SECTION 12

RADIATION PROTECTION

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

LIST OF TABLES

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF TABLES (Cont)

LIST OF FIGURES

LIST OF FIGURES (Cont)

LIST OF FIGURES (Cont)

LIST OF FIGURES (Cont)

(Historical Information)

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

12.1.1 Policy Considerations

12.1.1.1 Management Policy

The policy of PSEG Nuclear LLC is to maintain occupational radiation exposures "as low as reasonably achievable" (ALARA) at Hope Creek Generating Station (HCGS). This includes maintaining the annual integrated dose to station personnel and to individuals working at the station ALARA. PSEG management is firmly committed to performing all reasonable actions to ensure that radiation exposures are maintained ALARA.

Sections 12.1.2, 12.3.1, and 12.3.2 discuss the ALARA considerations that have been incorporated into the design of HCGS.

HCGS is operated and maintained in such a manner as to ensure occupational radiation exposures are ALARA. The operational ALARA program is described in Section 12.5. Training programs ensure that personnel understand both how and why occupational radiation exposures are maintained ALARA. A corporate ALARA program ensures implementation of the station ALARA policy by various program reviews.

12.1.1.2 Management Responsibilities

Figures 13.1-1 through 13.1-4 show the PSEG Nuclear management organizational structure for HCGS.

The President and Chief Nuclear Officer (P/CNO) has the corporate responsibility for the ALARA program. The responsibility for coordination and administration of the ALARA program is delegated to the Radiation Protection Manager who is responsible for ensuring that radiation protection policies and

(Historical Information)

support the PSEG Nuclear LLC ALARA program. The Vice Presidents are responsible for implementing the PSEG Nuclear LLC ALARA program at Hope Creek.

During the design and construction phase, an ALARA program is implemented to ensure that the design and construction of the facility is such that occupational exposures are ALARA. This included ensuring that, to the extent practicable:

1. Design concepts and station features reflected consideration of the activities of station personnel that might be anticipated and that might lead to personnel exposure to substantial sources of radiation.

2. Specifications for equipment reflect the objectives of ALARA, including among others, considerations of reliability, serviceability, and limitations of internal accumulations of radioactive material.

During the startup and operation phases, the Plant Manager is responsible for controlling radiation exposure in a manner consistent with ALARA requirements and is specifically responsible for the onsite radiation protection program. Responsibilities with respect to the ALARA program include:

1. Ensuring support from all station personnel.

2. Participating in the selection of specific ALARA goals and objectives for the station.

3. Supporting the designated Radiation Protection Manager in formulating and implementing the station operational ALARA program.

4. Expediting the collection and dissemination of data and information concerning the program to the corporate management and staff.

(Historical Information)

The ALARA responsibilities of the Plant Manager are implemented by the Radiation Protection Department which, in accordance with ALARA principles, develops the radiation protection program and procedures, reviews other applicable station procedures, estimates and monitors personnel exposures, and disseminates information related to the ALARA program.

The major ALARA responsibilities of the designated Radiation Protection Manager include:

1. Participating in reviews of design changes for facilities and equipment that can affect potential radiation exposures.

2. Identifying locations, operations, and conditions that have the potential for causing significant exposures to radiation.

3. Initiating and implementing an exposure control program that includes the establishment of person-rem goals.

4. Developing plans, procedures, and methods for keeping radiation exposures of station personnel ALARA.

5. Reviewing, commenting on, and recommending changes in applicable procedures to maintain exposures ALARA.

6. Participating in the development and approval of training programs related to work in radiation areas or involving radioactive material.

7. Supervising the radiation surveillance program to maintain data on exposures of and doses to station personnel by specific job functions and type of work.

8. Supervising the collection, analysis, and evaluation of data and information attained from radiological surveys and monitoring activities.

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

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

Section 13 contains additional information concerning ALARA responsibilities and reporting relationships at HCGS.

