintenance, radioactive material handling, decontamination and cleanup, and access to vital plant areas during normal plant operation, including anticipated operational occurrences. Postulated accidents are also considered in the determination of radiation exposure of Engineered Safety Features and other materials. In addition, these accidents are evaluated for access to, and habitability of, the control room, including ingress and egress, for the duration of the accident.

Offsite radiation exposures following postulated accidents are discussed in Chapter 15; those from processed radioactive material releases during normal operation are discussed in Chapter 11.

The primary objective of plant shielding is to provide for the protection and safety of all plant personnel and the general public under all normal and anticipated abnormal plant operating conditions. Reactor shielding, along with the radiation monitoring system and access control procedures, supplemented by periodic radiation surveys and radiochemical analysis, ensure that radiation exposures of the general public and plant personnel do not exceed the limits set by the federal regulatory agencies. The maximum allowable design dose rates for all plant areas, in conjunction with anticipated occupancy, limit the integrated whole body dose to less than 5 rem per calendar year. All areas that house radioactive materials are appropriately marked in accordance with Part 20 of Title 10 of the Code of Federal Regulations (10 CFR 20).

Plant operating personnel and the general public are protected by radiation shielding wherever a potential radiation source may exist. The shielding design philosophy embodies the following objectives:

a. To restrict the potential radiation dose to operating personnel during normal operation to within the limits of 10 CFR 20.

b. To adequately protect the operating personnel in the unlikely event of an accident, in order to allow termination of accident conditions and mitigation of the consequences without undue risk to the general public.

c. To protect equipment from excessive radiation exposure to prevent malfunctions due to radiation-induced failures.

d. To maintain the radiation exposure of the general public from normal operation to within the limits of 10 CFR 20.

The guidance provided by Regulatory Guide 8.8 has been utilized extensively in the plant radiation protection design philosophy, as described in the following pages of this section.

The radiation sources and associated input parameters, assumptions, and methodology described in Section 12.2 represent those used to establish original shielding design. The radiation sources utilized for the design of the primary shield were developed by the reactor vendor and are based on a standard four loop Pressurized Water Reactor. The radiation sources in the spent fuel and the N-16 source were also developed by the reactor vendor, and are based on a reactor power of 3565 MWt. The radiation sources in the primary coolant system and the auxiliary systems are based on a reactor power of 3654 MWt, a one-year fuel cycle, and 1 percent fuel element defects.

The impact on plant shielding requirements was evaluated for an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle. This represents a minor change from the original design basis. The conservative analytical techniques used to establish the original shielding requirements, and the Station Technical Specification which will restrict the reactor coolant activity to levels significantly less than the 1 percent fuel defects, ensure that operation at the licensed core power level will have no significant impact on shielding requirements and safe plant operation.

12.3.1.1 Radiation Zones and Access Control

The shielding design bases for work areas are a combination of the design radiation level and anticipated occupancy times. The plant is divided into zones dependent upon the intensity of radiation within the given area. Areas within these zones are posted in accordance with the regulations of 10 CFR 20. Occupied areas within a zone are limited to the same radiation ranges as prescribed for that zone. Zone classifications are presented in Table 12.3-1 and Table 12.3-2.

Zone boundaries, decontamination facilities and location of radiation monitors are shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, and Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16, and Figure 12.3-17. The arrangement of the chemistry lab, health physics facilities and counting room is shown in Figure 12.3-17. Shield wall thicknesses for all major sources of radiation are given in Table 12.3-3, Table 12.3-4, Table 12.3-5, Table 12.3-6, Table 12.3-7, Table 12.3-8, Table 12.3-9, Table 12.3-10, Table 12.3-11, Table 12.3-12, and Table 12.3-13.

Access control points are also shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16, and Figure 12.3-17, while discussion of access control and its implementation can be found in Section 12.5. The entire Containment Building is a controlled access area.

The turbine generator structure areas, administrative offices, turbine plant service areas and the control room are designated Zone I. Areas such as the local control space in the Primary Auxiliary Building, the waste disposal area, and the operating deck of the spent fuel storage area, are generally designated Zone II. Intermittently occupied work areas, such as valve galleries, are designated Zone III. Typical Zone IV areas include steam generator compartment areas (after reactor shutdown) and areas outside of containment pipe penetrations. A typical Zone V is the volume control tank area. Certain areas of the Containment are accessible for a limited time during normal plant operation.

The radiation counting room is designated Zone I (less than 0.5 mrem/hr); however, sufficient shielding is provided to assure that the background dose rate is low enough (less than 0.1 mrem/hr) to permit accurate operation of counting equipment.

Shielding has been designed by identifying source strengths within an area and then providing sufficient shielding to achieve the specified dose rate in adjacent zones. The source strengths are based on 1 percent failed fuel and maximum expected activation product levels, so that the actual radiation levels experienced within the station are expected to be less than the design values. Concrete shield thickness was in most cases determined by rounding from the calculated required thickness to the next higher 6-inch increment. In a few cases, where space was limited, the next 3-inch increment was used.

12.3.1.2 Handling of Nuclear Materials

Most systems included in the Primary Auxiliary and Waste Processing Buildings are used to process the radioactive byproducts produced in, and which leak from, the Reactor Coolant System during normal power operation. The design of the systems in the letdown purification and general waste processing systems reflects the concept of minimizing the exposure of plant personnel.

New fuel handling is discussed in Subsection 9.1.4.

Storage and handling of radioactive sources is discussed in Subsection 12.5.3.7.

12.3.1.3 Shielding and Layout Features

a. Plant Layout

Plant layout includes optimal location of radioactive components. The most radioactive systems are located toward the interior and on the lower plant levels, with less radioactive systems located toward the outside.

The plant layout provides for personnel access which maintains occupational doses as low as reasonably achievable (ALARA). Passage through a higher radiation area to obtain access to a lower radiation area is avoided, thus minimizing unnecessary accumulation of occupational exposure.

Wherever practical, components are shielded individually to keep exposure ALARA during maintenance periods. Shielded pipe chases are utilized extensively to segregate radioactive piping from normal occupied areas. Reach rods are used where required to place the operator in a low radiation area during valve operation. Auxiliary control boards are located with ALARA occupational exposure in mind.

When sources of sufficient strength are present, labyrinth entrances are utilized to minimize the contribution in the walkways. One or two scatter labyrinths are used as required by scattering calculations.

Periodic review of plant design and equipment arrangement aimed at maintaining occupational radiation exposures (ORE) ALARA resulted in a number of design changes, as illustrated in the examples below:

1. Access to the stairways in the RHR Vaults may be restricted below elevation 3'-2" due to equipment and pipe shine.
2. Shielding above the demineralizers in the Primary Auxiliary and Waste Processing Buildings was increased.
3. Primary sample heat exchanger and sink room was rearranged to reduce shine.
4. Evaporator equipment was rearranged to minimize radiation levels in adjacent walkways.
5. Areas of potentially excessive radiation levels were identified and space for possible future shielding was reserved.

b. Equipment Layout

The criteria for the arrangement of equipment containing radioactive sources were developed specifically for maintaining occupational doses ALARA and for ease of maintenance.

1. Filters (Liquid)

(a) Each potentially radioactive filter is located inside an individual shielded compartment. This minimizes the contribution to radiation levels from adjacent filters during maintenance periods. One exception to this are the vendor supplied waste liquid processing system filters. These filters may be in a common shielded compartment with demineralizer vessels, or outside a shielded area if the expected dose is Zone III or lower.

(b) For PAB and WPB filters, where changeouts are required, adequate space is provided for use of the remote filter handling cask in removing the filter (lateral room to swing the vessel head clear, and head room for lifting the cartridge), loading the filter into the cask, and transportation to the solid waste area. Attention is given to ensure that there are no interferences or obstructions in the path. The shield wall for waste liquid processing equipment is vendor supplied for the vendor system. This ensures interferences or obstructions in filter/resin manipulations are accounted for.

(c) All valves and instrumentation associated with filters are located outside the compartment. Normally, filter process valves are operated by remote manual mechanical linkage which extends to a low radiation zone, to minimize operator exposure during normal operation.

Where practical, filters are located near the solid waste area to minimize the chance of spillage in transit.

2. Demineralizers

The primary means of processing radioactive water to be discharged is through a vendor-supplied system. This system meets the intent of the design requirements in Regulatory Guide 1.143 and NUREG 0800. (See Section 1.8)

Other aspects of radioactivity processing systems are:

(a) Capability is provided to remotely remove all primary- side resins by flushing. All initially installed resin transport lines are designed to avoid resin traps. Butt welds instead of socket welds and five diameter bends instead of fittings are used in downward-sloping pipe without any sections horizontal or upward-sloping. Procedural controls are in place to prevent normal access during times of elevated dose rates.

(b) The resin fill line is shaped to prevent direct radiation streaming.

(c) Each potentially radioactive primary-side demineralizer in the Primary Auxiliary Building and Waste Processing Building is located inside shielded compartments or cubicles in order to reduce shielding problems during maintenance periods.

(d) All valves and instrumentation associated with these demineralizers are located outside the compartment. All demineralizer process valves, except for the vendor-supplied system, are operated by remote manual mechanical linkage which extends to a low radiation zone to minimize operator exposure during normal operation. The vendor-supplied system is designed to process water from floor drain tanks. The vendor-supplied components which concentrate radionuclides are all shielded to reduce any potentially elevated dose rates.

(e) Only the required process lines enter into, or pass through, the demineralizer cubicles, as access to the cubicles during normal operation will be strictly controlled.

(f) The demineralizers associated with the Steam Generator Blowdown System are not located in individual shielded cubicles, and are not provided with the capability for remote sluicing. These units normally treat secondary coolant with only minor or no contamination present. Processing blowdown through these demineralizers is based on maintaining the general area near the vessels as a radiation Zone II (<2 mr/hr). If significant primary-to-secondary leakage occurs, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system (WL-SKD-135) to the waste test tanks is the preferred method (reference Subsection 11.2.2.1). In addition blowdown processing may be through the blowdown evaporators in place of the Blowdown Demineralizer System.

3. Adsorber Beds

(a) The first and second waste gas adsorber beds are located in individual shielded compartments.

(b) The third-through-fifth beds are located in a common compartment with the most active bed farthest from the entrance.

(c) All valves are located outside the bed cubicles, and can be operated from a low radiation area.

4. Recombiners

(a) Post-accident recombiners are located inside the Containment Building, and are designed for operation in the accident environment.

(b) Shielding is provided by the 4½-foot-thick containment walls.

5. Tanks

(a) Tank overflow lines are connected to prevent spillage on the floor and to prevent dissolved gases from escaping the tank. In limited cases the overflow lines are directed toward floor drains.

(b) Controlled ventilation is provided for tanks containing aerated or hydrogenated fluids.

(c) Tanks located outside of heated buildings are protected from freezing by steam heating panels.

(d) Manual valves are located in, or have handwheel extensions to, low radiation zones.

6. Evaporators

(a) Shielding is provided for individual evaporator units.

(b) Instruments, valves in service lines, and evaporator sample points are located in low radiation zones.

(c) Connections are provided for flushing and draining of pipe and equipment prior to maintenance.

7. Pumps

(a) In order to perform major maintenance in as low a radiation zone as reasonably achievable, the pump or motor can be decontaminated, if necessary, and moved to a low radiation area. Temporary local shielding may also be used.

(b) Pumps are designed with double mechanical seals to give a minimum leakage of radioactive fluid.

(c) Remote instrumentation and switching is provided, as required.

8. Steam Generators

The portions of the steam generators containing reactor coolant are shielded by the 4-foot-thick secondary shield walls.

9. Sampling Station

(a) The sample sink room is separated from the sample heat exchangers by a shield wall.

(b) Sample rates are limited to 1.5 gallons per minute by design.

(c) The sample hood is ventilated to prevent the accumulation of gases.

(d) The sample system is designed for a closed system line purge prior to sampling.

(e) Shielding is provided at local sampling points, as required.

10. Penetrations

    (a) Where possible, penetrations through shield walls are offset from line-of-sight of the source.

    (b) Where necessary, the annulus between a pipe and its sleeve is packed with lead wool or lead-silicone foam with a nominal density of 150 lb/ft^3.

11. Instrumentation

    (a) Drains from instrument blowdowns are routed to radioactive drains.

    (b) Diaphragm seals or clean water seal legs are used, wherever practical, to minimize the volume or radioactive fluids entering low radiation areas via instrument impulse lines.

    (c) Radioactive gas and liquid samples are returned to process lines wherever practical.

    (d) Wherever practical, instruments are located in low radiation zones to permit extended access for calibration and testing.

12. Piping and Valves

    In order to minimize concentrated pockets of crud in permanently installed radioactive systems:

    (a) Piping 2½" and larger is butt-welded.

    (b) Spent resin sluicing lines utilize five diameter bends to minimize the number of fittings.

    (c) Valves are selected to avoid crud pockets.

    (d) Piping layout avoids pockets wherever possible.

12.3.2 Shielding

The material most commonly employed for shielding is concrete. Where space is limited, steel or lead is substituted for ordinary concrete in equivalent thicknesses. Whenever cast-in-place concrete is replaced by concrete blocks (removable or fixed), the design assures protection on an equivalent shielding basis.

Analytical models were selected according to the source geometry under consideration. Tanks, vessels, and large pipe-sections containing radioactive materials were shielded by considering uniform cylindrical volumetric sources. Appropriate line source approximations were used for small pipes and tall vessels with small diameters. For a conservative estimate of dose rate, sources of irregular geometry were modeled by a point source of strength equivalent to the volumetric source.

The following techniques, codes, models, and assumptions were used:

- Point kernel integration methods were used.
- Buildup factors were accounted for inside the integrals.
- Self-shielding was taken into consideration.
- Concrete density was assumed to be 2.35 gm/cc.
- The maximum calculated shield thickness was specified for each component or radiation area.
- Source term data corresponds to 1 percent failed fuel with a power level of 3654 MWt.

Special protective design features to ensure that occupational radiation exposures will be ALARA are described in Subsection 12.3.1.3. The guidance given in Regulatory Guide 8.8 can be seen in these features.

Compliance with Regulatory Guide 1.69 is addressed in Section 1.8.

12.3.2.1 Shield Configurations

a. Reactor Shielding

1. Primary Shield

The primary shield is a large mass of reinforced concrete, 7½ feet thick at core midplane, that surrounds the reactor vessel and extends upward from the containment floor to form the walls of the refueling cavity. The primary shield is designed to:

(a) Reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and Reactor Coolant System, and allow limited access to the Containment during normal operation.

(b) Limit the radiation level after shutdown from sources within the vessel, and permit limited access to the areas containing reactor coolant system equipment.

(c) Limit neutron flux activation of component and structural materials over the life of the plant.

2. Secondary Shield

The secondary shield is a reinforced-concrete structure that surrounds the reactor coolant equipment, pipes, pumps and steam generators. This shield protects personnel from gamma radiation emanating from reactor coolant activation products and fission products that are transported from the core by the reactor coolant. The neutrons emitted by the decay of carried-over N-17 are effectively eliminated by the concrete shielding employed for N-16. The secondary shield also supplements the primary shield function of attenuating direct core radiation. In addition, it permits limited access to the Containment during normal operation so that inspection of essential equipment may be accomplished without requiring plant shutdown.

3. Neutron Shield

In order to reduce dose rates and equipment activation on the containment operating floor during power operation, a supplementary shield has been designed to minimize the streaming of neutrons from the reactor cavity. A neutron shield consisting of Reactor Experiments Type 277 borated concrete, and which is integral to the permanent reactor cavity seal ring, is installed around the reactor vessel refueling flange. The neutron shield is suspended from the permanent seal ring and fills the annular area between reactor vessel refueling flange and the cavity wall. The neutron shield is approximately fourteen inches thick and supported from the bottom by a one-inch thick steel plate. Sectional view of the permanent reactor cavity seal ring/neutron shield is shown in Figure 6.2-26.

b. Containment Shielding

The containment shielding is a steel-lined, reinforced-concrete containment structure that completely surrounds the reactor building equipment. At full power operation, this shield attenuates the radiation level outside the primary-secondary shield complex, including radiation sources which become airborne during normal operation due to primary system leakage, to ensure that radiation levels outside the Containment are less that 0.5 mrem/hr. The containment structure also shields against radiation sources inside the Containment due to fission products released following postulated accidents. The integrated direct dose is less than 260 mrem immediately outside the Containment over a period of two hours after the design basis accident (DBA), which will permit access to such vital areas as the control room. The containment wall and dome are 4½-feet and 3½-feet thick, respectively.

c. Spent Fuel Shielding

This shielding provides protection during all phases of spent fuel removal and storage. Operations that require shielding of personnel are spent fuel removal from the reactor, spent fuel transfer through the refueling canal and transfer tube, spent fuel storage, and spent fuel shipping cask loading prior to transportation. All spent fuel removal and transfer operations are performed under borated water to provide radiation protection.

All accessible areas around the tube and canal are shielded. All shields were designed for a contact radiation dose rate of less than 100 mr/hr. Four inches of lead plate were added between the liner and concrete at the bottom of the canal. In the Enclosure Building a shield box was designed around the tube. This box consists of approximately 300 bricks weighing 50 pounds each. These bricks will be explicitly marked with a sign stating that potentially lethal radiation fields are possible if the bricks are removed during fuel transfer. The access point noted in Figure 1.2-3 is an inspection hatch (manway). This hatch is shielded with a three-foot concrete plug. This plug shall also be marked as noted above.

Minimum allowable water depth above a fuel assembly during fuel handling is 10 feet in the reactor cavity. This limits the dose at the water surface to less than 10.0 mrem/hr for an assembly in a vertical position. The minimum water depth in the spent fuel pool is 13 feet above the top of the fuel assemblies in the storage racks. For this depth, the dose rate at the water surface is less than 2.5 mrem/hr. Normal water depth above the stored assemblies is about 25 feet.

The 5-foot thick concrete walls of the fuel transfer canal and the 6-foot thick spent fuel pool walls supplement the water shielding, and limit the radiation dose levels in most working areas to less than 2.5 mrem/hr, and a maximum dose in some areas less than 100 mrem/hr.

The refueling water and concrete walls also shield personnel from activated control rod clusters and reactor internals that are removed at refueling times. Dose rates are generally less than 2.5 mrem/hr in working areas. However, certain manipulations of fuel assemblies, control rod clusters, or reactor internals may produce short-term dose rate levels in excess of 2.5 mrem/hr. Radiation levels in the working areas will be closely monitored during refueling operations to ensure that exposures for plant personnel do not exceed the integrated doses specified in 10 CFR 20.

d. Control Room Shielding

The control room shielding is designed in accordance with applicable regulations, to permit continuous occupancy by control room personnel following a DBA. This enables control room operators to maintain full control and to shut down the plant without personal hazard. The control room shielding is 2-feet thick, based upon an integrated dose during the 30 days following the DBA which does not exceed 5 rems whole body, or its equivalent to any part of the body, as required by General Design Criterion 19 of 10 CFR 50, Appendix A. A layout drawing of the control room is shown in Figure 1.2-32.

e. Plant Auxiliary Systems Shielding

Auxiliary shielding includes all concrete walls, covers and removable blocks that shield the numerous radiation sources in the radioactive waste disposal, makeup and purification, and chemical addition and sampling systems. Typical components that require shielding include the volume control tank, thermal regeneration demineralizers, waste drumming area and reactor coolant system drain tank. Shield wall thicknesses for components in auxiliary systems are given in Table 12.3-3, Table 12.3-4, Table 12.3-5, Table 12.3-6, Table 12.3-7, Table 12.3-8, Table 12.3-9, Table 12.3-10, Table 12.3-11, Table 12.3-12, and Table 12.3-13.

f. Turbine Shielding

The radioactive material inventory in the Turbine Building is very small since only secondary steam enters the area with small amounts of primary coolant leakage. Shielding is not, therefore, a major concern here, with wall and floor thicknesses determined from structural considerations.

g. General Plant Yard Areas

All shielding is designed so that the dose rates in plant yard areas which are frequently occupied by plant personnel remain below 0.5 mr/hr. These areas are surrounded by a security fence, and are closed off from areas accessible to the public for general safety.

12.3.2.2 Plant Shielding to Provide Access to Vital Locations for Post-Accident Operations

Following an accident, significant radioactivity may be released from the reactor core, presenting unusual hazards to operating personnel. A review was conducted to assess the projected amount of activity released, systems involved in transport of this activity, effect of the transported activity on plant dose rates and acceptability of dose rates in locations requiring access for necessary operations (vital locations).

This assessment employed core fission product release source terms consistent with NUREG-0737, Section II.B.2 (100% noble gas, 50% halogen, 1% other fission product). The assessment addressed both pressurized and depressurized accidents, and projected consequences of the release at post-accident times ranging from the onset of cold leg recirculation to 1 year. The assessment was based on an analyzed power level of 3565 MWt and a 1-year fuel cycle length.

The systems considered in this assessment included containment spray, chemical and volume control, safety injection, residual heat removal and combustible gas control.

Using the source term and system transport information described above, dose rates in various plant areas where projected. These projections considered shine, scatter and radiation streaming, including effectiveness of facility shielding. Levels in areas which must be accessed for operational tasks (vital locations) were tabulated, along with occupancy times, to verify projected exposures are within applicable limits. Such locations included the control room, technical support center, post-accident sample station, chemistry laboratory, switch gear room, radwaste control station, radiation controlled area tunnels and hydrogen analyzer area. High dose-rate areas are graphically depicted on area zone maps of the plant. These maps will aid in projecting exposures for potential post-accident operations not explicitly identified in the vital location table. Results of these projections demonstrate projected exposures in vital locations are within the GDC-19 and NUREG-0737 (Item II.B.2) criteria.

The assessments described above were incorporated into the Post-Accident Dose Engineering Manual, which is used in planning for post-accident operations. Rationale for not including several areas noted in NUREG-0737 (Item II-B.2) is delineated in this manual. A copy of the manual was provided to the NRC. The information in this document will be factored into the overall post-accident response actions.

The impact of an analyzed core power level of 3659 MWt and operation with an 18-month fuel cycle was evaluated, and it was determined that the post-accident operator exposure will continue to remain within regulatory limits. The information from the Post-Accident Dose Engineering Manual, modified to reflect the licensed core power level, is factored into the overall post-accident response actions.

12.3.3 Ventilation

The station ventilation system has been designed to provide a maximum of safety and convenience for operating personnel, construction workers and site visitors working both within the station radiologically controlled area and in station buildings outside the radiologically controlled area during normal operating and anticipated operational occurrences. The potential exposure to onsite personnel and to members of the general public resulting from airborne radionuclides from station operation complies with 10 CFR Part 20 and 10 CFR Part 50, respectively.

12.3.3.1 Ventilation Design Bases

Descriptions of the ventilation systems for each building which can be expected to contain radioactive materials, including design bases, are contained in Section 9.4. Diagrams associated with the descriptions show equipment, air flow patterns, and expected flow rates for normal and emergency conditions.

A description of the ventilation systems for the control room complex is contained in Subsection 9.4.1, Figure 9.4-1, Figure 9.4-2, and Figure 9.4-3, and shows equipment, air flow patterns, and expected flow rates. Section 6.4 discusses the habitability and life support systems of the control room complex with respect to NRC General Design Criterion 19.

In each case, air flow has been directed from areas of low potential airborne radioactivity to areas of higher airborne radioactivity by exhausting from the areas of higher radioactivity. The ventilation rate for the areas of higher radioactivity was determined both from the ventilation rate required to remove equipment heat, piping and electrical losses and from the ventilation rate required to control the concentration of airborne radioactivity. The ventilation was designed to meet the exposure limits for airborne concentrations listed in 10 CFR 20. Maintenance of a negative pressure by the exhaust systems in the areas of higher radioactivity induces an air flow from corridors and operating areas preventing the exfiltration of airborne radioactivity to clean areas normally occupied by operating or maintenance personnel.

Testing and maintenance will be performed in accordance with the criteria presented in Regulatory Guide 1.52 for the safety-related filter systems.

Failure to meet these in-place testing criteria will necessitate the change-out of filters or adsorbers.

12.3.3.2 Provisions for Localized Ventilation

Provisions for localized ventilation during maintenance and refueling operations are provided to the extent practicable to reduce concentrations of airborne radioactivity in areas not normally occupied where maintenance of in-service inspection has to be performed.

12.3.3.3 Exhaust Filtration

The exhaust from each area which can be expected to contain significant airborne radioactivity is processed through HEPA or HEPA and carbon air cleaning systems before being discharged to the unit plant vent.

Air cleaning units are provided for the containment enclosure emergency exhaust, the fuel storage building emergency exhaust, the containment purge exhaust, the primary auxiliary building exhaust for areas with a potential for significant airborne radioactivity, and the waste processing building exhaust for areas with a potential for higher airborne radioactivity.

In addition, recirculation air cleaning units are provided for the main control rooms and the containment structures.

Descriptions of the air cleaning systems, including design bases for the containment enclosure emergency exhaust and the fuel storage building emergency systems, including the design bases for the remainder of the buildings, are contained in Section 9.4.

Compliance with Regulatory Guide 1.52, Revision 2, of the containment enclosure emergency air cleaning unit, the fuel storage building air cleaning unit, and the control room emergency filtration subsystem is detailed in Table 6.5-1, Table 6.5-2 and Table 6.5-3 respectively.

The containment purge exhaust air cleaning unit, the primary auxiliary building normal exhaust air cleaning unit and the waste processing building air cleaning unit are equipped with quick-release filter clamps to minimize exposure of maintenance personnel when changing prefilters, medium efficiency filters and HEPA filters. The remainder of the air cleaning units employ standard threaded filter clamping devices because they are not expected to be exposed to more than minimal quantities of radioactive particulates.

The containment purge exhaust air cleaning unit, the primary auxiliary building normal exhaust air cleaning unit, the fuel storage building emergency exhaust air cleaning units and the containment enclosure emergency exhaust air cleaning unit are provided with bulk fill adsorber beds and guard beds, where applicable. The carbon for the adsorber and guard beds is pneumatically removed and filled, which minimizes exposure of the maintenance personnel to contaminated carbon. The control room emergency recirculation air cleaning unit and the containment recirculation air cleaning unit employ tray-type carbon adsorbers. The waste processing building air cleaning unit has no adsorber bed.

Access from both sides of the air cleaning units for maintenance and changing of filters is provided for the containment purge exhaust air cleaning unit, the fuel storage building emergency exhaust air cleaning unit and the primary auxiliary building normal exhaust air cleaning unit. Aisle space and clear means of ingress and egress are provided for the handling of filters and carbon bed carbon removal/fill equipment.

A layout of the primary auxiliary building normal exhaust unit provided in Figure 12.3-18 is an example of the filter bank spacing and access for maintenance.

The radiation control area (RCA) of the Administration and Service Building is provided with a once-through ventilation system. The exhaust system maintains a negative pressure on the entire RCA portion of the building, preventing the exfiltration of airborne radioactivity to the clean areas. The exhaust air is processed through 55 percent medium efficiency filters and then through HEPA filters. There is no adsorber bed provided for this system. The air is directed within the RCA from areas of low potential airborne radioactivity to areas of higher potential airborne radioactivity.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

a. Objectives and Design Basis

1. Detectors are located in areas that may be normally occupied without restricted access and which may have a potential for radiation fields in excess of the radiation zones described in Subsection 12.3.1.
2. The detectors provide on-scale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located, as well as the maximum dose rate for anticipated operational occurrences.
3. Each monitor has local visual and audible alarms, with variable setpoint.
4. Indication and annunciation are available in the main control room.
5. The design objectives and location criteria are in conformance with 10 CFR Part 20, Part 70 and Part 50, Appendix A, General Design Criteria 63 and 64, and Regulatory Guides 1.21, 8.2 and 8.8.
6. Post-accident monitoring instrumentation is provided as discussed in Section 7.5.

b. System Description

The digital computer-based Radiation Data Management System (RDMS) consists of local microprocessors for each channel, interconnected by a redundant communication loop to a redundant host computer system. Either of the two computers can by itself provide the total computing capacity required for satisfactory operation of the RDMS. The host computer system, in turn, is connected to an operator display/control console in the control room, the health physics control point, the RDMS computer room, the Main Plant Computer System (MPCS) computer room and the hot chemistry lab. The area radiation monitoring system instrument engineering diagram, Figure 12.3-19, shows an overview of the system, its components and location.

Table 12.3-14 lists the various area radiation monitoring channels provided and their pertinent design information, such as detector type, range, background radiation, safety class, alarm setpoints, referenced drawings for location of area radiation detectors, etc.

Class 1E area radiation equipment is supplied from Class 1E uninterruptible power supplies (UPS).

Except for the post-LOCA containment monitors and other high-range monitors, each channel is equipped with a radioactive check source which can be actuated from the main control room during test. The post-LOCA monitors and other high-range monitors use an electronic signal to test the circuit.

A typical channel is shown in Figure 12.3-19.

High radiation levels during refueling at the manipulator crane area in the containment structure initiates isolation of the containment purge and vent system.

Those detectors which are designated as non-Class 1E, and are located inside the containment structure are not designed to operate following a major LOCA, and are assumed to be not available to monitor post-LOCA conditions inside Containment.

Refer to Subsection 11.5.2 for a discussion of the local microprocessor provisions and operating details.

1. Area Monitor Detectors

The area monitors employ Geiger-Mueller and ion chamber gamma detectors, as indicated on Table 12.3-14.

2. Class 1E Requirements

Separate redundant cabinets are provided in the control room for control, recording and remote indication for those monitors in Table 12.3-14 designated as Class 1E. These cabinets and Class 1E area monitors are powered from their respective Class 1E inverters. Class 1E monitors supply their data to the RDMS host computer through an IEEE 279 acceptable isolation device. No information or alarm setting is permitted between the RDMS host computer and the Class 1E equipment. All setpoint changes and check-source insertions are performed locally or from hard-wired modules in the control room.

3. In-Containment High Range Monitoring

Redundant Class 1E monitors are provided to monitor containment conditions under accident situations. The detector range is 10°-10^{8} R/hr. The electronics cabinet is located outside Containment in the electrical tunnels. Indication is provided on the RDMS video displays and the RDMS racks in the main control room. These monitors will be designed, located, calibrated and qualified in accordance with Table II.F.1-3 of NUREG-0737.

The detectors are located on the steam generator biological shield wall (the "A" detector is near steam generator "D" and the "B" detector is near steam generator "B") at an approximate elevation of +31'. These locations were selected to provide the detectors as large a view of Containment as possible, consistent with affording ease of access for maintenance and calibration.

4. Area Monitor Channel Description

(a) Containment Manipulator Crane Area Monitor-Channels 6535 A and B

Redundant Class 1E detectors are located on the manipulator crane. In the event of a fuel handling accident, these monitors in conjunction with safeguards actuation signals isolate the containment online and offline purge isolation valves, trip containment air pre-entry, refueling supply and containment online purge fans. Indication and alarm are provided locally and in the main control room.

(b) Personnel Hatch (Post-LOCA) - Channels 6536-1,2

This area monitor is located external to the Containment and is aligned with the personnel hatch. This radiation monitor is intended to monitor ambient radiation conditions following a LOCA.

(c) Containment Post-LOCA - Channels 6576A, B

These detectors are intended to monitor conditions inside Containment for Post-LOCA and are Class 1E.

(d) Volume Control Tank - Channel 6540

The detector is located inside the volume control tank area.

(e) High Range Area Monitors - Channels 6508-1,2, 6563-1,2, 6517-1,2 and 6518.

There are seven high range ion chamber detectors. In compliance with Regulatory Guide 1.97, these detectors have an upper range of 10^4 R/hr. These monitors are located in areas which may require entry after an accident or which contain recirculating post-accident fluids. These monitors are:

1) PAB - High Range Area Monitor - Channels 6508-1,2 and 6563-1,2

2) RHR - High Range Area Monitor - Channel 6517-1,2

3) FSB - High Range Area Monitor - Channel 6518

(f) Other Area Radiation Monitors - Channels 6534, 6537, 6538, 6539, 6541, 6543, 6544, 6545, 6546, 6547, 6549, 6550, 6551, 6552, 6553, 6554, 6555, 6556, 6557, 6558, 6559, 6570, and 6571

These channels use Geiger-Mueller detectors and monitor the ambient radiation at various points throughout the facility as listed in Table 12.3-14.

5. Calibration and Maintenance

Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

a. Objectives and Design Basis

The ventilation airborne Radioactivity Monitoring System provides radiation measurements, indications, records, alarms and controls at selected locations to detect and control radiation levels within Containment, Service Building, Radwaste Building and the plant vent, and to verify compliance with applicable limits of 10 CFR 20 and 10 CFR 50, General Design Criteria 19, 63 and 64.

On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

Monitored points within the station ventilation system are in areas where potential personnel exposure to radiation is most likely and in several ventilation exhaust ducts. A tabulation of the airborne Radiation Monitoring System is found in Table 12.3-15, Table 12.3-16 and Figure 12.3-20.

Those monitors which are Class 1E are listed in the above tables and further discussed in Subsection 12.3.4.1b.2.

The sensitivity of the airborne radioactivity monitors is such that they should be capable of detecting ten MPC-hours of particulate and gaseous radioactivity in those plant areas that have contained sources of airborne radioactivity and which may be occupied by personnel. Typical airborne concentrations for various plant areas are given in Table 12.2-31, Table 12.2-32, Table 12.2-33, Table 12.2-34, Table 12.2-35, Table 12.2-36, and Table 12.2-37.

As discussed in Subsection 12.5.3.1, the Health Physics Program includes requirements to perform sampling and analysis for airborne radioactivity, routinely and during specific evolutions such as opening of the primary system. Sampling equipment includes portable continuous air monitors and portable samplers. The monitoring and sampling capabilities, when combined, provide sufficient information to permit adequate protection of personnel from exposure to airborne radioactivity.

b. System Description

Subsection 12.3.4.1b describes the digital computer based RDMS. Subsection 11.5.2 describes the local microprocessor provisions. The airborne Radioactivity Monitoring System consists of two basic types of monitoring systems:

- Particulate and gaseous monitors (with iodine sampling) which are skid-mounted and utilize pumping systems.

- Gross activity monitors which consist of detectors mounted directly in duct air stream.

A typical channel is shown in Figure 12.3-20.

1. Particulate and Gaseous Monitors

   Each airborne particulate and gaseous monitor has some or all of the common equipment as follows:

   (a) Isokinetic Sampler (RM 6528)

   Sampler and lines adhere to requirements of ANSI N13.1. Sample line sizes are one-half inch with flow rate designed for 2-3 scfm. All sample lines slope from high point (isokinetic sampler) to the low point (sample pump).

   (b) Pumping System (RM 6526, 6528, 6531, 6532 and 6548)

   (1) The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "fresh" sample) to the detector.

   (2) The pump unit consists of:

   a A pump to obtain the air sample

   b A flowmeter to indicate the flow rate

   c A flow control valve to provide flow adjustment

   d A flow alarm assembly to provide low and high flow alarm signals.

   (3) Selector valves are used to direct the desired sample to the detector for monitoring and to block flow when the channel is in the maintenance or "purging" condition.

   (4) Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "fresh" sample.

   (5) A sample flow rate indicator is calibrated linearly from 0 to 4 scfm.

   (6) Indicator lights are actuated by the following:

   a Flow alarm assembly (low or high flow),

   b The filter paper sensor (paper drive malfunction), or

   c The pump power control switch (pump motor on).

(c) Detectors

(1) Particulate: (RM 6526)

The particulate channel air sample is drawn in a closed system monitored by a scintillation counter-filter paper detector assembly. The filter paper collects 99 percent of all particulate matter greater than 0.3 micron in size on its continuously moving surface, and is viewed by a photomultiplier-scintillation crystal combination.

The air sample is returned after it passes through the series-connected iodine filter and gas monitors.

The detector is a hermetically sealed scintillator crystal combination. The pulse signal is transmitted to the radiation monitoring system local cabinet.

Lead shielding reduces the background radiation level to prevent interference with the sensitivity of the detector. The filter paper mechanism, and electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit. The unit contains an approximate 25-day filter paper supply at normal speed.

(2) Iodine: (RM 6526, 6528, 6531, 6532 and 6548)

The particulate filter is followed by a cartridge-type charcoal iodine filter. The iodine filter is a charcoal filter which will meet certain specifications for iodine removal. The iodine filter is suitable for laboratory analysis.

The iodine filter is not monitored continuously. The iodine filter is periodically removed and analyzed as appropriate.

(3) Gaseous: (RM 6526, 6528, 6531, 6532 and 6548)

The gaseous channel uses a beta scintillator to view the air sample passing through a fixed shielded volume. The sample is then returned to its environment.

The detector assembly is in a completely enclosed housing containing a beta scintillator mounted in a constant gas volume container. Lead shielding reduces the background radiation level to prevent interference with the detector's sensitivity.

Monitor 6528 has two additional fixed volume gas chambers for mid and high range channels that use solid state Cd-Te detectors.

2. Particulate and Gaseous Monitor Channel Descriptions

(a) Containment Monitor - Channel 6526 and 6548

Monitor 6526 draws a containment air sample through redundant pumps from the containment atmosphere. The sample is then returned to the containment atmosphere.

Monitor 6548 is located inside Containment at zero foot elevation and acts as a backup to monitor 6526. Monitor 6548 draws air sample from the Containment via the sample pump and then discharges back to the Containment.

These monitors are classified seismic Category I.

Indication and alarm is available locally and in the main control room.

See Subsections 5.2.5.3b.2, 5.2.5.5b and 5.2.5.5c for a further discussion of monitoring requirements.

(b) Waste Process Building Monitor - Channel 6531

The major potential release of airborne radioactivity in the Waste Processing Building is that associated with the Gaseous Waste Processing System. The gas dryers, carbon delay beds and the two gas compressors are situated in their individual compartments, and these compartments are ventilated in such a way that they are at a negative pressure with respect to surrounding areas. The ducted ventilation exhaust is continuously sampled and monitored. The sample is returned to the ducted ventilation exhaust line which is directed to the plant vent. Both the sampling point and the return are downstream of the filters in the Waste Processing Building ventilation exhaust. Information from this channel is displayed and alarmed on the radiation monitoring system panel in the main control room and locally.

(c) Primary Auxiliary Building Monitor - Channel 6532

Three minimum ventilation areas have been defined for the Primary Auxiliary Building:

(1) Heat exchanger, thermal regeneration demineralizer, and mixed bed demineralizer area

(2) Volume control tank area

(3) Charging pump area.

These areas, which are potential sources of airborne activity, are maintained at a negative pressure with respect to surrounding areas. The PAB ventilation system collects potentially contaminated air through a duct system and discharges it to the plant vent via filter train F-16. The sample withdrawal point for this monitor (RM-6532) is downstream of filter train F-16. The location of this sample withdrawal point provides an early warning to the operating personnel in the event that radioactive material becomes airborne in the PAB.

Indication and alarm is available locally and in the main control room. An alarm indication on these monitors would trigger a radiological evaluation within the areas served by these monitored ventilation lines. The evaluation would be performed by station HP personnel using portable survey and/or air sampling equipment, as necessary, to locate the source of the elevated ventilation line indication.

(d) Plant Vent Monitors - Channels 6528-1, 6528-2, 6528-3, and 6495

These detectors monitor the air exhausted by the Primary Auxiliary Building, Waste Process Building, Fuel Storage Building, containment structure and containment enclosure via the plant vent. An isokinetic probe, supplemented with an integral pumping system is used to withdraw an air sample from the plant vent. The air quantities exhausted via the plant vent are indicated on Figure 9.4-5, Figure 9.4-6, Figure 9.4-7, Figure 9.4-8, Figure 9.4-9, and Figure 9.4-10.

Multisensor flow transmitters and microprocessor provide a signal to the radiation monitor (RM-6528) to permit this monitor to calculate the microcuries per cubic centimeter flowing in the duct and release rate in microcuries per second.

The air collected by the isokinetic probe passes through the wide-range gas monitor, WRGM, (RE-6528-1, RE-6528-2, and RE-6528-3). Sampling provisions are located downstream of the isokinetic nozzle. The air flow enters the sample conditioning skid of monitor RM-6528 at a flow rate of ≈0.06 cfm during postulated accident conditions. This skid is intended to provide representative particulate and radioiodine samples for laboratory analysis (for normal operation as well as accident conditions) and to prevent contamination of the gas monitors. A multiple filter arrangement is provided to allow sampling capabilities for the duration of the measurement period. Each filter is equipped with a 4π solid lead shielding and quick disconnect fittings to minimize personnel exposures. In addition, all functional control is performed remotely.

The wide-range gas monitor has the capability of detecting a wide range of radiogas concentrations over 12 decades. The monitor meets the requirements of NUREG-0737, item II.F.1 by providing an upper range of 10^5 μCi/cc for noble gases and the capability to collect post-accident plant vent grab samples. A seven-day composite grab sampling system for normal operation is also provided for particulates and iodine determination. Indication and alarm from the WRGM are available locally and in the main control room.

A portable pump with detector (RE-6495) provides backup to the WRGM mid and high range. Indication and recording are provided in the Technical Support Center (TSC).

3. Gross Activity Monitors

These units do not utilize a pumping system. The detectors are located directly in the duct air stream. These monitors employ Geiger-Mueller type detectors. The local microprocessor cabinet provisions are described in Subsection 12.3.4.1b.

4. Gross Activity Monitor Channel Descriptions

(a) Administration and Service Building Fume Hoods Monitor -Channels 6523, 6524 and 6525

These detectors are Geiger-Mueller counters and are located in each of the chemistry fume hoods. They measure the gross activity of air exhausted from the shop fume hoods to atmosphere. Local alarm and indication are available near the fume hoods. No control function is provided.

These monitors provide data to the RDMS host computer for alarm, display and documentation. Remote indication and alarm are available in the main control room.

(b) Fuel Storage Building Exhaust Monitor - Channel 6562

This detector is a Geiger-Mueller counter and is located in the fuel storage building ventilation exhaust duct downstream of the fans. This detector measures the gross activity vented from the Fuel Storage Building to the plant vent. Indication and alarm is available locally in the Fuel Storage Building near the spent fuel storage pool, and remotely in the main control room.

(c) Containment Enclosure Emergency Exhaust Monitor – Channel 6566

This detector is a Geiger-Mueller counter and is located downstream of the containment enclosure emergency exhaust filter fans and measures the gross activity exhausted to the plant vent stacks. Indication and alarm are available locally near the filter fans and remotely in the main control room.

(d) Primary Auxiliary Building, Miscellaneous Ventilation Exhaust – Channel 6567

This detector is a Geiger-Mueller counter and is located at the inlet to the primary auxiliary building cleanup filter. The following areas are monitored by this detector: valve aisle, volume control tank area, sample heat exchanger room, sample room fume hood, degasifier area, PAB lower level elevation (-)6', and PAB filter and heat exchanger area.

Indication and alarm are available locally near the cleanup filter, and remotely in the main control room.

(e) Containment Enclosure Monitor - Channel 6568

This detector is a Geiger-Mueller counter and is located in the exhaust duct from the containment enclosure at the inlet to the cleanup filter. The detector monitors the gross activity exhausted from the containment enclosure. Indication and alarm are available locally near the cleanup filter, and remotely in the control room.

(f) Control Room Air Intake Monitors - Channels 6506A and B, 6507A and B

Four detectors are located in the east air intake piping and four detectors are located in the west air intake piping. These detectors are located in the Control and Diesel Building. These GM detectors, which are Class 1E, monitor the control room air intake and automatically shut down, on a high radiation signal, the control room ventilation fans and isolation dampers. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal. These detectors are directly mounted in the air intake stream and do not require shielding.

Indication and alarm are provided locally. Indication, recording and alarm are provided in the main control room.

(g) Containment Online Purge Monitor - Channels 6527A, 6527B

For a description of this monitor see Subsection 11.5.2.1.

5. Portable Continuous Air Monitors (CAM)

Four portable continuous air monitors are available. The CAMs are equipped to monitor particulate and noble gas. Portable CAMs used to satisfy this commitment may be interfaced with the Airborne Radioactivity Monitoring system or they may be portable instrumentation as described in Section 12.5.2.2.

The normal locations for the CAMs are as follows:

- Waste Process Building
- Primary Auxiliary Building
- Fuel Storage Building
- Containment (on the operating floor during refueling outages)
- Control Building (during normal operations)

CAMs may be moved to other station locations as radiological conditions dictate.

6. Calibration and Maintenance

Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.3 Post-Accident In-Plant Iodine Assessment

The capability exists for the determination of airborne radioiodine levels in-plant under accident conditions. This capability includes the use of air samplers with radioiodine-specific sample cartridges and the use of gamma spectroscopy instrumentation for sample analysis. Information on portable air sampling and counting room equipment is discussed in Subsection 12.5.2.

This sampling and analysis is described in station procedures in which station personnel are trained. Training includes the proper handling and preparation of high-level radioactive samples and the operation and calibration of gamma ray spectroscopy equipment for post-accident sampling in addition to normal sampling techniques.

12.4 DOSE ASSESSMENT

12.4.1 Criteria and Objectives

Seabrook Station and its operational procedures are designed to ensure that radiation exposures to the station's operating and maintenance personnel during normal operations and anticipated operational occurrences are within the limits specified in 10 CFR 20 and as low as reasonably achievable (ALARA).

Regulatory Guide 8.19 was not available until after the plant was already past the early stages of construction; however, in all phases of the design and construction of the plant, careful consideration has been given to eliminating unnecessary exposures and, wherever practical, minimizing the exposures of station personnel.

The radiation exposures to station personnel presented in Subsection 12.4.2 are pre-operation estimates based on PWR operating experience in the 1970s and Seabrook Station's design features. The occupational radiation exposures in Light Water Reactors have since been reduced significantly. The estimated man-rems presented in Tables 12.4-1 through 12.4-7 are largely independent of the fine variation of the reactor power. For an operating nuclear power station, the actual occupational exposures are carefully recorded and reported. Following operation at the licensed core power level, the occupational exposures are expected to increase approximately in proportion to the reactor power.

The estimated annual direct and scattered doses at the Site Boundary presented in Subsection 12.4.3 were calculated conservatively and remain valid following operation at the licensed core power level.