12.1.1.3 Policy Implementation

PSEG Nuclear's ALARA policy is implemented at HCGS by the radiation protection staff under the direction of the Radiation Protection Superintendent. ALARA philosophy and considerations are incorporated into permanent station procedures dealing specifically with ALARA concerns. The operational ALARA considerations identified in Sections 12.1.3 and 12.5.3.2 are implemented by these procedures.

Section 12.5.3.6 describes the radiation protection training program that provides the necessary knowledge to appropriate station personnel so that they understand how and why they should maintain their occupational radiation exposures ALARA.

Radiation Protection develops policies for and verifies the effectiveness of the PSEG Nuclear LLC ALARA program. Radiation Protection also reviews station implementation of the PSEG Nuclear LLC ALARA program on a continuing basis.

(Historical Information)

12.1.2 Design Considerations

This section describes those general design considerations that are applied to the plant design and layout for the purpose of incorporating features that ensure that occupational radiation exposures are as low as reasonably achievable (ALARA). This section also describes the review program used during the individual design stages to control, audit, and enforce all ALARA program efforts. All ALARA program efforts are based on the guidelines of Regulatory Guide 8.8. This Regulatory Guide information is needed to ensure that all aspects of the plant design, layout, and construction are directed toward the reduction of radiation exposures to individuals. The information is translated into design reviews for Hope Creek Generating Station (HCGS).

For additional detailed descriptions of the design features for maintaining personnel radiation exposures ALARA, refer to Sections 12.3.1, 12.3.2, and 12.3.3.

(Historical Information)

Experience and data from operating plants have been evaluated to decide if and how equipment or facility designs can be improved to reduce overall plant personnel exposures. Surveys made by the architect/engineer in operating BWR plants added to the knowledge of the behavior of radioactivity during all modes of plant operating conditions. Lessons learned from Public Service Electric and Gas Company (PSE&G) experiences at the Salem Generating Station have also been incorporated in the overall design program to the greatest extent possible. During plant design, operating reports and data such as those given in WASH-1311, NUREG-75/032, NUREG-0109, and Compilation and Analysis of Data on Occupational Radiation Exposure experienced at Operating Nuclear Power Plants, AIF/NESP-005, September 1975, found in References 12.1-1 through 12.1-4, have been reviewed to determine which operations, procedures, or types of equipment were most significant in causing personnel exposures. Methods to mitigate such exposures have been implemented wherever practicable. Reports and data supplied by the NSSS supplier also contributed to the overall effort.

12.1.2.1 General Design Considerations

Besides providing overall radiation protection, the main objective of the plant design for ALARA purposes is to minimize the need for personnel access into high radiation areas and to reduce the duration of personnel exposure. Wherever possible, ALARA oriented design features minimize the need for personnel to enter high radiation areas, enable personnel to perform necessary activities in low radiation areas, and also ensure radiation levels as low as is practicable in routinely occupied areas.

Both equipment and facility designs have been considered in achieving these objectives during plant operations, including normal operations, maintenance and repair, refueling operations, fuel storage, inservice inspection, waste handling, storage and disposal, and anticipated operational occurrences.

(Historical Information)

In addition to equipment and facility designs, system designs have been considered to ensure that exposures are maintained ALARA. Many plant systems expected to carry radioactive liquids and/or gases are provided with cleanup capability to reduce the inventory of circulating fission and corrosion products. This is one of the methods employed to minimize both the activation of corrosion products and their subsequent deposit on the interior surfaces of piping and equipment.

The design organization has been responsible for ensuring that the design and construction of the facility are such that occupational exposures are ALARA. To the extent practicable, this includes:

1. Design concepts and station features that reflect consideration of the activities of station personnel that can be anticipated and that might lead to personnel exposure to sources of radiation, and assurance that plant design features have been provided to reduce the anticipated exposures of plant personnel to these sources of radiation.

2. Specifications for equipment that reflect the objectives of ALARA, including, among others, considerations of reliability, durability, serviceability, and limitations of internal accumulations of radioactive materials.

3. Design audit and enforcement through the plant ALARA model design review program.

12.1.2.2 Equipment Design Considerations

Considerations for equipment design include:

1. Reliability, long service life, maintenance and calibration requirements, and durability

2. Convenience for servicing, including disassembly and reassembly, modular design concept for rapid component replacement, and removal for servicing in a lower radiation area.

3. Remote operation, inspection monitoring, servicing, and repair, including the use of special tools or equipment.

4. Redundant equipment to reduce the urgency for immediate repairs, thus providing time for planning repairs during system shutdown with ALARA in mind.

5. Isolation, venting, draining, flushing, or decontamination of systems to reduce crud deposit and thus reduce radiation levels.

6. Isolation of components from contaminated process fluids.

7. Use of high quality materials for components, such as valves, that minimize or preclude the leakage of radioactive liquids or gases.

8. Use of closed and hard piped drain and vent systems for contaminated process fluids to minimize airborne contamination and subsequent spread of radioactivity due to spillage.

9. Use of quick disconnect features to reduce the length of time personnel need to spend in radiation areas.

12.1.2.3 Facility Design Considerations

Considerations for facility design include:

1. Location of equipment according to the need for access to maintain, inspect, monitor, or operate to minimize radiation exposure.

(Historical Information)

2. Use of remote or semi-remote viewing devices such as TV cameras and shielding windows, minimizing the need to enter radiation areas.

3. Transport of contaminated components for service in low radiation areas.

4. Separation of sources of radiation such as pipes, pumps, storage tanks, and filters from normally low radiation areas.

5. Use of permanent shielding between sources of radiation and non-radioactive access and service areas.

6. Ventilation flow paths from clean areas to potentially contaminated areas.

7. Use of surface coatings to facilitate decontamination.

8. Use of labyrinth entrances to shielded cavities.

9. Use of control devices for entries into high radiation areas in accordance with 10CFR20 and technical specifications.

10. Space allocations for the establishment of temporary control points in low radiation areas near potential work areas in radiation zones.

11. Provisions for platforms to enhance accessibility for inspection, maintenance, and repair of such components as valves, motors, instruments, etc.

12. Use of local shielding for high radiation sources near active components such as pumps, motors, valves, etc, where permanent shielding is not practical.

13. Use of steel shield doors at high radiation areas where concrete labyrinths are not possible.

14. Use of handrail or chain designs to minimize personnel exposures by increasing distances between personnel and equipment sources.

15. Radiation monitoring systems designed to monitor system activities and general area radiation levels. The radiation monitoring systems provide early warning of any abnormal or accident condition.

12.1.2.4 ALARA Model Design Review Program

For the design and layout of the various plant structures and the equipment and components for HCGS, a plant model has been prepared. This plant model is an accurate replica of designed and constructed plant features at a reduced and convenient scale of 1/2-inch to the foot. Besides being used as a major tool in documenting and disseminating design information, the plant model is used as an important means to control, audit, and enforce the design efforts of all engineering disciplines to ensure that, in the overall design, nuclear radiation exposures to plant personnel, visitors, and offsite maintenance crews are ALARA.

Bechtel Power Corporation, as architect/engineer for PSE&G, was given the responsibility for the performance of the ALARA Model Design Review Program. This review is performed under the direction of PSE&G and with PSE&G as an active participant, as described in the following paragraphs. A major portion of this ALARA Design Review Program uses the HCGS model. The review of the model is multidisciplinary to confirm and ensure that the design maintains plant personnel radiation exposure ALARA.

(Historical Information)

The review of the model includes the review of the layout of equipment, piping components, and the overall general layout, and involves all disciplines to varying degrees. System engineers carry the major responsibility in the reviews. The design aspects that are reviewed for ALARA considerations include facility arrangement, shielding, system design, piping and valve design, component design, and plant facilities and support services. A system design oriented review approach has been followed and has facilitated the documentation and coordination of the ALARA model review effort.