12.4.2 Occupational Radiation Exposures

In general, several factors have been found which effect personnel exposures. These factors are:

a. The number of years a plant has operated. PWR exposures tend to show large increases during the first few years of operation, but tend to level out after several years

b. Plant design and equipment layout

c. The extent that maintenance is required in a specific year

d. The amount of corrosion products, fission products and activation material deposited on or circulating through various parts of the plant systems

e. Training and experience of workers

f. The extent of direct supervision inside the radiation control area

g. The extent that a utility uses unfamiliar or contractor personnel

h. Extended operation with failed fuel

i. The integrity of the primary fluid systems.

Radiation exposures in the station are primarily due to direct and scattered radiation from components and equipment containing radioactive fluids. In some plant radiation areas, personnel can also be exposed to airborne radionuclides.

Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16 and Figure 12.3-17 identify the locations of the major sources of radioactivity and the classifications of the various radiation zones in the plant. In Zone I areas, such as offices and control rooms, the maximum dose rate does not exceed 0.5 mrem/hour. Occupancy in Zones II through V is limited by administrative controls to ensure that personnel do not receive doses in excess of the 10 CFR 20 limits. (See Table 12.3-1 for estimated occupancies of radiation zones by work function.)

12.4.2.1 Operating PWR Data

Reports summarizing the occupational radiation exposures at operating nuclear power stations (References 1-5) have provided important input data for the design of Seabrook Station. To cite specific examples, the operational exposure data indicate that steam generator maintenance and in-service inspection are key high exposure activities (Reference 1). An average of 27 percent of the total annual exposures at PWRs can be attributed to steam generator work. In-service inspection is the next most significant activity, causing an average of 5.6 percent of the total annual exposures.

Operating data given in the above references has helped to identify the most significant dose-causing activities, and to establish the priorities for ALARA-related design reviews and improvements.

12.4.2.2 Direct Radiation Dose Estimates

Average annual doses within Seabrook Unit 1 were estimated by comparing the plant's design features with the appropriate historical exposure data from operating plants. Expected radiation fields and exposure times were projected for all major activities within the plant. The estimated dose rates and exposure times for each task were multiplied to obtain an estimate of the occupational radiation exposure for each task or activity, and summed to obtain the expected annual dose for the unit.

The estimated annual occupational radiation exposure is expected to be approximately 372 man-rem. Table 12.4-1 summarizes the exposures by major task; Table 12.4-2, Table 12.4-3, Table 12.4-4, Table 12.4-5, Table 12.4-6 and Table 12.4-7 provide the detailed dose and exposure time estimates by major task.

12.4.2.3 Inhalation Exposure

The annual man-rem dose from airborne activity in the Seabrook plant will be very small compared with the annual man-rem doses due to direct and scattered radiation. Potential sources of airborne radioactivity (Subsection 12.2.2) have been analyzed and reduced to levels that are considered as low as reasonably achievable for both occupational and nonoccupational exposures.

As a result of operational occurrences and maintenance activities, personnel are required to enter areas where airborne radioactivity exists, or is expected. In such cases, the airborne concentrations are monitored by health physics personnel. The necessary protective devices (respiratory protective devices, protective clothing, etc.) and, if appropriate, control devices (portable ventilation, containment, etc.) are specified according to the concentration(s) and radionuclide(s) present. The respiratory protective devices are used in conjunction with administrative controls (Section 12.5) to limit occupancy times for personnel working in a contaminated atmosphere. Records of the inhalation exposures accrued by plant personnel are maintained.

In summary, inhalation doses to personnel are limited by:

a. Maintaining positive control to ventilation air in contaminated work areas

b. Health physics survey of work areas to identify the radionuclide concentrations present in personnel work areas.

c. Controlling the occupancy time in areas with airborne contamination, and, when necessary, requiring personnel to use respiratory protective devices and protective clothing

d. Actions taken in response to positive whole body or bioassay results.

The measures described above minimize exposures to airborne radioactivity and ensure that the doses to individual employees from airborne radioactivity are small fractions of the 10 CFR 20 limits for occupational workers, and that annual man-rem doses comply with the ALARA criteria. Periodic whole body counts and special bioassays, as described in Subsection 12.5.3.5, are conducted to monitor personnel internal exposure and to evaluate the effectiveness of the program.

12.4.2.4 Exposure Reduction Methodology

Plant design features which are applicable to the Occupational Radiation Exposure (ORE) - ALARA program are enumerated in Subsection 12.3.1.

The plant layout is designed so that passage through a high radiation area to obtain access to a lower radiation area is avoided.

Wherever practicable, components are shielded individually to keep exposures ALARA during maintenance and inspection activities. The reactor cavity is specially shielded to reduce neutron dose rates at the operating floor.

Auxiliary control boards are located with ALARA concepts in mind, and shielded pipe chases are used extensively in order to segregate radioactive piping from normally occupied areas.

The plant's ventilation system (Section 9.4) is designed to maintain the concentration of airborne radioactivity in occupied areas well below the occupational concentration values specified in 10 CFR 20, Appendix B, Table 1, Column 3. The airborne radioactivity monitoring instrumentation (Subsection 12.3.4) provides radiation measurements, alarms, and controls at selected locations in areas where the potential exposures are most likely.

To further reduce exposures, Seabrook Station uses redesigned steam generators that need considerably less maintenance. Improvements in secondary system water chemistry and tube support plates have reduced the likelihood of the need to plug tubes. Also, the volatile chemical treatment employed for the secondary system alleviates the need for sludge lancing of the secondary side.

Steam generator cladding and tube inspections will be performed wherever possible with remotely operated equipment, and the mechanical tube plugging technique will be used when feasible. This process has proved successful in reducing the doses from this task by more than a factor of two compared to the manual welding method.

Seabrook Station has incorporated a simplified reactor vessel head assembly (SHA) into the plant design to reduce personnel exposure and outage time related to reactor vessel head removal and reinstallation.

The SHA is a modification to the reactor vessel head assembly which replaces the existing missile shield, eliminates the CRDM ductwork simplifies the control rod drive mechanism and control rod drive rod position indication cable configurations, and allows the reactor vessel head lift rig to remain permanently mounted on the reactor vessel head.

Installation of the SHA is estimated to save 3 person-rem each refueling outage by eliminating missile shield/ventilation duct removal and installation. These changes are reflected in the updated dose estimate shown in Table 12.4-4. Table 12.4-4 also has been updated to eliminate disconnecting/reconnecting reactor head instrument port thermocouples. Reactor head instrument port thermocouples are not employed in the Seabrook design. This results in an additional reduction of 4 person-rem from the refueling operations dose estimate shown in Table 12.4-4.

The summary in Table 12.4-1 has also been updated to reflect these changes.

12.4.3 Estimated Annual Dose at the Site Boundary and to the Populationat Large

12.4.3.1 Direct And Scattered Radiation Dose Estimate

For normal plant operations, the maximum direct and scattered dose rate external to the reactor, the Primary Auxiliary Building, and the Waste Processing Building is less than 0.5 mrem/hour by design. The shielding design is based on a conservative assumption of a 1 percent failed fuel fission product source term. Actual levels are expected to be significantly less than the 0.5 mrem/hour design value. A level of 0.5 mrem/hour at the outside surface of the Primary Auxiliary Building would result in a dose rate of less than 0.05 mrem/year at the exclusion area boundary.

The principal sources of radioactivity not stored in plant structures are the radioactive liquids stored in the reactor makeup water storage tanks and the refueling water storage tanks that are located in the Tank Farm. The location of the Tank Farm is shown on Figure 1.2-1. The maximum expected radionuclide inventories in each of these tanks is provided in Table 12.2-21 and Table 12.2-22. The total direct radiation dose rate at the exclusion area boundary from all four of these tanks with the maximum radionuclide inventories is calculated to be less than 5 mrem/year.

12.4.3.2 Doses Due to Liquid And Gaseous Releases

Estimates of the doses at the site boundary and beyond from station liquid and gaseous releases are provided in Subsections 11.2.3 and 11.3.3, respectively.

12.4.4 References

1. "Compilation and Analysis of Data On Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum, Inc., National Environmental Studies Project, September 1974.
2. "Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974," NUREG-75/032, June 1975.
3. "Occupational Radiation Exposures at Light Water Cooled Power Reactors 1976," NUREG-0323, March 1978.
4. "Occupational Radiation Exposure, Tenth Annual Report, 1977," NUREG-0463, October 1978.
5. "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1978," NUREG-0594, November 1979.
6. "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980.

7. "Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw, presented at the International Radiation Protection Conference, Paris, December 1979.

8. "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," (to be published) prepared for Atomic ndustrial Forum, Inc. by Catalytic, Inc., (1979-1980).

12.5 HEALTH PHYSICS PROGRAM

The health physics functions are organized and equipped to protect plant employees from unnecessary exposure to radiation and radioactive materials.

12.5.1 Organization

The Seabrook Station staff and the organization are discussed in Section 13.1. Health physics personnel meet the qualification requirements in Section 13.1.3.1.

The health physics organization is responsible for the overall implementation of the operational health physics program. The health physics organization has the responsibility and authority to report to the Station Director on any aspect of the radiation protection program, or its implementation, as deemed necessary. The health physics organization is responsible for ensuring station compliance with the applicable federal and state radiation protection regulations.

The health physics support staff may include supervisory personnel, health physicists and other personnel to assist with implementing the radiation protection program. The supervisors are responsible for directing the activities of the health physics technicians.

The health physicists provide administrative and technical assistance to the health physics supervisors. Their principal responsibilities include providing program level guidance in such technical areas as internal/external dosimetry, radiation detection/measurement and dose reduction (i.e., ALARA).

Health physics technicians are responsible for performing the routine and daily operations of the department. These duties include performing various surveys, collecting air samples, maintaining department equipment and instrumentation and providing radiation protection/control coverage, as necessary, during station operations and maintenance activities.

Assistant health physics technicians may be employed to assist in department activities. Their duties may include assisting the technicians in the performance of surveys, sampling, radiation protection, dosimetry, whole body counting, and maintaining department equipment.

The assistant technicians participate in the health physics technician qualification program, and when an individual meets the technician qualification requirements, he/she is eligible for the technician position.

Section 13.2 contains information on training that will be given to the Health Physics department. In addition to the formal training provided by the training department, the health physics staff will provide additional specialized instruction to their department technicians.

12.5.2 Equipment, Instrumentation and Facilities

The selection criteria for equipment and instrumentation presented can be met by several manufacturers. Equipment is purchased from manufacturers that can supply suitable equipment and instrumentation, provide repair services when required, and provide replacement parts without undue delay.

Facility design and equipment are selected to facilitate dose reduction. The facilities are designed with adequate working spaces and for ease of access from working locations. Decontamination facilities are located at the Fuel Storage Building, Waste Processing Building, and in the Administration and Services Building.

12.5.2.1 Counting Room Equipment

The instrumentation in the counting rooms is used for determining airborne radionuclide concentrations, removable contamination, and radionuclide concentrations in liquid samples.

Two counting rooms house laboratory radiation detection equipment, one for health physics support and the other for chemistry support. The health physics count room is equipped with alpha, beta and gamma detection equipment to analyze routine air samples and contamination survey smears. The health physics counting equipment is supplemented by the chemistry counting room when additional analytical capabilities in a low background area are required. The gamma detection equipment includes germanium detectors coupled to gamma spectroscopy equipment. This gamma detection capability is available in both the chemistry and health physics counting rooms.

This equipment will be capable of detecting, as a minimum, alpha, beta and gamma activity (as specified above). This counting room equipment is used primarily for quantitative and qualitative analysis of liquid, smear, and air samples.

Criteria for equipment selection are numerous and include accuracy, stability under various atmospheric conditions, sensitivity, and compatibility with many types of peripherals.

Calibration and operational equipment checks are performed and documented on a routine basis. Radiation background and detection performance factors are normally checked prior to equipment use each day. Calibration of equipment is performed by the appropriate department technicians periodically and will include an efficiency check, operational checks, and (when applicable) a plateau check. All equipment is recalibrated upon completion of major maintenance.

12.5.2.2 Portable Instruments for Measuring Radiation

The portable instrumentation consists of low- and high- range ion chamber dose rate meters, Geiger-Mueller count rate meters, scintillation or proportional alpha counters, neutron rate meters, and air samplers. The selection of instrumentation has been partially based on equipment suitability for use during emergencies, because emergency conditions require a wide range of equipment capable of withstanding extreme usage conditions.

Criteria considered for equipment selection include linearity of response at different temperatures and humidities, reliability of response, accuracy, geometric and energy dependence, stability, and dependability. Other desirable criteria include versatility, ease of obtaining spare parts on a timely basis, sturdiness, compactness, weight, and ease of operation.

Sufficient quantities of each type of instrument are available to permit calibration, preventative and corrective maintenance, and handling of peak work loads without diminishing the radiation protection program. The types, quantities, and other information about instrumentation are presented in Table 12.5-1.

Most instruments are normally stored at the main health physics control station where they are easily accessible. Extra equipment not intended for daily use is stored in other health physics storage areas. Instruments may also be stored at temporary control points to make them more accessible from some work areas.

Calibration of portable survey instrumentation in use will be conducted on an annual basis or as necessary due to maintenance, questionable accuracy, or possible damage. Prior to use, an operational check will be performed which will include a battery check and a source check.

Calibration techniques of air samplers will vary according to type. The air samplers will be calibrated annually, after maintenance that may affect operation, and when accuracy is questionable. The calibration procedure for the low volume, high volume, and personnel air samplers consists of determining the actual flow rate. Continuous air monitor (C.A.M.) calibration procedures are dependent upon type chosen, but generally consist of flow rate adjustments, source checking of detectors, and checking filter paper speed and adjustment, if necessary.

12.5.2.3 Personnel Monitoring Instruments

The instruments used for detecting personnel contamination and exposure, listed in Table 12.5-2, are state-of-the-art when practicable. The official and permanent records of accumulated external and internal radiation exposure will be generated from data acquired by this instrumentation.

Two types of friskers (personnel external contamination monitors) are used throughout the station. The first type of frisker is a portal monitor or similar large area detection device and the second type is of the portable hand held probe design. These friskers are designed for routine use by all personnel that exit the station or Radiologically Controlled Area (RCA). Criteria for frisker selection include ease of use, sensitivity to small amounts of radioactive materials, and the ability to locate contamination on the body.

Frisker locations throughout the plant are dependent upon projected and actual personnel traffic. At least one portal monitor and one hand-held probe-type frisker are located at the facility access building during normal conditions. The number of friskers and portal monitors at the facility access point may be increased during peak personnel traffic. Personnel external contamination devices are located at the RCA entrance-exit and other locations as necessary to control contamination. Other locations may include, but are not limited to:

- Waste Processing Building
- Spent Fuel Storage Building
- Primary Auxiliary Building
- RCA machine shop
- Radio-chemistry laboratory
- Containment personnel hatch

Calibration of hand-held probe-type friskers is performed at annual intervals, when damage may have occurred, accuracy or reliability is questionable, and when maintenance is performed that could affect operation. Calibration procedures include source response checks, sensitivity correction, and alarm operability.

Instrumentation for determining individual exposure from external sources, such as self-reading pocket dosimeters (SRPDs) or electronic dosimeters are available on site and controlled by the health physics department. Electronic Dosimeters (EDs) are the primary means of monitoring incremental exposures to individuals. SRPDs will be used as a backup. The SRPDs are of various ranges: predominately 0 to 200 mR for general use, with 0 to 1000 mR and 0 to 5000 mR for use during specific high-exposure tasks or emergency conditions.

SRPD selection is based primarily upon results of drift test, accuracy test, drop test, and usable lifetime as determined by those in use at other facilities. Dosimeters are issued by health physics personnel and stored near the Radiologically Controlled Area control point. Calibration of SRPDs is performed every six months or prior to putting the dosimeters into use. The calibration procedure provides criteria for acceptance of both the source and drift tests which are performed as calibration checks. Calibration of SRPDS will be performed prior to placing them in service.

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from the reading of thermo-luminescence dosimeters (TLD). These dosimetry devices are capable of measuring beta, gamma, and neutron radiations in mixed radiation fields in accordance with industry standards. Dosimetry devices are issued to radiation workers after completion of check-in requirements at the station. Criteria for issuance are outlined by procedures developed by the Health Physics Department. TLD badges and processing services are provided by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited laboratory.

The TLDs are processed at a frequency not exceeding the period for which dose limits are listed in 10 CFR 20 for occupational exposure. Any official dosimetry device used at Seabrook Station requiring processing is supplied and evaluated by a laboratory accredited through the National Voluntary Laboratory Accreditation Program.

Selection criteria for the whole body counter consist of quantitative accuracy, qualitative accuracy, and the ability to determine low body or organ burdens of common PWR gamma emitting radionuclides.

The whole body counter is located onsite in a low background area. Calibration of the whole body counter will be performed or verified annually, when major maintenance is performed, and when results appear to be incorrect in accordance with health physics procedures. Operational checks are performed prior to daily use and include a background count, quality control sample measurement and software verification.

An in-vitro bioassay program has been established. A vendor laboratory having the capability of analyzing urine and fecal samples for body burdens of common PWR radionuclides, including tritium is used. The health physics organization verifies that the vendor's quality assurance program assures accurate analysis. The program is used for accidents, incidents, and suspected uptakes of radionuclides not detectable by whole body counting equipment.

12.5.2.4 Personnel Protective Clothing

Protective clothing is provided for employees required to work in areas where this clothing is necessary. The protective clothing is kept at the RCA change area and, when required, at the location of work. The quantity of protective clothing that is maintained for use is calculated considering normal usage, quantity in the laundry process, and anticipated work. Examples of the types of protective clothing that may be available are as follows:

- Coveralls
- Lab coats
- Plastic suits
- Surgeon caps or similar head covers
- Cloth hoods
- Plastic shoe covers, low and high
- Rubber boots
- Rubber shoe covers
- Cotton gloves
- Rubber gloves
- Goggles

- Face shields
- Welding shields

12.5.2.5 Respiratory Protection Equipment

A respiratory protection program has been established and procedures written to reasonably ensure that personnel exposure to radionuclide concentrations do not exceed the exposure limits for internal dose as specified in 10 CFR 20. The protection factors assigned to respiratory equipment are those specified in Appendix A of 10 CFR 20.

The equipment consists of facepiece-fitting instrumentation, full-face respirators, self-contained breathing apparatus (SCBA), airfed hoods, airlines, filter canisters and cartridges, communication devices, eyeglass adapters, welding equipment plus maintenance and repair supplies.

New respirators are checked prior to initial use by a visual inspection. Used respirators are cleaned, disinfected, checked for contamination and mechanically inspected prior to reissue. Emergency respirator equipment is checked each month and an inspection log is maintained.

Respirators and associated accessories are maintained and controlled by health physics. A health physics-controlled storage area is used to store new or infrequently used respiratory equipment. The RCA access area contains the equipment that may be used on a daily basis. Each respirator not equipped with its own storage case is placed in a bag to maintain cleanliness and stored to protect it from damage. Emergency respirators are located throughout the facility for rapid accessibility.

12.5.2.6 Health Physics Support Facilities

An instrument calibration facility has been established for the calibration of portable health physics instruments. The facility is equipped with the items necessary for instrument calibration and repair.

Personnel decontamination equipment and showers are located near the Radiologically Controlled Area Control Point. Soap, chemicals, brushes, and other materials necessary for personnel decontamination are available at this location. Instrumentation capable of detecting external contamination is also available near the RCA shower area.

A facility for the decontamination and cleaning of monitoring instruments and small tools is provided. The decontamination facility contains the necessary chemicals, cleaning agents, and utensils to meet the majority of small tool and instrument decontamination needs.

A respiratory equipment cleaning, sanitizing and repair facility is provided. Located at the facility are equipment and chemicals needed to check, clean, sanitize and repair all reusable respirators and respiratory equipment.

Two separate clothing change facilities (men and women) are available for preparing for use of anti-contamination clothing and are located near the health physics control point.

The health physics control point is used for instrumentation storage, as the primary access point, radiological data information center, and health physics operations. Figures showing the location of the health physics control point and other areas discussed in this subsection are included as part of Section 1.2.

In order to support increased Radiologically Controlled Area access/egress requirements, such as those normally experienced during maintenance and refueling outages, alternate facilities may be established by health physics personnel. The facilities will be provided with the necessary equipment and capabilities to ensure equivalent control of radioactive materials and personnel, as that provided by the primary RCA Control Point.

12.5.3 Procedures

12.5.3.1 Radiation Surveys

Radiological surveys are performed to determine the concentration of radio-activity in air, direct radiation from plant components and loose or fixed contamination on surfaces. Written procedures are available which provide the method and frequency of surveys.

Surveys are performed on a routine and nonroutine basis. Some factors that determine the frequency and scope of any survey are the location of the subject area or equipment, expected occupancy of any area, type of work to be performed, handling of radioactive materials, movement of station personnel and plant operations. The frequency of routine surveys varies from once per shift to annually. Nonroutine surveys are performed as necessary to monitor dismantled equipment contamination, to establish radiological controls for and during work activities, to determine operational requirements such as the need to change process system filters, and to follow projected changes of radiation fields due to station operations.

Instrumentation of the type listed in Table 12.5-1, Portable Health Physics Instrumentation, is used to determine radiation levels. Routine surveys to determine direct radiation from sources consist of measurements taken at fixed and/or nonfixed locations. The radiation survey data enables health physics personnel to determine trends and locations within an area that may require action to prevent unnecessary exposure accumulation.

Routine surveys to determine area contamination levels may be performed simultaneously with the direct radiation surveys. This type of survey enables personnel to determine contamination trends and also provides data used to determine if contamination control procedures such as those for step-off-pad usage and anti-contamination clothing usage are adequate and effective.

Surveys to determine airborne radionuclide concentrations are performed routinely and nonroutinely. Areas such as the Spent Fuel Building, sections of the Primary Auxiliary Building, and Waste Processing Buildings are routinely surveyed for airborne radioactivity. Nonroutine air surveys are provided for initial opening of the primary system and, when necessary, to ensure worker health protection.

The surveys mentioned above are in addition to the fixed monitoring instrumentation which has been placed throughout the facility and is described in Subsection 12.3.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation.

12.5.3.2 Procedures to Maintain Exposures as Low as Reasonably Achievable (ALARA)

The general operational concept of ALARA evaluation, analysis and review is presented in Subsection 12.1.3.1. Specific ALARA review levels and guide-lines are contained in health physics procedures as discussed in Subsection 12.1.3.2. Elements of Regulatory Guide 8.8, Revision 3 are exercised through the use of pre-emptive analysis and planning, ongoing observations and audits and historical analysis and review. Examples of these include daily individual dose-tracking through the use of self-reading pocket dosimeters (SRPDs) or electronic dosimeters and dose-tracking with respect to specific jobs, work groups and components through correlation of data obtained from the radiation work permit system and exposure records. Dose-tracking in these areas is applied as time, personnel and radiological conditions warrant or health physics supervision deems necessary.

Additionally, general performance guidelines have been identified for certain evolutions and operational tasks, such as those discussed below, to further promote personnel radiation exposure reduction.

Refueling procedures state the monitoring requirements, minimum fuel-to-water surface distance, and degassification of the Reactor Coolant System prior to removing the reactor vessel head for refueling operations. During refueling, the reactor coolant is filtered and cooled to help reduce airborne contamination caused by coolant evaporation.

Prior to initial handling of spent fuel, the equipment used is checked for possible damage and proper operation. During fuel movement the minimum top of fuel-to-water surface distance is maintained above a level as stated in procedures, and airborne activity will be monitored. During the initial use of the fuel transfer canal and tube, appropriate areas are barricaded, locked to access, and detailed radiation area surveys are conducted. Thereafter, radiation protection controls are used as necessary.