The plant systems or plant personnel activities that have been reviewed are shown in Table 12.1-1.

The Bechtel mechanical nuclear group is responsible for the overall coordination of the project ALARA Model Design Review Program and for interfacing between the various project disciplines, the Bechtel radiation protection group, and PSE&G. A project ALARA coordinator is responsible for the coordination of all ALARA model design review efforts.

Each Bechtel discipline has assigned one engineer to coordinate that discipline's model review and to disseminate ALARA information. The discipline ALARA coordinator interfaces with the project ALARA coordinator.

Each discipline ALARA coordinator ensures that responsible discipline engineers for each system are familiar with the available ALARA guidance provided by the discipline chief or the project ALARA coordinator.

The responsibilities of the project ALARA coordinator are to provide coordination between project disciplines and between the project and PSE&G, and to document the ALARA plant model review effort.

(Historical Information)

General design considerations and methods employed to maintain in-plant radiation exposures ALARA, in accordance with Regulatory Guide 8.8, have two objectives:

1. To minimize the necessity for, and amount of, time spent in radiation areas by personnel

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

Existing Regulatory Guides, project discipline design criteria, and project engineering design procedures are used to design, review, approve, verify, and coordinate all discipline designs for ALARA conformance and implement changes based on review results. Additional documents provided specifically for the proper conduct and documentation of the ALARA effort are the ALARA Design Review Considerations Checklist, the ALARA Discipline Problem Log, and most important, the ALARA Design Modification Action Items Log.

The model is used as the principal design review tool. However, to perform an accurate review, the documents listed in Table 12.1-2 are used. This procedure identifies problems, and the suggested revision is marked on these documents for discussion, distribution, and/or corrections during the model review. In addition, these documents are distributed to all disciplines for coordination, locations, and identification of problem areas.

Following an in-house pre-client ALARA model design review of a particular system(s) or area(s), the meeting notes, together with any pertinent information, are sent to PSE&G for their review and to make them aware of any problems. An ALARA model design review is then held with PSE&G, reviewing the same system(s) or area(s). Each ALARA activity noted during the in-house pre-client meeting is discussed and resolved. The ALARA action items are then entered into the ALARA Design Modification Action Item Log

for closeout by project. It is the responsibility of the project ALARA coordinator to pursue the closeout of the individual action items with the appropriate engineering disciplines.

An ALARA documentation file maintained by the project ALARA coordinator contains the history of the ALARA Design Modification Action Items Logs. Final documentation packages on each system/area are assembled by the project ALARA coordinator and submitted for approval.

The ALARA Design Modification Action Items Log is maintained by the project ALARA coordinator and is kept in the ALARA documentation file. This log contains the list of all design changes that are recommended as a result of the ALARA review and also serves as the document to list the closeout status of the individual ALARA action items.

12.1.2.5 Field ALARA Design Program

As an extension of the ALARA model design review program performed in the Bechtel San Francisco engineering home office, and also as a supplement to the overall project ALARA effort, a Field ALARA Design Program is conducted.

The purpose of the Field ALARA Design Program is:

1. To provide the field with ALARA guidelines for routing and locating items not engineered in the San Francisco home office.

2. To emphasize the more critical items of Regulatory Guide 8.8 affecting radiation exposure reduction.

3. To provide general and specific guidelines to field engineering that are followed while implementing field design and layout responsibilities, in order to minimize

future occupational radiation exposures during maintenance activities at all conditions of plant operation, shutdown, and anticipated occurrences

4. To furnish the field office with guidelines for the development of a Field ALARA Design Review Program and the associated documentation

5. To provide field engineering with reference design drawings and project documents that show criteria and information affecting radiation exposure reductions

6. To ensure field engineering will incorporate design features to facilitate the reduction of radiation levels in areas with maintenance-prone components by discussing the optimization of the plant layout for field routed and located items; this is to separate these components from highly radioactive piping or equipment.

An ALARA Field Design Guide has been compiled for the use by field engineering in their ALARA-related efforts.

Verifying the implementation of ALARA action items identified by the Bechtel San Francisco home office and the field engineering office is the final step in the overall ALARA design review program. Verification is done by a combined walk-through of plant areas by the field ALARA review team and the San Francisco home office project engineering ALARA representative(s).

12.1.3 Operational Considerations

This section describes the development and implementation of operating procedures, including procedures for radiation protection and the as low as reasonably achievable (ALARA) program.

Specific activities detailed below ensure that occupational radiation exposures are maintained ALARA during the operation of the Hope Creek Generating Station (HCGS).

Plant procedures are further described in Section 13.5. Radiation protection operations are described in Section 12.5.

12.1.3.1 <u>Procedure Development</u>

Various procedures will be written for the different activities associated with plant operations. These include procedures for operating, maintenance, surveillance, testing, fuel handling, emergencies, radiation protection, and administration. Station procedures are prepared, reviewed, and approved in accordance with procedures and policies described in Section 13.5.

12.1.3.1.1 Station ALARA Procedures

The Station ALARA program is included as a portion of the PSEG Nuclear LLC administrative procedure implementing the Radiation Protection Program. This procedure provides the necessary basis for implementation of the Station ALARA program.

12.1.3.1.2 Station Procedures

Administrative requirements are implemented to ensure that applicable procedures developed by other plant disciplines have adequately incorporated the principle of minimizing personnel exposure. Station administrative documents describe the criteria for selection of those procedures and revisions that are reviewed

by radiation protection. Recommendations made by radiation protection normally are resolved with the appropriate plant discipline prior to submission for final review and approval.

Procedures are subject to revision whenever improved techniques or increased safety are indicated.

12.1.3.2 Station Organization

As described in Section 12.5.1, the station organization provides the Radiation Protection Manager with direct access to the Plant Manager to ensure uniform support of radiation protection and ALARA requirements. This organization allows the Plant Manager direct involvement in the review and approval of specific ALARA goals and objectives, as well as review of data and dissemination of information related to the ALARA program.

The station organization includes an ALARA Radiological Engineer who is free from routine radiation protection activities to implement the station ALARA program. This individual is primarily responsible for coordination of station ALARA activities and routinely interacts with first-line supervision in radiation work planning and post-job review.

12.1.3.3 Operating Experience

Experience gained during the operation of Salem Units 1 and 2, along with information from other boiling water reactors (BWRs), serves as the basis for procedures, techniques, and administration controls for HCGS. In addition, the radiation work permit process described in Section 12.5.3.2 provides a mechanism for collection and evaluation of data related to personnel exposure. Information collected by systems and/or components and job function assists in evaluating design or procedure changes intended to minimize future personnel radiation exposures.

12.1.3.4 Exposure Reduction

The PSEG Nuclear LLC ALARA program procedure ensures that applicable activities are completed with adequate preparation and planning, work is performed with appropriate radiation protection recommendations and support, and the evaluation of data from post-job debriefings is applied to implement improvements.

In addition, the radiation protection staff continually solicits employee suggestions, evaluates the origins of plant exposures, investigates unusual exposures, and ensures that adequate supplies and instrumentation are available.

Corporate management, through the Radiation Protection organization, performs periodic reviews of station programs to ensure that workers are receiving adequate instruction in radiation protection and ALARA requirements. The reviews emphasize the implementation of the radiation protection program. The results of these reviews may include recommendations on ways to reduce personnel exposure. The Radiation Protection Manager evaluates noted recommendations or deficiencies; and as a result of these evaluations, corrective actions or improvements may be implemented. Implemented corrective actions will be verified during subsequent reviews.