When shipping spent fuel, an approved cask is used. Fuel is loaded into the cask underwater, then the cask is capped or sealed, raised, surveyed, decontaminated, if necessary, and resurveyed as necessary for compliance with appropriate federal regulations prior to release.

When spent fuel is to be transported to the dry fuel storage site, a licensed cask and transport system is used. Fuel is loaded into the cask underwater, then the cask is capped and sealed, raised, surveyed, decontaminated and resurveyed if necessary for compliance with appropriate federal regulations prior to movement to the dry fuel storage facility.

Prior to performing in-service inspection (ISI), the personnel review prior inspections, verify proper equipment operation, and review general area radiation levels and hot spot locations in the vicinity of their work. Where practicable, remote ISI techniques are used.

Procedures for radwaste handling reflect the ALARA philosophy. Radwaste operators are usually stationed at a remote control console from which remote operations can be performed. Remote operations include moving, filling, and sealing containers. Prior to container usage, when possible, shielding and labeling operations are performed. Vehicles used for shipping radwaste are surveyed in accordance with Department of Transportation regulations and Seabrook Station procedures. A ventilation system with exhaust through a High Efficiency Particulate Air (HEPA) Filter System will be used at the dry waste compaction station.

During normal operation plant personnel will receive job-related training. A direct benefit of the training should be a reduction of exposure, which can be attributed to improved efficiency. Design features of the facility such as reach rods, remote activation controls, and control panels make it possible for most operations to be performed from a low radiation area.

Routine maintenance consists of activities such as scheduled and preventive maintenance. Maintenance procedures are reviewed to ensure completeness. Work on the RCA side of the plant requires a Radiation Work Permit (RWP) with its associated radiological information and requirements. Normally, when work is to be performed involving radiation and/or radioactive materials, the responsible group requests health physics to determine if a job specific RWP is required. Health physics personnel make the determination in accordance with procedures and, if necessary, investigate the actual and/or anticipated radiological conditions. If an RWP is necessary, Health Physics specifies radiological protection requirements and authorizes, by signature, the issuance of an RWP. Each individual working under the RWP shall initial it or sign an associated form signifying he/she has read and understands the conditions of work. (The control room is aware of work being performed in the Station through the work control procedures. Health Physics notifies the control room if significant changes in the physical or radiological status of the unit occur.)

Sample collection by chemistry of the reactor coolant system gas or water is normally obtained at one of the sample stations. The sample stations are, when required, equipped with a hood, ventilation system and HEPA filter. Shielded containers and long tongs are available for handling samples when this is necessary. A shielded area has been provided in the radio-chemistry laboratory for the storage of radioactive samples.

Some instrumentation requires in-place exposure to a radioactive source during calibration. The storage and handling of sources is described in Subsection 12.5.3.7. Transfer of the sources to and from the place of calibration is usually done by using a shielded container. During calibration, when possible personnel will use shielding, time, and distance to reduce their exposure.

12.5.3.3 Physical and Administrative Measures for Controlling Access

Access to the RCA is limited to those persons whose entry is requested by station supervisors and authorized by health physics personnel. Every area inside the RCA in which radioactive materials and radiation are present shall be surveyed and conspicuously posted with the appropriate radiation caution sign(s).

Access to high radiation areas is controlled through procedures developed by health physics in accordance with Technical Specification 6.11.

12.5.3.4 Contamination Control

The limits for surface contamination in the RCA are specified in procedures. When the contamination level limits are exceeded the area will be posted as a "Contaminated Area." Additionally, all personnel are required to wear protective clothing for entry to "Contaminated Areas." The contaminated area will, when possible, be decontaminated to minimize the spread of contamination.

Material and equipment will be given an unconditional release by Health Physics if they meet the criteria as specified by station procedures. Authorization for use or movement of radioactive material outside the RCA is obtained from health physics. Control over the packaging, labeling and movement of this material is provided by health physics personnel.

The limits for personnel contamination are specified in procedures. When the contamination levels are exceeded, decontamination is performed to reduce the contamination levels to an acceptable value.

12.5.3.5 Personnel Monitoring

Dosimetry is issued to visitors and station personnel in accordance with 10 CFR 20 and health physics procedures.

Surveillance for internal deposition of radionuclides is performed using both in-vivo and in-vitro analysis methods. Health physics procedures delineate analysis frequencies, sensitivities and exposure records dispositions for all analysis performed.

12.5.3.6 Respiratory Protection Program

Respiratory protective devices are usually required in situations arising from station operations in which airborne radioactivity areas exist or are expected. In such cases, the airborne concentrations are monitored by health physics personnel and the necessary protective devices are specified according to the concentration and type of airborne contaminants present.

The respiratory protection training program contains information on wearing respirators, a description of how they function, why they are used, precautions, how they are cared for, and how they are worn. A second phase of the training consists of a test, putting on a respirator, and checking for proper fit.

Respiratory protection information concerning types, uses, cleaning, and storage is given in Subsection 12.5.2.5, Respiratory Protection Equipment.

12.5.3.7 Storage and Handling of Radioactive Sources

Radioactive nonexempt sources, not part of an instrument, are controlled by health physics. The controls include source receipt, documentation, leak testing when required by Technical Specification 3/4.7.8 and assisting with the shipping process.

12.5.3.8 Updating and Auditing of the Health Physics Program

In order to keep the health physics program up-to-date and effective, procedures are reviewed periodically, rewritten as necessary, and audits are performed periodically.

Health physics procedures are reviewed periodically by health physics supervisory personnel. This review will be conducted to determine if existing procedures meet the needs of the department and, if not, to determine what modifications are necessary.

Also, when regulatory requirements change, the procedures that may need to incorporate those requirements are reviewed and, when necessary, changed.

Assessments of the health physics program effectiveness are performed regularly in accordance with 10 CFR 20 requirements.

12.5.3.9 Radiation Protection Training Program

The training given plant employees and contractors on radiation protection topics is discussed in Chapter 13.

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CHAPTER 12
RADIATION PROTECTION

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TABLE 12.2-1 NEUTRON FLUX SPECTRUM AT REACTOR VESSEL SURFACE [HISTORICAL]

Energy	Flux (n/cm^2-sec)
E > 1 MeV	7.3+8*
5.53 keV E ≤ 1 MeV	1.2+10
0.625 eV ≤ E ≤ 5.53 keV	6.8+9
E < 0.625 eV	1.7+9

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 7.3+8 = 7.3×10^8

TABLE 12.2-2 GAMMA FLUX SPECTRUM AT REACTOR VESSEL SURFACE [HISTORICAL]

Energy MeV	Flux (γ/cm^2-sec)
7.5	3.5+9*
4.0	3.2+9
2.5	1.6+9
0.8	9.6+8

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 3.5+9 = 3.5×10^9

TABLE 12.2-3 SPENT FUEL SOURCE TERM [HISTORICAL]

(4 DAYS AFTER SHUTDOWN)[1]

Energy (MeV)	*MeV/cc-sec*
0.4	1.16×10^{11}
0.8	4.90×10^{11}
1.3	1.80×10^{11}
1.7	2.50×10^{11}
2.2	9.68×10^{9}
2.5	2.68×10^{10}
3.5	9.80×10^{8}

(100 HOURS AFTER SHUTDOWN)[2]

Energy (MeV)	*MeV/cc-sec*
0.2-0.4	5.9×10^{10}
0.4-0.9	4.6×10^{11}
0.9-1.35	4.9×10^{10}
1.35-1.8	1.8×10^{11}
1.8-2.2	1.2×10^{10}
2.2-2.6	1.1×10^{10}
2.6-3.0	1.8×10^{8}
3.0-4.0	7.1×10^{7}

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

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[1] Based on S.L. Anderson, L. Clemons, J.S. Moser, J. Sejvar, "Radiation Analysis Design Manual," WCAP-7664, Rev. 1.

[2] Based on "Radiation Analysis Manual Seabrook Units 1 and 2," NAH/3-1, Rev. 3, 11/78.

TABLE 12.2-3 SPENT FUEL SOURCE TERM

(PEAK ASSEMBLY SOURCE TERM 80 HOURS AFTER SHUTDOWN)[3]

Energy (MeV)	*MeV/cc-sec*
0.01	1.592×10^{10}
0.025	4.804×10^{9}
0.0375	1.242×10^{10}
0.0575	7.864×10^{9}
0.085	3.611×10^{10}
0.125	1.396×10^{11}
0.225	1.648×10^{11}
0.375	1.040×10^{11}
0.575	4.759×10^{11}
0.85	7.895×10^{11}
1.25	1.046×10^{11}
1.75	4.682×10^{11}
2.25	2.789×10^{10}
2.75	2.747×10^{10}
3.50	2.939×10^{8}
5.0	8.157×10^{2}

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

[3] Based on SBC-833, Revison 1, "Minimum Water Depth Shielding Requirements for Peak Power Fuel Assembly with Extended Burnup."

TABLE 12.2-4 REACTOR COOLANT N-16 ACTIVITY* [HISTORICAL]

Position in Loop	Loop Transit Time (sec)	N-16 Activity µCi/gram
Leaving core	0	189
Leaving reactor vessel	1.1	170
Entering steam generator	1.4	164
Leaving steam generator	5.4	112
Entering reactor coolant pump	6.0	106
Entering reactor vessel	6.8	98
Entering core	9.0	86
Leaving core	9.7	189

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* S.L. Anderson, L. Clemons, J.S. Moser, J. Sejvar, "Radiation Analysis Design Manual," WCAP-7664, Rev. 3.

TABLE 12.2-5 SOURCE TERMS FOR THE CHEMICAL AND VOLUME CONTROL SYSTEM [HISTORICAL]

1. Regenerative Heat Exchanger

And

Excess Letdown Heat Exchanger

Gamma Energy (Mev/gamma)	Specific Source Strength (MeV/gm-sec)
0.4	4.5(+5)
0.8	2.7(+5)
1.3	1.7(+5)
1.7	1.2(+5)
2.2	1.4(+5)
2.5	1.6(+5)
3.5	1.9(+4)
6.1	2.2(+6)
7.1	1.8(+5)

2. Letdown Hbdt Exchanger

See Table 11.1-1

3. Mixed Bed Demineralizers

and

Cation Bed Demineralizer

Isotope	(Curies) Mixed Bed	(Curies) Cation Bed	Isotope	(Curies) Mixed Bed	(Curies) Cation Bed
I-131	1.2(+4)	-	Cs-134	4.0(+3)	4.0(+3)
I-132	7.3(+2)	-	Cs-136	3.8(+1)	3.8(+1)
I-133	2.9(+3)	-	Cs-137	2.7(+4)	2.7(+4)
I-134	1.4(+2)	-	Te-132	5.3(+2)	5.3(+2)
I-135	8.6(+2)	-	Ba-140	3.1(+1)	3.1(+1)
			La-140	3.2(+1)	3.2(+1)
Sr-89	9.1(+1)	9.1(+1)	Ce-144	3.6(+1)	3.6(+1)
Sr-90	2.1(+1)	2.1(+1)			
Sr-91	9.0(-1)	9.0(-1)	Mn-54	6.8(+1)	6.8(+1)
			Mn-56	1.9(+0)	1.9(+0)

Y-90	2.0(+1)	2.0(+1)			
Y-91	2.1(+1)	2.1(+1)	Co-58	1.0(+3)	1.0(+3)
Y-92	5.4(-2)	5.4(-2)	Co-60	8.6(+1)	8.6(+1)
Zr-95	2.4(+1)	2.4(+1)	Fe-59	2.9(+1)	2.9(+1)
Nb-95	1.4(+1)	1.4(+1)	Cr-51	1.5(+1)	1.5(+1)
Mo-99	1.3(+3)	1.3(+3)			

4. Letdown Degasifier Regenerative Heat Exchanger, Tube Side
   Letdown Degasifier Preheater
   Letdown Degasifier, Liquid-Vapor Region
   Moderating Heat Exchanger, Tube Side
   Letdown Chiller Heat Exchanger, Tube Side
   and
   Letdown Reheat Heat Exchanger, Shell Side

Isotope	µCi/gm	Isotope	µCi/gm
I-131	2.5(-1)	Mn-54	7.9(-5)
I-132	9.1(-2)	Mn-56	3.0(-3)
I-133	4.0(-2)	Co-58	2.6(-3)
I-134	5.8(-2)	Co-60	7.7(-5)
I-135	2.2(-1)	Fe-59	1.1(-4)
		Cr-51	9.5(-5)
Sr-89	4.1(-4)		
Sr-90	1.8(-5)	Kr-83m	4.3(-1)
Sr-91	3.1(-3)	Kr-55m	1.7(+0)
Y-90	2.2(-5)	Kr-85	1.3(-1)
Y-91	5.8(-4)	Kr-87	1.3(+0)
Y-92	1.0(-4)	Kr-88	3.4(+0)
Zr-95	6.7(-5)		
Nb-95	6.8(-5)	Xe-131m	6.7(-2)
Mo-99	3.3(-1)	Xe-133m	5.7(-1)
Cs-134	2.2(-1)	Xe-133	2.5(+1)
Cs-136	1.1(-1)	Xe-135m	8.2(-1)
Cs-137	1.1(+0)	Xe-135	3.1(+0)
Te-132	2.6(-2)	Xe-137	1.7(-1)

Ba-140	4.5(-4)	Xe-138	7.1(-1)
La-140	1.4(-4)		
Ce-144	4.4(-5)		

5. Letdown Degasifier Regenerative Heat Exchanger, Shell Side
Letdown Degasifier, Liquid Region
Letdown Degasifier Recirculation Pump
and
Letdown Degasifier Trim Cooler

Isotope	µCi/gm	Isotope	µCi/gm
I-131	2.5(-1)	Mo-99	3.3(-1)
I-132	9.1(-2)	Cs-134	2.2(-1)
I-133	4.0(-2)	Cs-136	1.1(-1)
I-134	5.8(-2)	Cs-137	1.1(+0)
I-135	2.2(-1)	Te-132	2.6(-2)
		Ba-140	4.5(-4)
Sr-89	4.1(-4)	La-140	1.4(-4)
Sr-90	1.8(-5)	Ce-144	4.4(-5)
Sr-91	3.1(-3)		
Y-90	2.2(-5)	Mn-54	7.9(-5)
Y-91	5.8(-4)	Mn-56	3.0(-3)
Y-92	1.0(-4)	Co-58	2.6(-3)
Zr-95	6.7(-5)	Co-60	7.7(-5)
Nb-95	6.8(-5)	Fe-59	1.1(-4)
		Cr-51	9.5(-5)

6. Letdown Degasifier, Vapor Region

Isotope	µCi/cc
I-131	1.2(-1)
I-132	3.8(-2)
I-133	1.9(-1)
I-134	2.0(-2)
I-135	1.0(-1)

Kr-83m	2.4(+1)
Kr-85m	1.0(+2)
Kr-85	8.4(+0)
Kr-87	6.6(+1)
Kr-88	2.0(+2)
Xe-131m	4.3(+0)
Xe-133m	3.6(+1)
Xe-133	1.6(+3)
Xe-135m	2.4(+1)
Xe-135	1.9(+2)
Xe-137	1.8(+0)
Xe-138	1.9(+1)

7. Volume Control Tank, Liquid

   and

   Charging Pumps

   Same as reactor coolant, less noble gases.

8. Volume Control Tank, Vapor

Isotope	µCi/cc	Isotope	µCi/cc
I-131	9.8(-1)	Kr-87	5.1(+0)
I-132	4.6(-2)	Kr-88	2.7(+1)
I-133	9.4(-1)		
I-134	1.2(-2)	Xe-131m	3.7(+0)
I-135	2.6(-1)	Xe-133m	2.5(+1)
		Xe-133	1.3(+3)
Kr-83m	2.4(+0)	Xe-135m	1.1(+1)
Kr-85m	2.0(+1)	Xe-135	8.5(+1)
Kr-85	7.4(+0)	Xe-137	3.6(-2)
		Xe-138	5.5(-1)

9. Seal Water Heat Exchanger

   See Table 11.1-1

10. Moderating Heat Exchanger, Shell Side

Isotope	µCi/gm	Isotope	µCi/gm	Isotope	µCi/gm
Cr-51	9.5-5	Mo-99	3.3-1	Kr-83m	4.3-1
Mn-54	7.9-5	Te-132	2.6-2	Kr-85m	1.7+0
Mn-56	3.0-3	Cs-134	2.2-1	Kr-85	1.3-1
Fe-59	1.1-4	Cs-136	1.1-1	Kr-87	1.3+0
Co-58	2.6-3	Cs-137	1.1+0	Kr-88	3.4+0
Co-60	7.7-5	Ba-140	4.5-4	Xe-131m	6.7-2
		La-140	1.4-4	Xe-133m	5.7-1
Sr-89	4.1-4	Ce-144	4.4-5	Xe-133	2.5+1
Sr-90	1.8-5	I-131	3.4+2	Xe-135m	8.2-1
Sr-91	3.1-3	I-132	3.4+1	Xe-135	3.1+0
Y-90	2.2-5	I-133	4.6+2	Xe-137	1.7-1
Y-91	5.8-4	I-134	8.4+0	Xe-138	7.1-1
Y-92	1.0-4	I-135	1.8+2		
Zr-95	6.7-5				
Nb-95	6.8-5				

11. Letdown Reheat Heat Exchanger, Tube Side

Same as reactor coolant, see Table 11.1-1

12. Thermal Regeneration Demineralizer

Iodine	Curies	µCi/cc
I-131	8.0(+1)	1.7(+2)
I-132	8.0(+0)	1.7(+1)
I-133	1.1(+2)	2.3(+2)
I-134	2.0(+0)	4.2(+0)
I-135	4.1(+1)	8.8(+1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-6 GEOMETRY OF EQUIPMENT IN THE CHEMICAL AND VOLUME CONTROL SYSTEM

Regenerative Heat Exchanger	
- Horizontal Sections (2)	
Diameter	6 in.
Length	15 ft
- Vertical Section (1)	
Diameter	10 in.
Length	3 ft 7 in.
Excess Letdown Heat Exchanger	
Diameter	8 in.
Length	13 ft^3
Volume of source fluid	1.15 ft
Letdown Heat Exchanger	
Diameter	22 in.
Length	15 ft
Volume of source fluid	75 gal.
Mixed Bed Demineralizer	
Diameter	32 in.
Height of resin	65 in.
Cation Bed Demineralizer	
Diameter	32 in.
Height of resin	65 in.
Letdown Degasifier Regen. Heat Exchanger	
Diameter	11 in.
Length	15 ft 4 in.
Degasifier Preheater	
Diameter	8 in.
Length	15 ft 4 in.
Volume source fluid	21 ft^3

Letdown Degasifier		
Upper Part:	Diameter	24 in.
	Length	12 ft 4 in.
Lower Part:	Diameter	36 in.
	Length	6 ft 11 in.
Letdown Degasifier Recirculation Pump		
	Diameter	8 in.
	Length	2 in.
Letdown Degasifier Trim Cooler		
	Diameter	11 in.
	Length	15 ft 4 in.
	Volume source fluid	3.6 ft^3
Volume Control Tank		
	Diameter	108 in.
	Height	144 in.
	Vapor Volume	420 ft^3
Charging Pumps		
- Centrifugal		
	Diameter	14 in.
	Length	48 in.
- Reciprocal		
	Diameter	4 in.
	Length	43 in.
Seal Water Heat Exchanger		
	Diameter	14 in.
	Length	13 ft 9 in.
	Volume of source fluid	14.7 ft^3
Moderating Heat Exchanger		
	Diameter	18 in.
	Length	18 ft 2 in.
	Shell Side Volume	17.2 ft^3

Letdown Chiller Heat Exchanger	
Diameter	20 in.
Length	205 in.
Volume source fluid	9.5 ft^3
Letdown Reheat Heat Exchanger	
Diameter	8 5/8 in.
Length	87 in.
Thermal Regeneration Demineralizers	
Diameter	48 in.
Height of resin	72 in.
Filters	
Diameter	7 in.
Length	28 in.

TABLE 12.2-7 SOURCE TERMS FOR RESIDUAL HEAT REMOVAL SYSTEM [HISTORICAL]

Isotope	µCi/gm	Isotope	µCi/gm
I-131	2.3+0	Zr-95	6.4-4
I-132	4.3-1	Nb-95	6.5-4
I-133	3.3+0	Mo-99	3.0+0
I-134	2.3-2	Cs-134	3.5-1
I-135	1.4+0	Cs-136	1.7-1
		Cs-137	1.7+0
Kr-83m	9.8-5	Te-132	2.4-1
Kr-85m	9.0-1	Ba-140	4.3-3
Kr-85	1.3-1	La-140	1.5-3
Kr-87	1.5-1	Ce-144	4.2-4
Kr-88	1.3+0		
		Mn-54	7.5-4
Xe-131m	6.6-2	Mn-56	9.6-3
Xe-133m	5.5-1	Co-58	2.5-2
Xe-133	2.5+1	Co-60	7.3-4
Xe-135m	4.5-1	Fe-59	1.0-3
Xe-135	2.8+0	Cr-51	9.0-4
Xe-137			
Xe-138	5.8-6		
Sr-89	3.9-3		
Sr-90	1.7-4		
Sr-91	2.2-2		
Y-90	2.1-4		
Y-91	5.6-3		
Y-92	4.4-4		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-8 GEOMETRY OF EQUIPMENT IN RHR SYSTEM

RHR Heat Exchanger:

Mass of source fluid	5400 lbs.
Length	285 inches
Diameter	44 inches

RHR Pump

Volume of source fluid	3.2 ft^3
Length	19 inches
Diameter	9.5 inches

TABLE 12.2-9 STEAM GENERATOR BLOWDOWN SYSTEM SOURCE TERMS [HISTORICAL]

System Input

Isotope	µCi/gm	Isotope	µCi/gm
I-131	1.2-2	Mo-99	1.5-2
I-132	4.2-3	Cs-134	2.0-3
I-133	1.9-2	Cs-136	1.0-3
I-134	2.7-3	Cs-137	1.0-2
I-135	1.0-2	Te-132	1.2-3
		Ba-140	2.1-5
Sr-89	1.9-5	La-140	6.5-6
Sr-90	8.3-7	Ce-144	2.0-6
Sr-91	1.4-4		
Y-90	1.0-6	Mn-54	3.7-6
Y-91	2.7-5	Mn-56	1.4-4
Y-92	4.6-6	Co-58	1.2-4
Zr-95	3.1-6	Mn-60	3.6-6
Nb-95	3.1-6	Fe-59	5.1-6
		Cr-51	4.4-6

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

1. Flash Tank

Isotope	Water (μCi/gm)	Steam (μCi/cc)	Isotope	Water (μCi/gm)	Steam (μCi/cc)
I-131	1.7-2	1.9-6	Mo-99	2.1-2	2.4-6
I-132	6.0-3	6.7-7	Cs-134	2.9-3	3.2-7
I-133	2.7-2	3.0-6	Cs-136	1.4-3	1.6-7
I-134	3.9-3	4.3-7	Cs-137	1.4-2	1.6-6
I-135	1.4-2	1.6-6	Te-132	1.7-3	1.9-7
			Ba-140	3.0-5	3.3-9
			La-140	9.3-6	1.0-9
Sr-89	2.7-5	3.0-9	Ce-144	2.9-6	3.2-10
Sr-90	1.2-6	1.3-10			
Sr-91	2.0-4	2.2-8	Mn-54	5.3-6	5.9-10
Y-90	1.4-6	1.6-10	Mn-56	2.0-4	2.2-8
Y-91	3.9-5	4.3-9	Co-58	1.7-4	1.9-8
Y-92	6.6-6	7.3-10	Co-60	5.1-6	5.7-10
Zr-95	4.4-6	4.9-10	Fe-59	7.3-6	8.1-10
Nb-95	4.4-6	4.9-10	Cr-51	6.3-6	7.0-10

2. Blowdown Heat Exchangers

See water above

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

3. Flash Steam Condenser/Cooler Flash Tank and Distillate Pumps

Isotope	µCi/gm	Isotope	µCi/gm
I-131	8.6(-4)	Mo-99	1.1(-3)
I-132	3.0(-4)	Cs-134	1.4(-4)
I-133	1.4(-3)	Cs-136	7.1(-5)
I-134	1.9(-4)	Cs-137	7.1(-4)
I-135	7.1(-4)	Te-132	8.6(-5)
		Ba-140	1.5(-6)
Sr-89	1.4(-6)	La-140	4.6(-7)
Sr-90	5.9(-8)	Ce-144	1.4(-7)
Sr-91	1.0(-5)		
Y-90	7.1(-8)	Mn-54	2.6(-7)
Y-91	1.9(-6)	Mn-56	1.0(-5)
Y-92	3.3(-7)	Co-58	8.6(-6)
Zr-95	2.2(-7)	Co-60	2.6(-7)
Nb-95	2.2(-7)	Fe-59	3.6(-7)
		Cr-51	3.1(-7)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

4. Evaporator Bottoms

Isotope	Ci	µCi/gm	Isotope	Ci	µCi/gm
Cr-51	6.3(-2)	1.3(-2)	Mo-99	2.5(+1)	5.0(+0)
Mn-54	1.2(-1)	2.4(-2)	I-131	5.4(+1)	1.1(+1)
Mn-56	8.6(-3)	1.8(-3)	Te-132	2.3(+0)	4.6(-1)
Co-58	2.9(+0)	5.9(-1)	I-132	2.5(+0)	5.1(-1)
Fe-59	1.0(-1)	2.0(-2)	I-133	9.3(+0)	1.9(+0)
Co-60	1.3(-1)	2.6(-2)	I-134	5.7(-2)	1.1(-2)
Sr-89	4.0(-1)	8.0(-2)	Cs-134	7.1(+1)	1.4(+1)
Sr-90	3.0(-2)	6.1(-3)	I-135	1.7(+0)	3.4(-1)
Y-90	3.0(-2)	6.2(-3)	Cs-136	7.6(+0)	1.5(+0)
SR-91	3.4(-2)	6.8(-3)	Cs-137	3.7(+2)	7.5(+1)
Y-91	6.1(-1)	1.2(-1)	Ba-140	1.5(-1)	3.1(-2)
Y-92	4.0(-4)	8.0(-5)	La-140	1.6(-1)	3.2(-2)
Zr-95	7.2(-2)	1.5(-2)	Ce-144	6.6(-2)	1.3(-2)
Nb-95	9.6(-2)	2.0(-2)			
			Total	5.5(+2)	1.1(+2)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

5. Evaporator Distillate

Isotope	µCi/cc	Isotope	µCi/cc
Cr-51	1.1(-9)	I-131	2.9(-5)
Mn-54	9.2(-10)	Te-132	3.0(-7)
Mn-56	3.4(-8)	I-132	1.0(-5)
Co-58	3.0(-8)	I-133	4.6(-5)
Fe-59	1.3(-9)	I-134	6.3(-6)
Co-60	8.9(-10)	Cs-134	5.1(-7)
		I-135	2.5(-5)
Sr-89	4.7(-9)	Cs-136	2.5(-7)
Sr-90	2.1(-10)	Cs-137	2.5(-6)
Y-90	2.5(-10)	Ba-140	5.2(-9)
Sr-91	3.6(-8)	La-140	1.6(-9)
Y-91	6.7(-9)	Ce-144	5.1(-10)
Y-92	1.1(-9)		
Zr-95	7.8(-10)	Xe-131m	3.6(-9)
Nb-95	7.9(-10)	Xe-133m	1.2(-7)
Mo-99	3.8(-6)	Xe-133	2.1(-6)
Xe-135m	1.4(-4)		
Xe-135	1.2(-5)		
		Total	2.8(-4)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

6. Evaporator Distillate Condensate

Isotope	µCi/gm	Isotope	µCi/gm
Cr-51	9.4(-7)	Mo-99	3.3(-3)
Mn-54	7.9(-7)	I-131	2.5(-2)
Mn-56	2.9(-5)	Te-132	2.6(-4)
Co-58	2.6(-5)	I-132	8.6(-3)
Fe-59	1.1(-6)	I-133	4.0(-2)
Co-60	7.6(-7)	I-134	5.4(-3)
Sr-89	4.0(-6)	Cs-134	4.4(-4)
Sr-90	1.8(-7)	I-135	2.2(-2)
Y-90	2.1(-7)	Cs-136	2.2(-4)
Sr-91	3.1(-5)	Cs-137	2.2(-3)
Y-91	5.8(-6)	Ba-140	4.5(-6)
Y-92	9.4(-7)	La-140	1.4(-6)
Zr-95	6.7(-7)	Ce-144	4.4(-7)
Nb-95	6.8(-7)		
		Total	1.1(-1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-10 GEOMETRY OF EQUIPMENT IN STEAM GENERATOR BLOWDOWN SYSTEM

Flash Tank	
Height (water/vapor)	45/104 inches
Diameter	36 inches
Steam/Water Mass Ratio	3/7
Flash Tank Bottoms Cooler	
Volume Source Fluid	25 gallons
Length	124 inches
Diameter	16 inches
Flash Steam Condenser/Cooler	
Volume Source Fluid	55 gallons
Length	136 inches
Diameter	24 inches
Flash Tank Distillate Pumps	
Diameter	8 inches
Length	2 inches
Blowdown Evaporator	
Volume of Source Liquid	174.8 ft^3
Diameter	78 inches
Height	112 inches

Bottoms Pump

Diameter	8 inches
Length	2 inches

Bottoms Cooler:

12 straight sections

11 curved sections

Straight section:

Volume Source Liquid	150 in.^{3}
Diameter	1.5 in.
Length	10 ft

Curved Section:

Diameter	1.25 in.
Radius of curve	4.5 in.

TABLE 12.2-11 SOURCE TERMS FOR BORON RECOVERY SYSTEM [HISTORICAL]

System Input

Isotope	µCi/gm	Isotope	µCi/gm
Cr-51	9.5(-5)	I-131	2.5(-1)
Mn-54	7.9(-5)	Te-132	2.6(-2)
Mn-56	3.0(-3)	I-132	9.1(-2)
Co-58	2.6(-3)	I-133	4.0(-1)
Fe-59	1.1(-4)	I-134	5.8(-2)
Co-60	7.7(-5)	Cs-134	2.2(-1)
I-135	2.2(-1)		
Sr-89	4.1(-4)	Cs-136	1.1(-1)
Sr-90	1.8(-5)	Cs-137	1.1(+0)
Y-90	2.2(-5)	Ba-140	4.5(-4)
Sr-91	3.1(-3)	La-140	1.4(-4)
Y-91	5.8(-4)	Ce-144	4.4(-5)
Y-92	1.0(-4)		
Zr-95	6.7(-5)		
Nb-95	6.8(-5)		
Mo-99	3.3(-1)		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Isotope	CRIE (Ci)	BWST (Ci)	Pump (μCi/gm)	Evaporator Bottom (Ci)	Bottom Pump & Cooler (μCi/gm)	Distillate Conc. (μCi/gm)	Distillate Condensate (μCi/gm)	Recovery Test Tk. (Ci)	Recovery Test Tk. Pump (μCi/gm)	Demin. (Ci)
Cr-51	5.0(-2)	4.6(-3)	9.5(-6)	4.3(-3)	7.5(-4)	6.3(-12)	5.4(-9)	3.7(-7)	Same as Evaporator Distillate Condensate Concentration	2.8(-6)
Mn-54	1.4(-1)	6.3(-3)	7.9(-6)	6.3(-3)	1.1(-3)	8.6(-12)	7.4(-9)	5.0(-7)		1.3(-5)
Mn-56	3.8(-1)	3.8(-2)	3.0(-4)	9.3(-4)	1.6(-4)	5.1(-11)	4.4(-8)	8.9(-7)		2.7(-7)
Co-58	2.5(+0)	1.7(-1)	2.6(-4)	1.7(-1)	2.9(-2)	2.3(-10)	2.0(-7)	1.4(-5)		1.9(-4)
Fe-59	7.8(-2)	6.4(-3)	1.1(-5)	6.1(-3)	1.1(-3)	8.7(-12)	7.5(-9)	5.1(-7)		5.2(-6)
Co-60	1.8(-1)	6.5(-3)	7.7(-6)	6.5(-3)	1.1(-3)	8.9(-12)	7.6(-9)	5.2(-7)		1.7(-5)
Sr-89	3.1(-1)	2.5(-2)	4.1(-5)	2.4(-2)	4.2(-3)	3.4(-11)	2.9(-8)	2.0(-6)		2.2(-5)
Sr-90	4.3(-2)	1.5(-3)	1.8(-6)	1.5(-3)	2.7(-4)	2.1(-12)	1.8(-9)	1.2(-7)		4.3(-6)
Y-90	4.4(-2)	1.6(-3)	2.2(-6)	1.6(-3)	2.8(-4)	2.2(-12)	1.9(-9)	1.3(-7)		4.3(-6)
Sr-91	6.1(-1)	6.1(-2)	3.1(-4)	5.6(-3)	9.9(-4)	8.3(-11)	7.1(-8)	3.3(-6)		3.6(-6)
Y-91	4.9(-1)	3.7(-2)	5.8(-5)	3.6(-2)	6.3(-3)	5.0(-11)	4.3(-8)	2.9(-6)		3.6(-5)
Y-92	1.5(-1)	1.5(-3)	1.0(-5)	4.9(-5)	8.7(-6)	2.0(-12)	1.7(-9)	4.5(-8)		1.9(-8)
Zr-95	6.0(-2)	4.3(-3)	6.7(-6)	4.1(-3)	7.3(-4)	5.8(-12)	5.0(-9)	3.4(-7)		4.5(-6)
Nb-95	8.4(-2)	5.2(-3)	6.8(-6)	5.1(-3)	9.1(-4)	7.1(-12)	6.1(-9)	4.2(-7)		6.9(-6)
Mo-99	8.3(+1)	8.3(+0)	3.3(-2)	4.2(+0)	7.4(-1)	1.1(-8)	9.7(-6)	6.2(-4)		1.9(-3)
I-131	0	7.8(+1)	2.5(-1)	6.1(+1)	1.1(+1)	1.1(-6)	9.2(-4)	6.1(-2)		2.6(-1)
Te-132	6.7(+0)	6.7(-1)	2.6(-3)	3.7(-1)	6.6(-2)	9.2(-10)	7.9(-7)	5.1(-5)		1.6(-4)
I-132	6.7(+0)	1.1(+1)	9.1(-2)	6.1(-1)	1.1(-1)	1.5(-7)	1.3(-4)	2.4(-3)		8.1(-4)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Isotope	CRIE (Ci)	BWST (Ci)	Pump (μCi/gm)	Evaporator Bottom (Ci)	Bottom Pump & Cooler (μCi/gm)	Distillate Conc. (μCi/gm)	Distillate Condensate (μCi/gm)	Recovery Test Tk. (Ci)	Recovery Test Tk. Pump (μCi/gm)	Demin. (Ci)
I-133	0	8.8(+1)	4.0(-1)	1.7(+1)	3.0(+0)	1.2(-6)	1.0(-3)	5.6(-2)		1.1(-1)
I-134	0	3.3(+0)	5.8(-2)	2.7(-2)	4.8(-3)	4.4(-8)	3.8(-5)	2.7(-4)		2.9(-5)
Cs-134	4.7(+2)	1.8(+1)	2.2(-2)	1.8(+1)	3.1(+0)	2.4(-8)	2.1(-5)	1.4(-3)		4.5(-2)
I-135	0	4.0(+1)	2.2(-1)	2.6(+0)	4.5(-1)	5.5(-7)	4.7(-4)	1.8(-2)		1.5(-2)
Cs-136	4.0(+1)	4.0(+0)	1.1(-2)	3.4(+0)	6.0(-1)	5.5(-9)	4.7(-6)	3.2(-4)		1.6(-3)
Cs-137	2.6(+3)	9.4(+1)	1.1(-1)	9.4(+1)	1.7(+1)	1.3(-7)	1.1(-4)	7.5(-3)		2.6(-1)
Ba-140	1.6(-1)	1.6(-2)	4.5(-5)	1.4(-2)	2.4(-3)	2.2(-11)	1.9(-8)	1.3(-6)		6.6(-6)
La-140	9.9(-2)	9.8(-3)	1.4(-5)	1.3(-2)	2.3(-3)	1.4(-11)	1.2(-8)	8.6(-7)		6.0(-6)
Ce-144	7.8(-2)	3.5(-3)	4.4(-6)	3.5(-3)	6.1(-4)	4.8(-12)	4.1(-9)	2.8(-7)		7.3(-6)
Xe-131m		1.3(-1)				1.9(-12)				
Xe-133m		1.5(-1)				4.5(-11)				
Xe-133		4.9(+0)				7.8(-10)				
Xe-135m		1.2(+1)				5.4(-0)				
Xe-135		7.5(+0)				3.6(-9)				
Total	3.2(+3)	3.7(+2)	1.1(+0)	2.0(+2)	3.6(+1)	3.2(-6)	2.7(-3)	1.6(-1)		6.9(-1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Recovery Evaporator Feed Concentrations

Isotope	µCi/gm	Isotope	µCi/gm
Cr-51	5.4(-6)	Mo-99	9.7(-3)
Mn-54	7.4(-6)	I-131	9.2(-2)
Mn-56	4.4(-5)	Te-132	7.9(-4)
Co-58	2.0(-4)	I-132	1.3(-2)
Fe-59	7.5(-6)	I-133	1.0(-l)
Co-60	7.6(-6)	I-134	3.3(-3)
Cs-134	2.1(-2)		
Sr-89	2.9(-5)	I-135	4.7(-2)
Sr-90	1.8(-6)	Cs-136	4.7(-3)
Y-90	1.9(-6)	Cs-137	1.1(-1)
Sr-91	7.1(-5)	Ba-140	1.9(-5)
Y-91	4.3(-5)	La-140	1.2(-5)
Y-92	1.5(-6)	Ce-144	4.1(-6)
Zr-95	5.0(-6)		
Nb-95	6.1(-6)		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

Recovery Test Tank Concentrations

Isotope	µCi/gm	Isotope	µCi/gm
Cr-51	5.4(-9)	Mo-99	9.1(-6)
Mn-54	7.4(-9)	I-131	9.0(-4)
Mn-56	1.3(-8)	Te-132	7.5(-7)
Co-58	2.0(-7)	I-132	3.5(-5)
Fe-59	7.5(-9)	I-133	8.2(-4)
Co-60	7.6(-9)	I-134	9.0(-6)
Sr-89	2.9(-8)	Cs-134	2.1(-5)
Sr-90	1.8(-9)	I-135	2.7(-4)
Y-90	1.9(-9)	Cs-136	4.6(-6)
Sr-91	4.8(-8)	Cs-137	1.1(-4)
Y-91	4.3(-8)	Ba-140	1.9(-8)
Y-92	6.3(-10)	La-140	1.3(-8)
Zr-95	5.1(-9)	Ce-144	4.1(-9)
Nb-95	6.1(-9)		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-12 GEOMETRY OF EQUIPMENT IN BORON RECOVERY SYSTEM

Cesium Removal Ion Exchanger

Diameter	48 in.
Height	7 ft^3
Volume of Resin	75 ft

Boron Waste Storage Tank

Diameter	32 ft
Height	38 ft 5 in.
Volume	225,000 Gal.

Recovery Evaporator Feed Pump

Diameter	6 in.
Length	2 in.

Recovery Evaporator	Lower Part	Upper Part	Vapor Space
Diameter	2 ft 6 in.	5 ft 6 in.	5 ft 6 in.
Height	13 ft 6 in.	3 ft 3 in.	8 ft 9 in.

Recovery Evaporator Bottoms Pump

Diameter	6 in.
Length	2 in.

Recovery Evaporator Bottoms Cooler

Straight Section (Qty 12)

Length	10 ft
Diameter	1.5 in.
Volume Source	150 in.^{3}

Curved Section (Qty 11)

Diameter	1.25 in.
Radius of Curve	4.5 in.

Recovery Evaporator Reboiler Pump

Volume	20 ft^3
Diameter	24 in.

Recovery Evaporator Reboiler

Diameter	36 in.
Height	14 ft
Volume Source Fluid	31 ft^3

Recovery Evaporator Distillate Condenser

Diameter	26 in.
Length	72 in.
Volume Source	16 ft^3

Recovery Evaporator Distillate Accumulator

Diameter	26 in.
Length	11 ft

Recovery Evaporator Distillate Pump

Diameter	6 in.
Length	2 in.

Recovery Evaporator Distillate Cooler

Length	11 ft 9 in.
Diameter	21 in.
Volume Source Liquid	8.7 ft^3

Recovery Test Tank Pump

Diameter	6 in.
Length	2 in.

Recovery Test Tank

Diameter	15 ft
Height	16 ft 6 in.
Volume	18,000 gallons

Recovery Demineralizer

Diameter	48 in.
Height	7 ft 8 in.
Volume of Resin	75 ft^3

Recovery System Filters

Diameter	7 in.
Height	28 in.

TABLE 12.2-13 SOURCE TERMS FOR THE PRIMARY DRAIN SYSTEM [HISTORICAL]

System Input

Same as reactor coolant concentrations (see Table 11.1-1).

1. Primary Drain Tank
   Primary Drain Tank Transfer Pump
   Primary Drain Tank Regenerative Heat Exchanger
   Primary Drain and Tank Degasifier Preheater

Same as reactor coolant concentrations (see Table 11.1-1).

2. Primary Drain Tank Degasifier
   Liquid and liquid-steam region: Same as reactor coolant concentrations less noble gases.
   Gas Region:

Isotope	µCi/cc
I-131	1.2(+0)
I-132	3.8(-1)
I-133	1.9(+0)
I-134	2.0(-1)
I-135	1.0(+0)
Kr-83m	2.4(+1)
Kr-85m	1.0(+2)
Kr-85	8.4(+0)
Kr-87	6.6(+1)
Kr-88	2.0(+2)
Xe-131m	4.3(+0)
Xe-133m	3.6(+1)
Xe-133	1.6(+3)
Xe-135m	2.4(+1)
Xe-135	1.9(+2)
Xe-137	1.8(+0)
Xe-138	1.9(+1)
Total	2.3(+3)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-14 GEOMETRY OF EQUIPMENT IN PRIMARY DRAIN SYSTEM

Primary Drain Tank (PDT)	
Diameter	11 ft
Height	147 in.
PDT Transfer Pump	
Diameter	6 in.
Length	2 in.
PDT Regenerative Heat Exchanger	
Diameter	11 in.
Length	15 ft 4 in.
PDT Degasifier Preheater	
Diameter	8 in.
Length	15 ft 4 in.
Volume Source Fluid	2.1 ft^3
PDT Degasifier	
Gas Region:	
Diameter	24 $in.^3$
Volume	7 ft
Liquid-Steam Region:	
Diameter	24 in.
Height	12 ft 4 in.
Liquid Region:	
Diameter	36 in.

Height	6 ft 11 in.
<u>PDT Degasifier Recirculation Pump</u>	
Diameter	8 in.
Length	2 in.
<u>PDT Degasifier Trim Cooler</u>	
Diameter	11 in.
Length	15 ft 4 in.
Volume Source Fluid	3.6 ft^3
<u>PDT Degasifier Prefilter</u>	
Diameter	7 in.
Length	28 in.

TABLE 12.2-15 SOURCE TERMS FOR SPENT RESIN SLUICE SYSTEM [HISTORICAL]

1. Spent Resin Sluice Tank and Spent Resin Transfer Pump

Isotope	Total (Ci)	Conc. (μCi/gm)
Cr-51	1.5(+1)	8.1(-1)
Mn-54	6.6(+1)	3.6(+0)
Mn-56	-	-
Co-58	9.9(+2)	5.4(+1)
Fe-59	2.8(+1)	1.5(+0)
Co-60	8.4(+1)	4.6(+0)
Sr-89	9.1(+1)	4.9(+0)
Sr-90	2.0(+1)	1.1(+0)
Y-90	2.0(+1)	1.1(+0)
Sr-91	-	-
Y-91	2.0(+1)	1.1(+0)
Y-92	-	-
Zr-95	2.4(+1)	1.3(+0)
Nb-95	1.3(+1)	7.1(-1)
Mo-99	1.3(+3)	7.1(+1)
I-131	1.2(+4)	6.5(+2)
Te-132	5.3(+2)	2.9(+1)
I-132	5.3(+2)	2.9(+1)
I-133	-	-
I-134	-	-
Cs-134	4.7(+3)	2.6(+2)
I-135	-	-
Cs-136	1.7(+2)	9.2(+0)
Cs-137	2.7(+4)	1.5(+3)

Ba-140	3.1(+1)	1.7(+0)
La-140	3.1(+1)	1.7(+0)
Ce-144	3.6(+1)	2.0(+0)
Total	4.8(+4)	2.6(+3)

2. Spent Resin Sluice Pump

Concentrations are 10^{-4} that of the sluice tank

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-16 GEOMETRY OF EQUIPMENT IN SPENT RESIN SLUICE SYSTEM

Spent Resin Sluice Tank	
Diameter	11 ft
Height	16 ft
Volume of Resin	650 ft^3
Spent Resin Transfer Pump	
Diameter	6 in.
Length	72 in.
Volume	0.6 ft^3
Spent Resin Sluice Pump	
Diameter	8 in.
Length	2 in.

TABLE 12.2-17 SOURCE TERM FOR SPENT FUEL POOL CLEANUP SYSTEM [HISTORICAL]

Spent Fuel Pool Demineralizer

Isotope	µCi/cc	Isotope	µCi/cc
Cr-51	4.62-4	Mo-99	8.65-2
Mn-54	5.16-4	Te-132	1.09-2
Mn-56	-	Cs-134	4.01+0
Fe-59	6.06-4	Cs-136	1.02+0
Co-58	1.54-2	Cs-137	2.03+1
Co-60	5.16-4	Ba-140	1.50-3
		La-140	1.71-3
Sr-89	2.31-3	Ce-144	2.87-4
Sr-90	1.21-4	I-131	5.51-1
Sr-91	-	I-132	1.12-2
Y-90	1.22-4	I-133	8.28-5
Y-91	3.47-3	I-134	-
Y-92	-	I-135	-
Zr-95	3.93-4		
Nb-95m	3.76-4		

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-18 GEOMETRY OF EQUIPMENT IN SPENT FUEL POOL CLEANUP SYSTEM

Spent Fuel Pool Demineralizer

Diameter	48.0 in.
Height	72.0 in.
Volume	75.0 ft^3

Spent Fuel Pool Demineralizer Prefilter

And

Spent Fuel Pool Demineralizer Postfilter

Diameter	7 in.
Height	28 in.