12.1.3.5 General ALARA Techniques

While ALARA considerations occur continuously, the activities conducted during outages are the most significant for dose reduction. During the outage, the ALARA Radiological Engineer will review the jobs planned and evaluate the need for an ALARA effort. Some techniques that apply to an ALARA evaluation are:

1. Reducing dose rate from a system by draining, flushing, filling, or decontaminating.

(Historical Information)

2. Installing permanent or temporary shielding if the net result is a reduction in person-rem.

3. Training workers to improve proficiency, e.g. using "dry runs" and mockups, thus reducing stay time in the radiation area.

4. Maintaining the work force in radiation areas to the minimum required to perform the job efficiently and safely.

5. Establishing control points in low radiation areas.

6. Avoiding excess conservatism in prescribing protective clothing and respirators to avoid undue stress and decreased efficiency of the workers.

7. Using special tools for remote handling of components.

8. Planning and preparing techniques and tools needed to accomplish the job before the job is started. This is supplemented through the use of "dry runs" to determine the adequacy of procedures and equipment.

9. Using historical data for comparable jobs as guidelines and to establish an expected dose limit for the job.

10. Using radiation protection monitors in high radiation areas to maintain close checks on stay times.

11. Providing adequate communications to facilitate performance of the job and to alert workers to adverse changes in radiation conditions.

12. Source identification and use of routine or special survey data.

13. Use of contamination containment devices such as glove boxes and tents.

14. Removing components to low radiation areas for servicing or repair.

15. Planning for adequate space and auxiliary services, such as lighting, welding leads, service air lines, and TV cameras, as necessary to accomplish the work quickly.

16. Establishing auxiliary ventilation/filtration systems.

17. Wet transfer or storage of contaminated components to prevent airborne contamination.

18. Contingency planning to account for known personnel hazards or accidents that may occur.

19. Isolating systems to be worked on or possible load reductions or plant shutdowns to reduce doses.

The ALARA Radiological Engineer communicates with other departments as appropriate to discuss the implementation of ALARA techniques. For most jobs, the cooperation of more than one discipline is usually required.

PSEG Nuclear LLC administrative procedures establish and outline the PSEG Nuclear LLC ALARA Program. The program is further described and implemented by station procedures. This program includes elements for a "work review process including job preplanning" and "improved work practices". Within the work review preplanning is an ALARA checklist which addresses pre-job meetings, rehearsals, and mockup training among many other items. It is intended to require accomplishment of each of the ALARA practices at various expected doses for each job. The use of the checklist is currently required at the one person rem exposure level, and includes Radiation Protection management review and approval. Pre-job meetings involve higher levels of supervision and worker

(Historical Information)

involvement as the expected doses increase. The 10 person-rem level currently requires worker rehearsals and mockup training, if available. The availability of mockups has been recognized as a need and some spares from the cancelled Unit #2 have been retained for this purpose. Additional training mockups and equipment are also available at the Nuclear Training Center for this purpose. Spare control rod drive mechanisms, control rod hydraulic units, and refueling tools are among the materials available for training. Additional mockup materials will be obtained as plant operation approaches. The PSEG Nuclear LLC ALARA program is based on successful ALARA program elements such as those described in INPO Good Practice document 82-001-OEN-08A.

12.1.4 References

12.1-1 T. D. Murphy, "A Compilation of Occupational Radiation Exposure from Light Water Cooled Nuclear Power Plants; 1969-1973," WASH-1311, USAEC Directorate of Licensing, May 1974.

12.1-2 U.S Nuclear Regulatory Commission, "Occupational Radiation Exposure at Light Water Cooled Power Reactors; 1969-1974," NUREG-75/032, June 1975.

12.1-3 U.S. Nuclear Regulatory Commission, "Occupational Radiation Exposure at Light Water-Cooled Power Reactors; 1969-1975, " NUREG-0109, August 1976.

12.1-4 "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," (AIF/NESP-005), SAI Services, Inc, September 1975.