TABLE 12.2-19 SOURCE TERMS FOR MISCELLANEOUS CHEMICAL DRAIN SYSTEM [HISTORICAL]

System Input:

Isotope	µCi/gm	Isotope	µCi/gm
		I-131	1.1(-l)
Cr-51	4.2(-5)	Mo-99	1.5(-l)
Mn-54	3.5(-5)	Te-132	1.1(-2)
Co-58	1.1(-3)	I-132	4.0(-2)
Fe-59	4.8(-5)	Cs-134	1.9(-2)
Co-60	3.4(-5)	Cs-136	9.7(-3)
		Cs-137	9.7(-2)
Sr-89	1.8(-4)	Ba-140	2.0(-4)
Sr-90	7.9(-6)	La-140	6.2(-5)
Y-90	9.7(-6)	Ce-144	1.9(-5)
Y-91	2.6(-4)		
Zr-95	2.9(-5)		
Nb-95	3.0(-5)		

Note: Short-lived Isotopes (T½ <l day) are neglected unless they are the daughter of a long-lived isotope.

1. Chemical Drain Tank

Isotope	Ci	µCi/gm	Isotope	Ci	µCi/gm
Cr-51	1.4(-4)	3.7(-5)	Mo-99	2.1(-l)	5.6(-2)
Mn-54	1.3(-4)	3.5(-5)	I-131	2.8E(-l)	7.5(-2)
Co-58	4.0(-3)	1.0(-3)	Te-132	1.8(-2)	4.6(-3)
Fe-59	1.7(-4)	4.4(-5)	I-132	2.0(-2)	5.2(-3)
Co-60	1.3(-4)	3.4(-5)	Cs-134	7.1(-2)	1.9(-2)
			Cs-136	2.9(-2)	7.6(-3)
Sr-89	6.4(-4)	1.7(-4)	Cs-137	3.7(-l)	9.7(-2)
Sr-90	3.0(-5)	7.9(-6)	Ba-140	5.9(-4)	1.6(-4)
Y-90	3.2(-5)	8.5(-6)	La-140	5.2(-4)	1.4(-4)

Y-91	9.3(-4)	2.4(-4)	Ce-144	7.1(-5)	1.9(-5)
Zr-95	1.0(-4)	2.7(-5)			
Nb-95	1.1(-4)	3.0(-5)	Total	1.0(+0)	2.6(-l)

2. Chemical Drain Transfer Pump

See specific source term for Chemical Drain Tank

3. Chemical Drain Treatment Tank and Chemical Drain Treatment Pump

Isotope	Ci	µCi/gm	Isotope	Ci	µCi/gm
Cr-51	3.5(-4)	2.4(-5)	Mo-99	8.7(-2)	5.9(-3)
Mn-54	4.9(-4)	3.3(-5)	I-131	3.3(-1)	2.2(-2)
Co-58	1.2(-2)	8.3(-4)	Te-132	8.4(-3)	5.7(-4)
Fe-59	4.9(-4)	3.3(-5)	I-132	8.7(-3)	5.9(-4)
Co-60	5.0(-4)	3.4(-5)	Cs-134	2.8(-l)	1.9(-2)
			Cs-136	4.8(-2)	3.3(-3)
Sr-89	2.0(-3)	1.3(-4)	Cs-137	1.4(+0)	9.7(-2)
Sr-90	1.2(-4)	7.9(-6)	Ba-140	1.0(-3)	6.8(-5)
Y-90	1.2(-4)	7.9(-6)	La-140	1.1(-3)	7.5(-5)
Y-91	2.9(-3)	1.9(-4)	Ce-144	2.7(-4)	1.8(-5)
Zr-95	3.3(-4)	2.2(-5)			
Nb-95	4.1(-4)	2.8(-5)	Total	2.2(+0)	1.5(-l)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-20 EQUIPMENT GEOMETRY FOR THE MISCELLANEOUS CHEMICAL DRAIN SYSTEM

Chemical Drain Tank	
Diameter	5 ft
Height	8 ft 8 in.
Chemical Drain Transfer Pump	
Diameter	8 in.
Length	2 in.
Chemical Drain Treatment Tank	
Diameter	8 ft
Height	10 ft 9 in.
Chemical Drain Treatment Pump	
Diameter	6 in.
Length	2 in.

TABLE 12.2-21 MAXIMUM EXPECTED RADIONUCLIDE CONTENT OF A REACTOR MAKEUP WATER STORAGE TANK [HISTORICAL]

Radionuclide	Tank Inventory (Curies)	Radionuclide	Tank Inventory (Curies)
I-131	1.1-1	Mo-99	2.2+0
1-132	3.8-2	Cs-134	1.1-1
I-133	1.7-1	Cs-136	5.9-2
I-134	2.3-2	Cs-137	5.5-1
I-135	9.3-2	Te-132	1.1-2
		Ba-140	5.9-3
Sr-89	1.6-3	La-140	1.4-5
Sr-90	5.5-5	Ce-144	1.4-5
Sr-91	8.1-4		
Y-90	6.4-5	Mn-54	3.3-5
Y-91	2.5-3	Mn-56	1.2-3
Y-92	3.1-4	Co-58	1.1-3
Zr-95	3.0-5	Co-60	3.1-5
Nb-95	2.9-5	Fe-59	4.2-5
		Cr-51	3.9-5

TOTAL = 3.4 Curies

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-22 MAXIMUM EXPECTED RADIONUCLIDE CONTENT OF A REFUELING WATER STORAGE TANK [HISTORICAL]

Radionuclide	Tank Inventory (Curies)	Radionuclide	Tank Inventory (Curies)
I-131	7.7+1	Mo-99	6.0+0
I-132	<1.0-7	Cs-134	4.2+1
I-133	7.4-5	Cs-136	8.0+0
I-134	<1.0-7	Cs-137	2.2+2
I-135	1.0-7	Te-132	6.3-1
		Ba-140	7.8-2
Sr-89	4.8-1	La-140	1.4-5
Sr-90	2.2-2	Ce-144	5.2-2
Sr-91	<1.0-7		
Y-90	1.4-4	Mn-54	1.2-1
Y-91	7.8-1	Mn-56	<1.0-7
Y-92	<1.0-7	Co-58	3.4+0
Zr-95	9.5-2	Co-60	1.2-1
Nb-95	7.7-2	Fe-59	1.2-1
		Cr-51	9.3-2
		TOTAL =	359 Curies

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-23 GEOMETRY OF WATER STORAGE TANKS

Reactor Makeup Water Storage Tank

Diameter	26 ft
Height	29 ft 3 in.

Refueling Water Storage Tank

Diameter	44 ft
Height	43 ft

TABLE 12.2-24 LIQUID WASTE SYSTEM SOURCE TERMS [HISTORICAL]

Isotope	Feed (µCi/gm)	Floor Drain Tank (Ci)	Floor Drain Tank Pump (µCi/gm)	Waste Evap Bottom (Ci)	Waste Evap Distillate (µCi/cc)	Distillate Condensate (µCi/gm)	Waste Test Tank (Ci)	Waste Test Tank Concentration (µCi/gm)	Waste Demin. (Ci)	Evap. Bottom Pump (µCi/gm)
Cr-51	7.1(-5)	2.7(-3)	9.5(-5)	3.8(-2)	8.3(-12)	7.1(-9)	6.7(-7)	7.1(-9)	3.0(-6)	6.7(-3)
Mn-54	5.9(-5)	2.2(-3)	7.9(-5)	3.3(-2)	6.8(-12)	5.8(-9)	5.5(-7)	5.8(-9)	1.3(-5)	5.8(-3)
Mn-56 (1)										
Co-58	2.0(-3)	7.6(-2)	2.6(-3)	1.1(+0)	2.3(-10)	2.0(-7)	1.9(-5)	2.0(-7)	2.0(-4)	1.9(-1)
Fe-59	8.3(-5)	3.1(-3)	1.1(-4)	4.5(-2)	9.6(-12)	8.2(-9)	7.7(-7)	8.1(-9)	5.5(-6)	7.9(-3)
Co-60	5.8(-5)	2.2(-3)	7.7(-5)	3.3(-2)	6.8(-12)	5.8(-9)	5.5(-7)	5.8(-9)	1.7(-5)	5.8(-3)
Sr-89	3.1(-4)	1.2(-2)	4.1(-4)	1.8(-1)	3.7(-11)	3.2(-8)	3.0(-6)	3.2(-8)	2.4(-5)	3.2(-2)
Sr-90	1.4(-5)	5.3(-4)	1.8(-5)	8.0(-3)	1.6(-12)	1.4(-9)	1.3(-7)	1.4(-9)	4.2(-6)	1.4(-3)
Y-90	1.7(-5)	5.3(-4)	1.8(-5)	8.0(-3)	1.6(-12)	1.7(-9)	1.3(-7)	1.4(-9)	4.2(-6)	1.6(-3)
Sr-91(1)										
Y-91	4.4(-4)	1.7(-2)	5.8(-4)	2.5(-1)	5.3(-11)	4.5(-8)	4.3(-6)	4.5(-8)	3.9(-5)	4.4(-2)
Y-92(1)										
Zr-95	5.0(-5)	1.9(-3)	6.7(-5)	2.8(-2)	5.8(-12)	5.0(-9)	4.7(-7)	5.0(-9)	4.7(-6)	4.9(-3)

(1) Those isotopes with half-life less than one day are neglected unless they are decay products of long-lived isotopes

(2) For the above decay product, parent activities are used for source term assuming secular equilibrium is achieved.

Isotope	Feed (µCi/gm)	Floor Drain Tank (Ci)	Floor Drain Tank Pump (µCi/gm)	Waste Evap Bottom (Ci)	Waste Evap Distillate (µCi/cc)	Distillate Condensate (µCi/gm)	Waste Test Tank (Ci)	Waste Test Tank Concentration (µCi/gm)	Waste Demin. (Ci)	Evap. Bottom Pump (µCi/gm)
Nb-95	5.1(-5)	1.9(-3)	6.8(-5)	2.8(-2)	5.8(-12)	5.0(-9)	4.7(-7)	5.0(-9)	7.1(-6)	4.9(-3)
Mo-99	2.5(-1)	9.4(+0)	3.3(-1)	8.8(+1)	2.9(-8)	2.5(-5)	2.2(-3)	2.3(-5)	9.6(-4)	1.5(+1)
I-131	1.9(-1)	7.2(+0)	2.5(-1)	9.1(+1)	2.2(-7)	1.9(-4)	1.7(-2)	1.8(-4)	2.2(-2)	1.6(+1)
Te-132	2.0(-2)	7.5(-1)	2.6(-2)	7.5(+0)	2.3(-9)	2.0(-6)	1.8(-4)	1.9(-6)	9.3(-5)	1.3(+0)
I-132	6.8(-2)	7.5(-1)	2.6(-2)	7.5(+0)	6.8(-8)	5.8(-5)	1.8(-4)	1.9(-6)	9.3(-5)	1.5(+1)
I-133(1)										
I-134(1)										
Cs-134	3.3(-2)	1.3(+0)	4.4(-2)	1.9(+1)	4.0(-9)	3.4(-6)	3.2(-4)	3.4(-6)	9.1(-3)	3.3(+0)
I-135(1)										
Cs-136	1.7(-2)	6.4(-1)	2.2(-2)	8.7(+0)	2.0(-9)	1.7(-6)	1.6(-4)	1.7(-6)	3.3(-4)	1.5(+0)
Cs-137	1.7(-1)	6.5(+0)	2.2(-1)	9.7(+1)	2.0(-8)	1.7(-5)	1.6(-3)	1.7(-5)	5.1(-2)	1.7(+1)
Ba-140	3.4(-4)	1.3(-2)	4.5(-4)	1.7(-1)	4.0(-11)	3.4(-8)	3.2(-6)	3.4(-8)	6.5(-6)	3.0(-2)
La-140	1.7(-4)	4.3(-3)	1.4(-4)	1.2(-1)	1.3(-11)	1.1(-8)	1.3(-6)	1.4(-8)	6.9(-6)	2.1(-2)
Ce-144	3.3(-5)	1.3(-3)	4.4(-5)	1.9(-2)	4.0(-12)	3.4(-9)	3.2(-7)	3.4(-9)	7.4(-6)	3.3(-3)
Xe-131m					4.0(-12)					
Xe-133m					1.3(-10)					

Isotope	Feed (µCi/gm)	Floor Drain Tank (Ci)	Floor Drain Tank Pump (µCi/gm)	Waste Evap Bottom (Ci)	Waste Evap Distillate (µCi/cc)	Distillate Condensate (µCi/gm)	Waste Test Tank (Ci)	Waste Test Tank Concentration (µCi/gm)	Waste Demin. (Ci)	Evap. Bottom Pump (µCi/gm)
Xe-133					2.3(-9)					
Xe-135m					1.8(-7)					
Xe-135					1.2(-8)					
TOTAL		2.7(+1)	9.2(-1)	3.2(+2)	5.4(-7)	3.0(-4)	2.2(-2)	2.3(-4)	8.4(-2)	6.9(+1)

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-25 GEOMETRY OF EQUIPMENT IN THE LIQUID WASTE SYSTEM

Floor Drain Tank

Diameter	11 ft
Height	15 ft 1 in.

Floor Drain Tank Pumps

Diameter	2½ in.
Height	13 in.

Waste Evaporator	Lower Part	Upper Part	Vapor Space
Diameter	2 ft 6 in.	5 ft 6 in.	5 ft 6 in.
Height	13 ft 6 in.	3 ft 3 in.	8 ft 9 in.

Waste Evaporator Distillate Condenser

Diameter	2 ft 2 in.
Height	6 ft

Waste Evaporator Distillate Accumulator

Diameter	2 ft 2 in.
Height	11 ft

Waste Evaporator Distillate Pump

Diameter	2½ in.
Height	13 in.

Waste Evaporator Distillate Cooler

Diameter	1 ft 9 in.
Height	10 ft

Waste Test Tank (WTT)

Diameter	11 ft
Height	35 ft

WTT Pumps

Diameter	2½ in.
Height	13 in.

Waste Demineralizer

Diameter	4 ft
Height	6 ft

Waste Evaporator Reboiler

Diameter	36 in.
Height	14 ft
Volume Source Fluid	31 ft^3

Waste Evaporator Reboiler Pump

Volume	20 ft^3
Diameter	24 in.

Waste Evaporator Bottoms Pump

Diameter	2½ in.
Height	13 in.

Waste Evaporator Bottoms Cooler
Straight Section (Qty-12)

Length	10 ft
Diameter	1½ in.
Volume of Source Fluid	150 in.

Curved Section (Qty-11)

Diameter	1¼ in.
Radius of Curve	4½ in.

TABLE 12.2-26 RADIONUCLIDE CONCENTRATIONS IN WASTE GAS STREAMS [HISTORICAL]

	Concentration (μCi/cc)		
ISOTOPE	LTDN DEGASIFIER	PDT DEGASIFIER	RGWS INPUT
Kr-83m	2.4+1*[*]	2.4+1*	2.4+1*
Kr-85m	1.0+2	1.0+2	1.0+2
Kr-85	8.4+0	8.4+0	8.4+0
Kr-87	6.6+1	6.6+1	6.6+1
Kr-88	2.0+2	1.0+2	1.0+2
Xe-131m	4.3+0	4.3+0	4.3+0
Xe-133m	3.6+1	3.6+1	3.6+1
Xe-133	1.6+3	1.6+3	1.6+3
Xe-135m	2.4+1	2.4+1	2.4+1
Xe-135	1.9+2	1.9+2	1.9+2
Xe-137	1.8+0	1.8+0	1.8+0
Xe-138	1.9+1	1.9+1	1.9+1
I-131	1.2-1	1.2+0	4.8-1
I-132	3.8-2	3.8-1	1.5-1
I-133	1.9-1	1.9+0	7.6-1
I-134	2.0-2	2.0-1	8.0-2
I-135	1.0-1	1.0+0	4.0-1

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

[*] 2.4+1 = 2.4×10^{1}

TABLE 12.2-27 RADIOACTIVE SOURCE TERMS FOR GAS WASTE SYSTEM [HISTORICAL]

Radionuclide Inventory *(Ci)

Isotope	I-Guard Bed Filter	Chiller	Dryer	H_2 Surge Tank	Hepa
Kr-83m	6.0-3	5.5-2	5.0-1	-	-
Kr-85m	2.5-2	2.3-1	2.1+0	-	-
Kr-85	2.1-3	1.9-2	1.7-1	1.4+2	2.3+0
Kr-87	1.7-2	1.5-1	1.4+0	-	-
Kr-88	4.5-2	4.4-1	4.0+0	-	-
Xe-131m	1.1-3	9.9-3	9.0-2	2.2+0	7.9-1
Xe-133m	9.0-3	8.2-2	7.5-1	-	-
Xe-133	4.0-1	3.8+0	3.4+1	1.1+1	5.8+0
Xe-135m	6.0-3	5.5-2	5.0-1	-	-
Xe-135	4.8-2	4.4-1	4.0+0	-	-
Xe-137	4.5-4	4.1-3	3.7-2	-	-
Xe-138	4.8-3	4.4-2	4.0-1	-	-
I-131	4.5+1	2.7-4	2.5-3	-	-
I-132	1.7-1	8.5-5	7.8-4	-	-
I-133	7.8+0	4.4-4	4.0-3	-	-
I-134	3.3-2	4.4-5	4.0-4	-	-
I-135	1.3+0	2.3-4	2.1-3	-	-

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* Source Terms apply to a single component.

Radionuclide Inventory (Ci)

Isotope	1st Bed	2nd Bed	3rd Bed	4th Bed	5th Bed
Kr-83m	8.2+1	1.5-1	--	--	--
Kr-85m	7.5+2	5.2+1	3.5+0	2.4-1	1.7-2
Kr-85	1.8+2	1.8+2	1.8+2	1.8+2	1.8+2
Kr-87	1.5+2	1.4-2	<1.0-2	--	--
Kr-88	1.0+3	1.5+1	2.2-1	<1.0-2	--
Xe-131m	1.1+3	5.7+2	2.8+2	1.4+2	7.0+1
Xe-133m	3.5+3	8.8+1	2.2+0	5.6-2	<1.0-2
Xe-133	3.0+5	6.1+4	1.3+4	2.6+3	5.4+2
Xe-135m	1.2+1	--	--	--	--
Xe-135	3.2+3	--	--	--	--
Xe-137	2.1-1	--	--	--	--
Xe-138	8.3+0	--	--	--	--
I-131	1.6+2	1.0+1	6.4-1	4.0-2	<1.0-2
I-132	5.9-1	3.7-2	<1.0-2	--	--
I-133	2.7+1	1.7+0	1.1-1	<1.0-2	--
I-134	1.2-1	<1.0-2	--	--	--
I-135	4.6+0	2.9-1	1.8-2	<1.0-2	--

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-28 GEOMETRY OF EQUIPMENT IN RADIOACTIVE WASTE GAS SYSTEM

Waste Gas Chiller

Diameter	3 in.
Length	34 in.

Waste Gas Dryer

Diameter	6 in.
Height	53 in.

Iodine Guard Bed

Diameter	2 ¾ in.
Height	13 in.

Carbon Delay Bed

Diameter	36 in.
Height	72 in.

Hydrogen Surge Tank

Diameter	36 in.
Height	72 in.

Regenerative Compressor

Diameter	24 in.
Height	6 in.
Gas Volume	85 $in.^3$

HEPA Filter

Diameter	2 ¾ in.
Height	11½ in.

TABLE 12.2-29 SHIELDING SOURCE TERMS FOR EQUIPMENT IN SOLID WASTE MANAGEMENT SYSTEM [HISTORICAL]

Nuclide	Shielding Source Terms (µCi/gm)				
	(a)	(b)	(c)	(d)	(e)
I-131	1.2+01*	1.0+01	6.5+02	2.2+03	3.5+01
I-132	4.7-01	4.1-01	2.9+01	9.7+01	1.4+00
Sr-89	3.0-02	2.6-03	4.9+00	1.6+01	8.8-02
Sr-90	2.0-03	1.7-03	1.1+00	3.7+00	5.8-03
Y-90	2.1-03	1.8-03	1.1+00	3.7+00	6.1-03
Y-91	4.4-02	3.8-02	1.1+00	3.7+00	1.3-01
Zr-95	5.3-03	4.6-03	1.3+00	4.3+00	1.5-02
Nb-95	6.7-03	5.8-03	7.1-01	2.4+00	2.0-02
Mo-99	5.4+00	4.7+00	7.1+01	2.4+02	1.6+01
Cs-134	5.9+00	5.1+00	2.6+02	8.7+02	1.7+01
Cs-136	1.1+00	9.5-01	9.2+00	3.1+01	3.2+00
Cs-137	3.2+00	2.8+00	1.5+02	5.0+02	9.3+00
Te-132	4.7-01	4.1-01	2.9+01	9.7+01	1.4+00
Ba-140	1.6-02	1.4-02	1.7+00	5.7+00	4.7-02
La-140	1.4-02	1.2-02	1.7+00	5.7+00	4.1-02
Ce-144	4.4-03	3.8-03	2.0+00	6.7+00	1.3-02
Mn-54	8.0-03	6.9-03	3.6+00	1.2+01	2.3-02
Co-58	2.1-01	1.8-01	5.4+01	1.8+02	6.1-01
Co-60	8.5-03	7.4-03	4.6+00	1.5+01	2.5-02
Fe-59	7.5-03	6.5-03	1.5+00	5.0+00	2.2-02
Cr-51	5.3-03	4.6-03	8.1-01	2.7+00	1.5-02

(a) Waste Concentrates Tank and Waste Conc. Transfer Pump
(b) Waste Feed Tanks and Waste Feed Recir. Pump
(c) Spent Resin Transfer Pump, Spent Resin Hopper, Spent Resin Recir. Pump, and Resin Centrifuge Metering Pump
(d) Spent Resin Centrifuge
(e) Waste Crystallizer/Evaporator, Crystallizer Recir. Pump, and Crystallizer Drain Pump
* 1.2+01 = 1.2×10^{1}

Nuclide	Shielding Source Terms (μCi/gm)				
	(f)	(g)	(h)	(i)	(j)
I-131	3.5+01	1.0-04	1.6-01	1.0-04	1.1+03
I-132	1.4+00	3.9-06	7.3-03	3.9-06	4.9+01
Sr-89	8.8-02	2.5-07	1.3-03	2.5-07	8.0+00
Sr-90	5.8-03	1.7-08	2.7-04	1.7-08	1.9+00
Y-90	6.1-03	1.8-08	2.7-04	1.8-08	1.9+00
Y-91	1.3-01	3.7-07	2.7-04	3.7-07	1.9+00
Zr-95	1.5-02	4.4-08	3.3-04	4.4-08	2.2+00
Nb-95	2.0-02	5.6-08	1.8-04	5.6-08	1.2+00
Mo-99	1.6+01	4.5-05	1.8-02	4.5-05	1.2+02
Cs-134	1.7+01	4.9-05	6.5-02	4.9-05	4.4+02
Cs-136	3.2+00	9.2-06	2.4-03	9.2-06	1.6+01
Cs-137	9.3+00	2.7-05	3.7-02	2.7-05	2.5+02
Te-132	1.4+00	3.9-06	7.3-03	3.9-06	4.9+01
Ba-140	4.7-02	1.3-07	4.3-04	1.3-07	2.9+00
La-140	4.1-02	1.2-07	4.3-04	1.2-07	2.9+00
Ce-144	1.3-02	3.7-08	5.0-04	3.7-08	3.4+00
Mn-54	2.3-02	6.7-08	9.0-04	6.7-08	6.0+00
Co-58	6.1-01	1.8-06	1.4-02	1.8-06	9.0+01
Co-60	2.5-02	7.1-08	1.2-03	7.1-08	7.5+00
Fe-59	2.2-02	6.3-08	3.8-04	6.3-08	2.5+00
Cr-51	1.5-02	4.4-08	2.1-04	4.4-08	1.4+00

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

(f) Conc. Bottom Tank, Conc. Bottom Tank Recir. Pump, Waste Metering Pump, and Alternate Station Conc. Feed Pump
(g) Entrainment Separator, Crystallizer Condenser (Shell Side), Crystallizer Dist. Tank, Crystallizer Dist. Pump, and Crystallizer Subcooler (Shell Side)
(h) Spent Resin Dewatering Pump
(i) Crystallizer Reflux Pot and Crystallizer Reflux Pump
(j) Extruder

TABLE 12.2-30 GEOMETRY OF EQUIPMENT IN SOLID WASTE MANAGEMENT SYSTEM

Waste Concentrates Tank	
Diameter	10 ft
Height	11 ft 9 in.
Waste Concentrates Transfer Pump	
Diameter (internal)	4 in.
Length	17 in.
Waste Feed Tank(s)	
Diameter	5 ft
Height (including heads)	9 ft 10 in.
Waste Feed Recirculation Pump(s)	
Diameter (internal)	2 in.
Length	1 ft 6 in.
Spent Resin Transfer Pump	
Diameter (internal)	2 in.
Length	6 ft
Spent Resin Hopper	
Diameter	5 ft
Height (cylinder)	6 ft
Height (cone)	3 ft

Spent Resin Recirculation Pump

Diameter	3 in.
Length	3 ft 9 in.

Resin Centrifuge Metering Pump

Diameter (internal)	2 in.
Length	1 ft 9 in.

Spent Resin Centrifuge

Diameter of bowl	1 ft
Depth of bowl	1 ft 3 in.

Evaporator/Crystallizer Vapor Body

Diameter	3 ft
Height (cylinder)	9 ft 6 in.
Height (cone)	2 ft 9 in.

Crystallizer Recirculation Pump

Diameter (internal)	10 in.
Length	6 ft

Crystallizer Drain Pump

Diameter (internal)	10 in.
Length	3 ft

Concentrates Bottom Tank

Diameter	5 ft
Height	8 ft 6 in.

Concentrates Bottom Tank Recirculation Pump

Diameter (internal)	10 in.
Length	3 ft 6 in.

Waste Metering Pump

Diameter (internal)	3/4 in.
Length	1 ft 6 in.

Alternate Station Concentrates Feed Pump

Diameter (internal)	20 in.
Length	4 ft 6 in.

Entrainment Separator

Diameter	2 ft
Height (including heads)	12 ft 6 in.

Crystallizer Condenser

Diameter	13 in.
Height	11 ft 6 in.

Crystallizer Subcooler

Diameter	6 in.
Height	11 ft 6 in.

Spent Resin Dewatering Pump

Diameter (internal)	1 in.
Length	1 ft 3 in.

Crystallizer Reflux Pot

Diameter	1 ft 6 in.
Height (head to head)	29 in.

Crystallizer Reflux Pump

Diameter (internal)	1 in.
Length	1 ft

Evaporator/Extruder

Diameter (internal)	2 in.
Length	15 ft 9 in.

TABLE 12.2-31 CONTAINMENT AIRBORNE ISOTOPIC CONCENTRATIONS AT SHUTDOWN [HISTORICAL]

Volume = 2.70×10^6 ft^3

Ventilation Rate = 1000 cfm, continuous prior to shutdown

Isotope	Specific Activity (μCi/cc)	MPC Fraction
H-3	4.02-04*	8.03+01
Kr-85m	3.09-06	5.15-01
Kr-85	2.65-06	2.65-01
Kr-87	7.30-07	7.30-01
Kr-88	4.01-06	4.01+00
Xe-131m	9.22-06	6.08-02
Xe-133m	6.19-06	6.19-01
Xe-133	3.76-04	3.76+01
Xe-135m	1.82-06	1.82+00
Xe-135	1.55-05	3.88+00
Xe-138	7.63-08	7.62-02
I-131	4.11-08	4.57+00
I-132	8.98-10	4.49-03
I-133	2.55-09	8.51-02
I-134	2.30-10	4.61-04
I-135	5.77-09	5.76-02
TOTAL	8.13-04	1.35+02

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 4.02-04 = 4.02×10^{-4}

TABLE 12.2-32 CONTAINMENT AIRBORNE ISOTOPIC CONCENTRATIONS 20 HOURS AFTER SHUTDOWN [HISTORICAL]

Volume	=	2.70×10^6 ft^3
Ventilation Rate	=	1000 cfm, continuous prior to shutdown
Ventilation Rate	=	11,000 cfm, for 17 hours following shutdown, then 40,000 cfm, after 17 hours following shutdown
Recirculation Rate	=	4000 cfm, for 16 hours, with 90% removal of iodine species

Isotope	Specific Activity (μCi/cc)	***MPC Fraction***
H-3	3.41-06*	6.81-01
Kr-85m	1.12-09	1.87-04
Kr-85	2.25-08	2.25-03
Kr-87	1.09-13	1.09-07
Kr-88	2.36-10	2.36-04
Xe-131m	1.02-08	2.53-05
Xe-133m	4.10-10	4.10-03
Xe-133	2.87-06	2.87-01
Xe-135m	4.48-08	4.48-02
Xe-135	6.75-08	1.69-02
Xe-138	0.	0.
I-131	1.33-10	1.47-02
I-132	7.19-15	3.60-08
I-133	4.54-12	1.52-04
I-134	0.	0.
I-135	2.53-12	2.53-05
TOTAL	6.43-06	1.05+00

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 3.41-06 = 3.41×10^{-6}

TABLE 12.2-33 TURBINE BUILDING AIRBORNE ISOTOPIC CONCENTRATIONS – SUMMER [HISTORICAL]

Volume = 7.72×10^6 ft^3

Ventilation Rate = 1.00×10^6 cfm

Isotope	Specific Activity (μCi/cc)	***MPC Fraction***
Kr-85m	1.26-13*	2.11-08
Kr-85	9.74-15	9.73-10
Kr-87	8.91-14	8.90-08
Kr-88	2.43-13	2.43-07
Xe-131m	5.07-15	2.53-10
Xe-133m	4.15-14	4.15-09
Xe-133	1.88-12	1.88-07
Xe-135m	7.78-14	7.79-08
Xe-135	2.35-13	5.87-08
Xe-138	3.59-14	3.59-08
I-131	1.04-12	1.15-04
I-132	6.79-14	3.39-07
I-133	1.16-12	3.87-05
I-134	1.67-14	3.31-08
I-135	3.70-13	3.70-06
TOTAL	5.41-12	1.59-04

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 1.26-13 = 1.26×10^{-13}

TABLE 12.2-34 TURBINE BUILDING AIRBORNE ISOTOPIC CONCENTRATIONS – WINTER [HISTORICAL]

Volume = 7.18×10^6 ft^3

Ventilation Rate = 0 cfm

Isotope	Specific Activity (µCi/cc)	***MPC Fraction***
Kr-85m	4.82-12*	8.03-07
Kr-85	1.72-10	1.72-05
Kr-87	1.04-12	1.04-06
Kr-88	5.95-12	5.96-06
Xe-131m	2.24-11	1.12-06
Xe-133m	2.40-11	2.39-06
Xe-133	2.24-09	2.24-04
Xe-135m	6.55-12	6.54-06
Xe-135	3.96-11	9.89-06
Xe-138	1.03-13	1.08-07
I-131	1.70-09	1.89-01
I-132	1.38-12	6.88-06
I-133	2.05-13	6.84-03
I-134	1.38-13	2.77-07
I-135	2.13-11	2.13-04
TOTAL	4.44-09	1.96-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 4.82-12 = 4.82×10^{-12}

TABLE 12.2-35 PRIMARY AUXILIARY BUILDING AIRBORNE ISOTOPIC CONCENTRATIONS [HISTORICAL]

Volume = 1.45×10^5 ft^3

Ventilation Rate = 41,500 cfm

Isotope	Specific Activity (µCi/cc)	MPC Fraction
Kr-85m	2.61-08*	4.35-03
Kr-85	2.03-09	2.03-04
Kr-87	1.92-08	1.92-02
Kr-88	5.12-08	5.12-02
Xe-131m	1.04-09	5.22-05
Xe-133m	8.60-09	8.59-04
Xe-133	3.87-07	3.87-02
Xe-135m	1.24-08	1.24-02
Xe-135	4.80-08	1.20-02
Xe-138	8.89-09	8.89-03
I-131	2.90-10	3.23-02
I-132	1.03-10	5.17-04
I-133	4.60-11	1.53-03
I-134	6.48-11	1.30-04
I-135	2.51-10	2.51-03
TOTAL	5.65-07	1.85-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

* 2.61-08 = 2.61×10^{-8}

TABLE 12.2-36 REFUELING AIRBORNE TRITIUM CONCENTRATION CONTAINMENT REFUELING POOL [HISTORICAL]

Volume = 2.70×10^6 ft^3

Ventilation Rate = 31,000 cfm

Isotope	Specific Activity (µCi/cc)	MPC Fraction
H-3	2.09-06	4.19-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.2-37 REFUELING AIRBORNE TRITIUM CONCENTRATION SPENT FUEL POOL [HISTORICAL]

Volume = 263,000 ft^3

Ventilation Rate = 34,000 cfm

Isotope	Specific Activity (µCi/cc)	***MPC Fraction***
H-3	2.67-06	5.35-01

Note: The information presented in the above Table represents the radiation sources and associated input parameters, assumptions and methodology described in Section 12.2 used for the original shielding design and is retained here for historical purposes. Updating this analysis using an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle represents a minor change in the original design basis and will have no significant impact on shielding requirements and safe plant operations.

TABLE 12.3-1 RADIATION ZONING AND ACCESS CONTROL

Zone	Dose Rate Mrem/Hour	Allowed Occupancy
I	Less than 0.5	No Restriction on Access
II	0.5 - 2.5	Occupational Access
III	2.5 – 15	Periodic Access
IV	15 – 100	Limited Access
V	Greater than 100	Restricted Access

TABLE 12.3-2 RADIATION ZONE AREAS

Location	Maximum Dose Rate (mrem/hr)
Normal Operation	
Site Boundary	0.0005
Reactor Building Interior	
During Operation (Below Elev. 25'-0")	15
(Above Elev. 25'-0")	150
Certain Equipment Areas in Auxiliary Building	15
Fuel Handling Areas	15
Auxiliary Building Corridors, Cable Room, Local Control Panels, Equipment Room, Containment Operating Floor During Shutdown	2.5
Control Room	0.5
Turbine Building	0.5
Administration Building	0.5
Radiation Counting Room	0.5
Accident Condition	
Inside Control Room following DBA	5 rem integrated whole body dose over duration of accident

TABLE 12.3-3 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-1

Bldg: Containment Elevation: (-) 26'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Reactor Coolant Drain Tank	18	36	24	48	24	-
Excess Letdown Heat Exchanger	48	42	30	32	24	-
Regenerative Heat Exchanger	36	36	36	36	24*	-
Reactor Coolant Drain Tank Heat Exch.	12	12	12	36	24	-

* Plus 4 inches of lead plate

TABLE 12.3-4 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-4

Bldg: Vaults Elevation: (-) 61'-0" & above

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
RHR Heat Exch. - A	48	30	30	30		30
RHR Pump - A	48	30	30	30	30	48
RHR Heat Exch. - B	30	30	36	30	24	30
RHR Pump – B	30	30	48	30	30	48

TABLE 12.3-5 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-5

Bldg: PAB Elevation: 2'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
BTR Demin – A	36	36	24	24	48	48
BTR Demin - B, C	24	36	24	24	48	48
BTR Demin - D, E	24	30	24	24	48	48
Cation Demin	24	42	36	30	48	48
Mixed Bed Demins	36	42	36	36	48	48
Spent Fuel Pool Demin	36	30	30	24	48	48
Letdown Heat Exch.	30	24	24	30	36	48
Letdown Reheat Hx	24	24	24	30	36	48
Letdown Chiller Hx	24	24	24	36	36	48
Moderating Heat Exch	24	24	24	36	36	48
Seal Water Hx	24	24	24	24	36	48
Seal Water Return Fltr	24	24	24	24	36	48
Seal Injection Filter	24	24	24	24	36	48
Reactor Coolant Fltr	24	24	24	36	36	48
Demin Pre-Filter	24	24	24	36	36	48
Fuel Pool Post-Filter	24	24	24	24	36	48
Fuel Pool Pre-Filter	24	24	24	24	36	48

TABLE 12.3-6 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-5

Bldg: PAB Elevation: 7'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Cent. Charging Pump – A	33	24	24	0 (15)*	36	36
Cent. Charging Pump – B	24	24	24	0 (15)	36	36
Recip. Charging Pump	24	24	30	0 (15)	36	36
Letdown Degasifier (Liquid)	30	30	30	24	--	48
Letdown Degasifier Pumps	30	24	30	30	24	48

* (n) - Denotes "n" feet of air available for additional shielding.

TABLE 12.3-7 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-6

Bldg: PAB Elevation: 25'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Letdown Degasifier (Vapor)	42	48	42	42	30	--
Sample Heat Exch's	36	24	24	24	48	24

TABLE 12.3-8 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-7

Bldg: PAB Elevation: 53'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Volume Control Tank	48	48	48	48	48	48

TABLE 12.3-9 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-8

Bldg: WPB Elevation: (-)31'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Resin Sluice Tanks	36	36	30	42	30	48
Resin Transfer Pump	30	24	36	42	24	48
Floor Drain Tanks	36	24	30	42	24	48
Floor Drain Tank Pumps	36	12	30	24	21	48
Waste Concentrates Tank	30	30	24	30	24	48
Waste Concentrates Transfer Pump	24	30	24	24	24	48
Chemical Drain Treatment Transfer Pump	24	30	24	0 (10)*	30	48
Chemical Drain Treatment Tanks	24	30	24	30	24	48

* (n) - Denotes "n" feet of air available for additional shielding

TABLE 12.3-10 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-9

Bldg: WPB Elevation: (-)3'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Cesium Removal Ion Exchanger – A	36	30	24	36	30	30
Cesium Removal Ion Exchanger – B	24	30	24	36	30	30
Recovery Demineralizers	24	30	24	30	30	30
Waste Demineralizer	24	30	24	30	30	30
Recovery Filter – A	36	30	24	24	30	30
Recovery Filter - B	24	30	24	24	30	30
Recovery Evaporator Filter – A	24	30	24	24	30	30
Recovery Evaporator Filter – B	24	30	66	24	30	30
Recovery Demineralizer Filter – A	66	30	24	24	30	30
Recovery Demineralizer Filter – B	24	30	24	24	30	30
Floor Drain Filter – A	24	30	24	24	30	30
Floor Drain Filter – B	24	30	66	24	30	30
Waste Demineralizer Filter	66	30	24	24	30	30
Primary Drain Tank Degasifier Filter	24	30	48	24	30	30

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Resin Sluice Filter	30	24	12	36	24	24
Primary Drain Tanks	36	24	30	36	30	30
Primary Drain Tank Transfer Pumps	36	18	30	24	21	21
Primary Drain Tank Degasifier Recirc Pumps	24	24	0(7)*	48	24	24
Waste Evaporator Reboiler	12	24	24	24	24	24
Waste Evaporator Bottoms Package	24	24	12	24	24	24
Boron Waste Storage Tank – A	30	24	24	30	24	24
Boron Waste Storage Tank – B	30	30	24	24	24	24
Recovery Evaporator Feed Pumps	24	24	24	24	24	24
Recovery Evaporator Reboilers	12	24	24	24	24	24
Recovery Evaporator Bottoms Packages	24	24	12	24	24	24

* (n) - Denotes "n" feet of air available for additional shielding

TABLE 12.3-11 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-10

Bldg: WPB Elevation: 25'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Primary Drain Tank Degasifier (Lower Half)	24	24	34	42	--	24
Primary Drain Tank Degasifier (Upper Half)	24	24	36	42	36	--
Refueling Water Storage Tank	24	24	24	24	--	--

TABLE 12.3-12 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-11

Bldg: WPB Elevation: 42'-5"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Crystallizer Recirc. Pump	18	12	30	30	24	24
Crystallizer Drain Pump	18	12	30	30	24	24
Crystallizer Reflux Pump	18	12	30	30	24	24
Evaporator/Extruder	*	30	18	18	24	18/24

* 2" lead brick (as needed)

TABLE 12.3-13 COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-12, 12.3-13

Bldg: WPB Elevation: 53'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Waste Hopper	36	36	36	30	30	30
Waste Gas Chiller-Dryer	24	36	24	24	24*	24
Carbon Delay Bed – A	48	36	36	36	36	36
Carbon Delay Bed – B	48	48	48	36	36	36
Carbon Delay Beds - C,D,E	42	30	30	30	30	24
Hydrogen Surge Tank	42	30	30	30	30	24
Iodine Guard Beds	24	36	36	36	30	24
Waste Feed Tanks	30	30	30	36	12	24
Waste Feed Recirc. Pumps	30	30	12	36	12	24
Spent Resin Recirc. Pump	36	30	30	36	12	24
Resin Centrifuge Metering Pump	24	36	18	12	12	24
Spent Resin Centrifuge	12	12	12	24	12	24
Evaporator/Crystallizer Vapor Body	12	30	30	24	15	24
Concentrates Bottom Tank	30	30	30	36	12	24
Concentrates Bottom Tank Recirc. Pump	30	30	12	36	12	24

* Plus 1" Steel Plate

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS				CEILING	FLOOR
	NORTH	EAST	SOUTH	WEST		
Waste Metering Pump	24	36	18	12	12	24
Alternate Station Concentrates Feed Pump	30	30	12	36	12	24
Entrainment Separator	12	30	30	24	15	24
Crystallizer Condenser	30	30	12	36	12	24
Crystallizer Subcooler	30	30	12	36	12	24
Spent Resin Dewatering Pump	36	30	30	36	12	24
Crystallizer Reflux Pot	12	30	30	24	15	24

TABLE 12.3-14 AREA RADIATION MONITORS

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UFSAR FIGURE REFERENCE
	Containment Structure							
6534	In-Core Instrument Seal Table	GM (Note 4)	15	10^{-1}-10^{4}		1	Non 1E	12.3-2
6535A, B (Note 1)	Manipulator Crane	GM (Note 4)	15	10^{-1}-10^{4}		2	1E	12.3-3
6536-1, 2	Personnel Hatch (Post-LOCA)	Ion Chamber	2.5	10^{+1}-10^{9}		2	Non 1E	12.3-3
6576A, B	Containment (Post-LOCA)	Ion Chamber	25	10^{0}-10^{8} r/hr		2	1E	12.3-3
	Primary Auxiliary Building							
6537	Sampling Room	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6538, 6539	RHR Pump Area	GM	>100	10^{-1}-10^{4}		2	Non 1E	12.3-4
6540	Volume Control Tank Area	Ion Chamber	$8x10^{4}$	10^{1}-10^{7}		1	Non 1E	12.3-7
6541	Lower Level	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-5
6543	Entrance	GM	>100	10^{-1}-10^{4}		1	Non 1E	12.3-5
6544	Entrance	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6545, 6546 6547	Charging Pump Room	GM	110	10^{-1}-10^{4}		3	Non 1E	12.3-5

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UFSAR FIGURE REFERENCE
6508-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6563-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6517-1,2	RHR - Pump Vault HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	
	Fuel Storage Building							
6549	Spent Fuel Pool Area	GM	2.5	10-1-10^{4}		1	Non 1E	12.3-15
6518	Spent Fuel - HRAM	Ion Chamber	2.5	10^{-2}-10^{4} r/hr		1	Non 1E	12.3-15
	Control Room							
6550	Main Control Board Area	GM	0.5	10^{-1}-10^{4}		1	Non 1E	1.2-32
	Waste Processing Building							
6551	Waste Gas Processing Area	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-11
6552	Truck Loading Area	GM	1.5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6553	Radwaste Control Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6554	Waste Management Control Panel Area	GM	5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6570	Extruder/Evaporator Manifold Area	GM	(Note 6)	10^{0}-10^{5}		1	Non 1E	12.3-10
6571	Compacted Rad Waste Storage Area	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-10

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UFSAR FIGURE REFERENCE
	Administration & Service Building							
6555	Hot Chemistry Laboratory	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6556	Decontamination Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6557	RCA Shop (Note 2)	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6558	RCA Personnel Decontamination Area	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16
6559	RCA Women's Locker Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-16

(Note 1) 6535-A and 6535-B will automatically terminate containment purge in the event of high radiation during fuel handling operations.

(Note 2) RCA – Radiologically Controlled Area.

(Note 3) Deleted.

(Note 4) GM – Geiger-Mueller.

(Note 5) Radiation monitoring setpoints are varied during operation to follow station conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the station operating procedures.

(Note 6) >100

TABLE 12.3-15 AIRBORNE RADIATION MONITORS (SKID MOUNTED DETECTORS)

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	REFERENCE ISOTOPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH µCi/cc	(Note 5) ALARM SET POINT µCi/cc	DETECTOR QTY.	IEEE CLASS	ENERGY LEVEL	LOOP DIAG. 1-NHY	P&ID 1-NHY
6526-1	Containment Air Particulate (Note 4)	Beta	I^{131}, Cs^{137}	2.5	10^{-10}-10^{-6}		1	Non 1E	Note 3	506135	20504
6526-2	Containment Radiogas	Beta	Xe^{133}	2.5	10^{-6}-10^{-2}		1	Non 1E	Note 1	506135	20504
6528-1	Plant Vent – WRGM (Low Range)	Beta	Xe^{133}, Kr^{85}	2.5	10^{-7}-10^{-1}		1	Non 1E	Note 1	506607	20494
6528-2	Plant Vent – WRGM (Mid Range)	Beta	Xe^{133}, Kr^{85}	2.5	10^{-3}-10^{3}		1	Non 1E	Note 1	506607	20494
6528-3	Plant Vent – WRGM (Hi Range)	Beta	Xe^{133}, Kr^{85}	2.5	10^{-1}-10^{5}		1	Non 1E	Note 1	506607	20494
6495	WRGM Backup	Ion Chamber	Xe^{133}, Kr^{85}	2.5	10^{-1}-10^{7} mr/hr		1	Non 1E	Note 1	506607	RM-20509
6531-2	WPB Radiogas	Beta	Xe^{133}	2.5	10^{-7}-10^{-3}		1	Non 1E	Note 1	506885	20498
6532-2	PAB Radiogas	Beta	Xe^{133}	2.5	10^{-7}-10^{-3}		1	Non 1E	Note 1	506598	20510
6548	Containment Radiogas Backup	Beta	Xe^{133}	2.5	10^{-6}-10^{-2}		1	Non 1E	Note 1	506136	RM-20510

NOTES

Note	Isotopes	Max. Beta Energy (Mev)	Predominant Gamma Energy (Mev)
1	Xe^{133}	0.346	0.081
	Xe^{135}	0.92	0.249
	Kr^{85}	0.67	0.514
	Kr^{85m}	0.82	0.150
2	I^{131}	0.606	0.364
	I^{133}	1.27	0.53
	Cs^{134}	0.662	0.604
	Cs^{137}	0.514	0.662
	Co^{58}	0.474	0.81
	Co^{60}	0.314	1.17, 1.33
3	Same as Note 2, plus:		
	Rb^{88}	5.3	1.863

4 Containment air particulate monitor functions as a leakage detector and must survive the SSE (reference Regulatory Guide 1.45 and Standard Review Plan 5.2.5).