(Historical Information)

TABLE 12.1-1

ALARA PLANT MODEL REVIEW

PLANT SYSTEMS, AREAS, AND PERSONNEL ACTIVITIES[1]

Breathing air
Condensate/feedwater
Condensate demineralizer
Condenser air removal
Core spray
Control rod drive maintenance/repair
Drywell
Fuel pool cooling and cleaning (FPCC)
Fuel handling and storage
High pressure coolant injection (HPCI)
Inservice inspection (ISI)
Liquid radwaste
Liquid radwaste drain system
Liquid/gaseous sampling
Main condenser
Main steam
Nuclear boiler
Off-gas treatment
Post-accident access/shielding
Reactor core isolation cooling (RCIC)
Residual heat removal (RHR)
Reactor Water Cleanup System (RWCU)
Resin regeneration
Radwaste solidification and volume reduction
Transversing in-core probes

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(1) Listed alphabetically.

(Historical Information)

TABLE 12.1-2

ALARA PLANT MODEL REVIEW
REVIEW DOCUMENTS

P&ID
Equipment specifications
Vendor drawings
Operating, control, and maintenance manuals
Plant model
Penetration drawings (matrix drawings showing location and sealing details)
Plumbing and drainage drawings
HVAC area drawings and air flow diagrams
Protective coating drawings
Instrument location drawings
Lighting area drawings
Radiation protection drawings (minimum shielding and zoning)
Radioactive pipe classification drawings
Plant design drawings, equipment location, area drawings, and system isometrics

12.2 RADIATION SOURCES

This section identifies and discusses the sources of radiation that form the basis for the radiation shielding design required for in-plant radiation protection, for the plant ventilation design to control airborne radioactive contamination, and for dose assessment. Sources due to normal full power operations and accident conditions are addressed.

The nuclear fission process in the reactor core produces a large number of radiation sources. Some of these sources are stationary and remain within the reactor vessel and its internal and external structures. Other radiation sources are released and transported in the form of radioactive fission and activation products. The transport and subsequent distribution of these sources is by way of the Reactor Coolant System (RCS), the Main Steam Supply System, and the numerous auxiliary plant support systems. Following an accident, additional and much stronger sources could be released to the RCS, the engineered safeguard systems, and throughout large volumes of the primary and secondary containments.

12.2.1 Contained Sources

The shielding design source terms for normal full power plant operations are based on a noble gas fission product release rate of 0.50 Ci/s (after 30 minutes of decay) and the corresponding fission, activation, and corrosion product concentrations in the primary coolant. The guidance provided in ANSI N237 was not followed for Hope Creek Generating Station (HCGS). Acceptance Criteria II.6 of SRP Section 12.2 addresses the use of ANSI Standard N237-1976 "Source Term Specification," and its use in establishment of a typical long term concentration of principal radionuclides in fluid streams. GE developed and upgraded source terms based on operating plant experience. These specifications are more applicable to boiling water reactors (BWRs). The specific alternate methods and design basis used for calculating source terms and their magnitudes are described in Section 11.1.

Throughout most of the primary coolant and main steam systems, activation products, principally N^{16}, are the primary radiation sources for the shielding design.

Other activation products, such as Co^{60}, and certain fission products become important as built-up or accumulation sources in filters, demineralizers, or in plateout processes at pipe and equipment surfaces. The majority of these sources remain contained within the confines of the closed plant piping systems and the various tank and container facilities.

Basic reactor data and core region descriptions used for this section are listed in Section 12.2.1.1.1.

The shielding design radiation source terms are presented by building, location, and system. Locations of the equipment discussed in this section are shown on the shielding and radiation zoning drawings, Figures 12.3-22 through 12.3-29 and Plant Drawings N-1031 through N-1038, N-1041 through N-1047 and N-1011 through N-1016. Detailed data on source descriptions for shielded areas are presented in Tables 12.2-42, 12.2-107, and 12.2-133.

Shielding source terms presented in this section, and associated tables, are based on conservative assumptions for the system and equipment operations and characteristics to provide conservative radioactivity concentrations for the shielding design. For all systems transporting radioactive materials, conservative allowance is made for transit decay while at the same time providing for daughter product formation.