5 Radiation monitoring setpoints are varied during operation to follow station operating conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the station operating procedures.

TABLE 12.3-16 AIRBORNE RADIATION MONITORS (DETECTORS MOUNTED IN DUCT)

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	BACK-GRD. mr/hr	RANGE LOW-HIGH CPM	(Note 1) ALARM SET POINT CPM	DETECTOR QTY.	IEEE CLASS	LOOP DIAG.	P&ID	LOCATION 9763-F-
6506A1, A2, B1, B2 (1 RM)	CTL Rm East Air Intake	GM	0.5	10^1-10^6			1E	506151	NHY-CBA-20303	500210
6507A1, A2, B1, B2, (1 RM)	CTL Rm West Air Intake	GM	0.5	10^1-10^6		8	1E	506152	NHY-CBA-20303	500210
	Administration Building									
6523	Fume Hood Exh FN-115	GM	0.5	10^1-10^6		1	Non 1E	506668	NHY-AAH-20004	615002 & 11
6524	Fume Hood Exh FN-116	GM	0.5	10^1-10^6		1	Non 1E	506668	NHY-AAH-20004	615002 & 11
6525	Fume Hood Exh FN-117	GM	0.5	10^1-10^6		1	Non 1E	506668	NHY-AAH-20004	615002 & 11
	Fuel Storage Building									
6562	FAH-Fuel Stor Bldg Exh	GM	0.5	10^1-10^6		1	Non 1E	506452	NHY-MAH-20497	500178 500332
	Containment Enclosure									
6566	EAH-Contn Encl Emerg Exh	GM	0.5	10^1-10^6		1	Non 1E	506602	NHY-MAH-20495	500179
	Primary Auxiliary Building									
6567	PAB-Misc Ventilation	GM	2.5	10^1-10^6		1	Non 1E	506603	NHY-MAH-20494	500175 & 8
6568	PAB-Contn Enclosure	GM	2.5	10^1-10^6		1	Non 1E	506596	NHY-MAH-20494	500175 & 8

TABLE 12.4-1 ESTIMATED AVERAGE ANNUAL DOSE OF SEABROOK PERSONNEL

Activity	Total Dose (Man-Rem)	Percent of Total Dose
Reactor Operations and Surveillance	35.0	9
Routine Maintenance	63.0	17
In-Service Inspection	42.0	11
Special Maintenance	164.0	44
Waste Processing	28.0	8
Refueling	39.84	11
	371.84	100 percent

TABLE 12.4-2 OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE OPERATIONS AND SURVEILLANCE

Activity	Radiation Field (R/Hr)	Exposure Time (Hrs/Year)[6]	Total Dose (Manrems/Year)
Walking In Radiation Zones	0.0005[1]	10,000	5.0
Control Room	0.0001[2]	25,000	2.5
Routine Surveillance Inside Containment	0.05[3]	100	5.0
Health Physics Surveys	0.0025[4]	3,600	9.0
Collection of Radioactive & Water Chemistry Samples	0.0025[4]	1,460	3.65
Radiochemistry Analysis	0.0012[5]	2,500	3.00
Routine Surveillance Primary Auxiliary Building	0.001[2]	3,000	3.00
Routine Surveillance Fuel Storage Building	0.005[2]	750	3.75
Miscellaneous			0.00
			34.9

Information Sources:

(1) The estimated occupancies of the different radiation zones provided in Table 12.3-1 indicate that approximately 75 percent of the total manhours of occupancy will be in Radiation Zone I areas. The average field was taken to be equal to the design maximum for Zone I areas. Since the average field in the Zone I areas is expected to be 20 percent or less of the design value, this conservatism takes into account the 25 percent of total time personnel spend in the radiation zones with more intense fields (II, III, and IV).

(2) Average fields assumed to be 20 percent of the maximum design fields for these areas. Radiation zones and justification for this assumption provided in Section 12.3.

(3) Estimated field consistent with other FSARs and the projected design range of exposures for these areas at Seabrook Station.

(4) Estimated occupancies of radiation zones by work function in Table 12.3-1 indicate that the majority of time spent in radiation zones by chemistry and health physics personnel is in Zone II areas. Therefore, the average exposure is conservatively assumed equal to the design maximum for Zone II.

(5) Information in appendix of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 33, 1980 (Reference 6) indicates that exposure for this activity is approximately half that generally encountered in the health physics surveys.

(6) Exposure time estimates based on engineering judgment.

TABLE 12.4-3 OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE MAINTENANCE

Activity	Radiation[1] Field (R/Hr)	Exposure [2] Time (Hours/Year)	Total Dose[3] (Man-Rems/Year)
Polar Crane Maintenance	0.005	160	0.8
Check & Repair Snubbers	0.015	200	3.0
Reactor Coolant Pump Motor Work	0.100	60	6.0
Pressurizer Work	0.005	60	0.3
Containment Pressure Test & Valve Repair Containment Instrument Calibration	0.050	50	2.5
Repair Containment Sump Pumps & Level Indicators	0.050	50	2.5
Replace Excore Detectors	0.050	90	4.5
Repair Dampers & Ducts	0.005	120	0.6
Primary Aux, Bldg. Valve Repair	0.010	200	2.0
Instrument Work & Calibration	0.005	800	4.0
Gen. Maintenance	0.005	1000	5.0
Filter, Ion Exchanger & Demineralizer Repair	0.150	40	6.0
Radwaste Evaporator Repair	0.150	40	6.0
General Decon. & Revamping	0.015	750	11.0
Remove/Replace Filters For Containment Cleanup	0.100	12	1.2
Radwaste Bldg. General Maintenance	0.100	40	4.0
Boron Recovery Evaporator Repair	0.100	40	4.0
Miscellaneous			0.00
			63.4

Information Sources:

(1) Exposure rates for these activities assumed to be comparable to exposure rates for similar types of activities reported in "Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw (Reference 7); and "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," Atomic Industrial Forum, Inc. Abstracts of both papers appear in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980 (Reference 6).

(2) Estimate of total exposure time based on engineering judgment.

(3) Total annual exposures consistent with historical data for similar activities reported in "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants" (Reference 1).

TABLE 12.4-4 OCCUPATIONAL DOSE ESTIMATE - REFUELING OPERATIONS

Activity	Radiation Field (Rem/Hr)	Exposure Time (Man-Hrs)	Total Dose (Man-Rem)
Move, Setup, Check Fuel Transfer Equip.	0.1-0.005	32	0.62
Remove Seismic Supports	0.025	8	0.20
Disconnect CRDM & DRPI Cables	0.025	28	0.70
Remove RV Head Insulation	0.1	20	2.00
Remove Transfer Tube Quick Closure Hatch	0.1	See Miscellaneous	
Retract In-Core Thimbles	0.005	20	0.10
Connect RV Head Eductor	0.1	2	0.20
Lower Stud Tensioners Racks to Cavity	0.005-0.025	12	0.15
Relax RV Studs	0.05	30	1.50
Remove RV Studs	0.05	21	1.05
Install Stud Hole Plugs	0.05	9	0.45
Install RV Guide Studs	0.05	12	0.60
Install Permanent Cavity Seal Ring Hatches	0.05	2	0.1
Install Head Lift Rig Load Cell	0.1	6	0.60
Remove RV Head	0.005-1	8	0.21
Disconnect Control Rod Drive Shafts	0.1	9	0.90
Remove Upper Internals	0.005-0.1	16	0.27
Index Refueling Crane	0.005	4	0.02
Shuffle Fuel	0.005	960	4.80
Map Fuel Core Locations	0.005	15	0.07
Clean, Inspect UT Studs, Nuts & Tensioners	0.005	60	0.30
Install New RV Head O Rings	0.5	4	2.00
Install Upper Internals	0.005-0.1	12	0.25

Activity	Radiation Field (Rem/Hr)	Exposure Time (Man-Hrs)	Total Dose (Man-Rem)
Latch Control Rod Drive Shafts	0.1	9	0.90
Drain Cavity	--	3	--
Clean Vessel Flange Surface	0.5	10	5.00
Install In-Core Thimbles	0.005	20	0.10
Install RV Head Lifting Rig	0.005-0.5	14	0.80
Cavity Decontamination	0.025	96	2.40
Remove Stud Hole Plugs	0.05	16	0.80
Clean Stud Holes	0.05	16	0.80
Install RV Head Studs	0.05	54	2.70
Tension RV Head Studs	0.05	105	5.25
Install Transfer Quick Closure Hatch	0.1	See miscellaneous	
Flood Cavity	--	3	--
Connect CRDM & DPRI Cables	0.025	28	0.70
Remove Permanent Cavity Seal Ring Hatches	0.05	2	0.1
Install RV Head Insulation	0.1	20	2.00
Install Seismic Supports	0.025	8	0.20
Miscellaneous*			1.00
		TOTAL	39.84

* Includes the removal and installation of the transfer tube quick closure hatchInformation Source:"Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw of Westinghouse Nuclear Energy Systems (Reference 7); paper presented at the International Radiation Protection Conference, Paris, December 1979. The abstract of this paper appeared in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," which was prepared for the Electric Power Research Institute by Stone & Webster Engineering Corporation, January 22, 1980 (Reference 6).

TABLE 12.4-5 OCCUPATIONAL DOSE ESTIMATES DURING WASTE PROCESSING

Activity	Average Dose Rate (Mrem/Hr)	Exposure[3] Time Per Event (Man Hours)	Frequency	Dose (Man-Rem/Year)
Control Room Panel	0.5[1]	8	3/day	4.37
Sampling & Filter Changing	15.0[2]	16	1/week	12.48[4]
Panel Operation, Inspection & Testing	0.5[1]	2	1/day	0.37
Operation of Waste Processing and Packaging Equipment	5.0[2]	40	1/week	10.4[4]
Miscellaneous				0.00
			Total	27.60

(1) Average dose rate assumed to be 20 percent of the maximum design dose rate for the radiation zone in which the equipment is located. Radiation zones and justification for this assumption provided in Table 12.3-1.

(2) Average dose rate for these activities assumed to be equal to the maximum design dose rate for the radiation zone in which the equipment is located.

(3) Exposure time estimates based on engineering judgment.

(4) Total annual exposure for the activity consistent with historical data reported in "Compilation and Analysis of Data on Occupational Radiation Exposure," (Reference 1).

TABLE 12.4-6 OCCUPATIONAL DOSE ESTIMATES DURING IN-SERVICE INSPECTION

Activity	Radiation[1] Field (R/Hr)	Exposure[2] Time (Hrs/Year)	Total Dose[3] (Man-Rem/Year)
Erect/Remove Scaffolding	0.02	50	1.0
Remove/Replace Insulation	0.05	100	5.0
Remove/Replace Primary Manway Covers	0.02	125	2.5
Remove/Replace Secondary Manway Covers	0.005	80	0.4
Complete Steam Generator Tube Inspection	0.12	125	15.0
Steam Generator Secondary Side	0.01	50	0.5
Reactor Coolant Pumps	0.01	50	0.5
Reactor Vessel	0.01	80	0.8
Makeup Pumps	0.005	80	0.4
Pressurizer	0.005	40	0.2
Containment Piping	0.15	60	9.0
Reactor Vessel Internals	0.05	140	7.0
		Total	42.3

Information Sources:

(1) Exposure rates for these activities assumed to be comparable to exposure rates for similar types of activities reported in "Radiation Exposure Management - The Westinghouse ALARA Program," J. D. Cohen and J. R. Magaw, and "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," Atomic Industrial Forum, Inc. Abstracts of both papers appear in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980.

(2) Estimate of total exposure time based on engineering judgment.

(3) Total annual exposures consistent with historical data for similar activities reported in "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants" (Reference 1).

TABLE 12.4-7 OCCUPATIONAL DOSE ESTIMATES DURING SPECIAL MAINTENANCE

Activity	Radiation Field (Rem/Hr)	Exposure Time (Man-Hrs)	Total Dose (Man-Rem)
Residual Heat Removal Pump Maintenance	0.200(1)	60	12.00
Reactor Coolant Pump Maintenance	0.100(1)	150	15.00
Steam Generator Maintenance and Tube Plugging	0.150(1)	360	54.00
Steam Generator - Eddy Current Testing	0.125(1)	240	30.00
Block Valve Maintenance	0.250(1)	72	18.00
Water Lancing	0.125(2)	120	15.00
Misc Pipe Repair	0.100(2)	50	5.00
Replace Upper Guide Structure on Control Rods	0.125(2)	120	15.00
Miscellaneous			0.00
		Total	164.00

Information Sources:

(1) "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Commercial Nuclear Power Industry," (to be published) prepared for Atomic Industrial Forum, Inc., by Catalytic, Inc., (1979-1980) (Reference 8). The abstract of this paper appeared in the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants" which was prepared for the Electric Power Research Institute by Stone & Webster, Engineering Corporation, January 22, 1980. (Reference 6).

(2) Exposure rates for these activities assumed to be the same as for similar types of activities reported in Reference 1.

(3) Average total annual exposures for these activities obtained from "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum, Inc., September 1974 (Reference 1); and material appended to the preliminary draft of "Technology Planning Study of Occupational Radiation Exposure in LWR Nuclear Power Plants," Electric Power Research Institute, January 22, 1980 (Reference 6).

TABLE 12.5-1 PORTABLE HEALTH PHYSICS INSTRUMENTATION

Type of Instrument	Quantity Minimum	Sensitivity	Range	Calibration Method	Calibration Frequency
Ion Chamber (Low Range)	16	Beta, Gamma	0 to 1 R/hr.	Source	Annual
Ion Chamber (Mid Range)	8	Gamma	0 to 50 R/hr.	Source	Annual
Ion Chamber (High Range)	4	Gamma	Up to 10,000 R/hr.	Source	Annual
Geiger Mueller Detector	6	Beta, Gamma	0 to 50,000 cpm	Source and Pulse Generator	Annual
	10	Beta, Gamma	0 to 200 mR/hr	Source	
Alpha Scintillation or Proportional Detector	4	Alpha	0 to 500,000 cpm	Source	Annual
Telescoping Survey Instrument	4	Gamma	0 to 1,000 R/hr.	Source	Annual
Neutron Dose Rate Detector	3	Neutron	0.001 to 5 rem/hr	Source and Pulse Generator	Annual
Air Sampler (Low Volume)	10	Particulate and Iodine	--	Flow Rate	Annual
Air Sampler (High Volume)	6	Particulate and Iodine	--	Flow Rate	Annual
Air Sampler (Personnel)	10	Particulate	--	Flow Rate	Annual
Continuous Air Monitor	4	Particulate and Noble Gas Monitor	--	Flow Rate and Response	Annual

TABLE 12.5-2 PERSONNEL MONITORING INSTRUMENTS

Type of Detector	Minimum Quantity	Sensitivity	Range
Hand held frisker	24	Beta, Gamma	0 to $5x10^4$ cpm
Portal Monitor or large area detection device	2	Beta or Gamma	Variable Alarm Setpoint
Self-reading dosimeter Note 1	No Min. Required	Gamma	Multiple ranges available
TLD	750 Station Use	Beta, Gamma, Neutron	Meets industry Standards
	250 Emergency	Beta, Gamma	
Whole body counter	1	Gamma	Meets industry Standards
Electronic Dosimeters	300	Gamma	Meets industry Standards

Note 1: The Station has transitioned to use of electronics dosimeters. SRPDs are available for use; however, they are not maintained calibrated. Calibration would be performed prior to use.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 12
RADIATION PROTECTION

FIGURES

12.3-1	Radiation Zone Map -Containment Structure- Plan at Elevation (-) 26'-0"
12.3-2	Radiation Zone Map - Containment Structure - Plan at Elevation 0'-0"
12.3-3	Radiation Zone Map - Containment Structure - Plan at Elevation 25'-0"
12.3-4	Radiation Zone Map - RHR, Containment Spray, SI Equipment Vault Plans
12.3-5	Radiation Zone Map - Primary Auxiliary Building - Plans at Elevation 7'-0" and Below
12.3-6	Radiation Zone Map- Primary Auxilia1y Building- Plans at Elevation 25'-0"
12.3-7	Radiation Zone Map- Primary Auxiliary Building- Plans at Elevation 53'-0" and 81'-0"
12.3-8	Radiation Zone Map- Waste Processing Building- Plans at Elevations (-) 31'-0", (-) 18'-3" and 13'-0"
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12.3-10	Radiation Zone Map -Waste Processing Building - Plan at Elevation 25'-0"
12.3-11	Radiation Zone Map - Waste Processing Building - Plans at Elevations 42'-5" and 65'-0"
12.3-12	Radiation Zone Map -Waste Processing Building - Plan at Elevation 53'-0"
12.3-13	Radiation Zone Map - Waste Processing Building - Partial Plans at Elevation 53' -0" and 9' -0"
12.3-14	Radiation Zone Map- Waste Processing Building- Plan at Elevation 86'-0"

12.3-15	Radiation Zone Map- Fuel Storage Building -Plan at Elevations 21'-6" and 25'-0"
12.3-16	Radiation Zone Map- Fuel Storage Building- Plan at Elevations 7'-0" and 10'-0"
12.3-17	RCA Boundary- Administration and Service Building - First Floor Plan at Elevation 21'
12.3-18	Normal Exhaust Unit - Primary Auxiliary Building
12.3-19	Area Radiation Monitors System - Instrumentation Engineering Diagram
12.3-20	Airborne Radiation Monitoring System- Instrumentation Engineering Diagram

See 805184

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Containment Structure - Plan at Elevation (-) 26'-0"	
		Figure 12.3-1

See 1-NHY-805185

See 805186

See 805187

See 805188

See 805189

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Primary Auxiliary Building - Plans at Elevation 25'-0"	
		Figure 12.3-6

See 805190

See 805896

See 805891

See 1-NHY-805892

See 1-NHY-805897

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plans at Elevations 42'-5 And 65 '-0	
		Figure 12.3-11

See 1-NHY-805893

See 1-NHY-805894

See 805895

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Waste Processing Building - Plan at Elevation 86'-0	
		Figure 12.3-14

See 805181

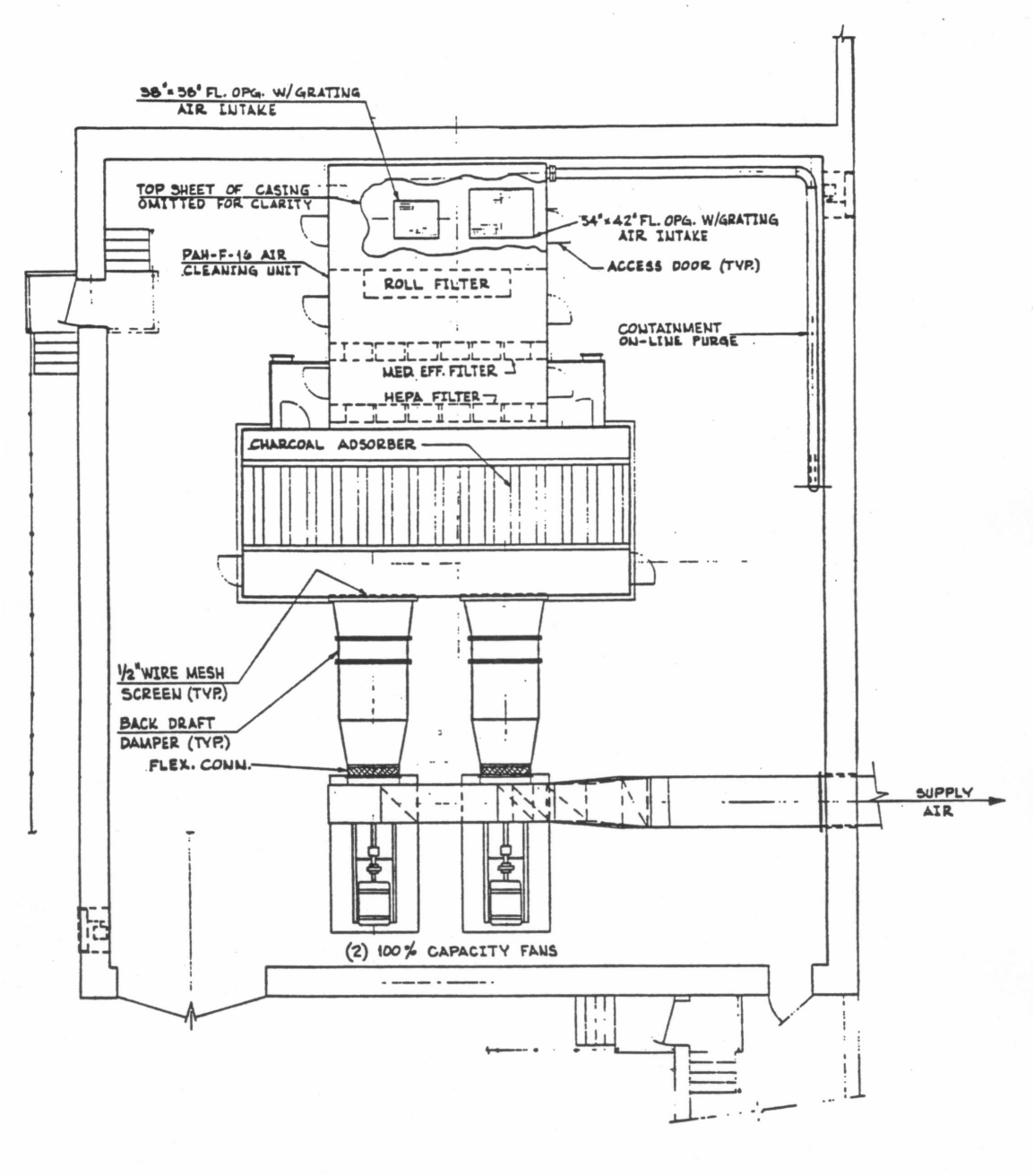
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Fuel Storage Building - Plan at Elevations 21'-6 And 25 '-0	
		Figure 12.3-15

See 805182

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radiation Zone Map - Fuel Storage Building - Plan at Elevations 7'-0 And 10 '-0	
		Figure 12.3-16

See 1-NHY-805183

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCA Boundary - Administration and Service Building - First Floor Plan at Elevation 21'	
		Figure 12.3-17



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normal Exhaust Unit – Primary Auxiliary Building	
		Figure 12-3-18

See 1-NHY-500017 Sh. 1

See 1-NHY-500017 Sh. 2

See 1-NHY-500016 Sh. 1

See 1-NHY-500016 Sh. 2