12.2.1.1 Primary Containment (Drywell)

12.2.1.1.1 Reactor Core

This section provides information needed to form a reactor core source model. It also provides multi-group neutron and gamma ray fluxes at important locations at and near the surface of the reactor vessel. This data is required for the calculations

(Historical Information)

determining the primary radiation shielding requirements for the drywell structures, and to establish the radiological environmental conditions inside the primary containment.

Table 12.2-1 presents physical data needed in forming a reactor core source model. The data includes the core thermal power, the power peaking factors (peak to average at core center and core boundary to average), volume fractions for the core, and the water density conditions in the non-core regions.

Table 12.2-2 presents the reactor vessel radial geometry data used in calculations to determine the radial radiation flux distributions at the core mid-plane.

Table 12.2-3 gives the material compositions used in calculating the radial radiation flux distributions at the core mid-plane. Tables 12.2-4 and 12.2-5 present calculated gamma ray and neutron fluxes at core mid-plane at the outside surface of the reactor pressure vessel (RPV), and at the outside surface of the biological shield, respectively. The gamma fluxes include those resulting from capture or inelastic scattering of neutrons within the RPV and core shroud, and the gamma radiation resulting from prompt fission and fission product decay. The calculated gamma ray fluxes in this section do not include any provisions for scattering from points outside of the vessel. Also, there are no provisions for gamma ray fluxes from deposits of radioactive isotopes within the vessel or from the neutron activation of reactor vessel materials.

The largest radiation sources after reactor shutdown are the decaying fission products in the fuel. Table 12.2-6 lists the reactor core fission product gamma sources at 3 days after shutdown. Secondary sources are the structural material activation of the RPV, its internals, and also the activated corrosion products accumulated or deposited in the internals of the RPV.

(Historical Information)

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the Reactor Coolant System (RCS) are both fission products released from a corresponding percentage of defective fuel, based on a noble gas fission product release rate of 0.50 Ci/s (after 30 minutes of decay), and activation and corrosion products that are circulated in the reactor coolant. These sources are listed in Tables 12.2-7 through 12.2-11. The N^{16} concentration in the reactor coolant is assumed to be 40 μCi/g of coolant at the reactor recirculation outlet nozzle.

12.2.1.1.3 Main Steam System

Radiation sources in the Main Steam System piping include activation gases, principally N^{16}, and the activated corrosion products and fission products carried over to the steam system.

The N^{16} concentration in the main steam is assumed to be 50 μCi/g of steam leaving the reactor vessel at the main steam outlet nozzle. The fission product activity for the purpose of personnel shielding design corresponds to an off-gas release rate of 500,000 μCi/s at 30 minutes decay from the reactor steam nozzle. Carryover fractions for activity into the steam system are 100 percent for the gaseous fission and activation products. Carryover of radioiodines from the reactor coolant water to the steam is taken as 2 percent by weight of the reactor water. Carry-over of other radioisotopes is taken as 0.1 percent by weight of the reactor water. Main steam radiation sources are shown in Tables 12.2-7 through 12.2-11.

12.2.1.1.4 Drywell Sumps

The concentrations of radioisotopes used for shielding design for the drywell equipment and floor drain sumps are listed in Tables 12.2-12 and 12.2-13.

12.2.1.2 Reactor Building

12.2.1.2.1 Reactor Water Cleanup System

Radiation sources in the Reactor Water Cleanup (RWCU) System consist of those radioisotopes carried in the reactor coolant water. The activity inventory is based on reactor coolant water radioactivities corrected for transit times to the individual components of the RWCU system.

The radioisotopes for the RWCU recirculation pumps, regenerative and nonregenerative heat exchangers, and associated piping are based on the inlet reactor coolant concentrations given in Tables 12.2-14 through 12.2-19, allowing for decay due to transit time.

Radioisotope concentrations in the filter demineralizers, holding and transfer pumps, and the backwash receiving tank are based on the accumulation of fission and activation products. Tables 12.2-20 through 12.2-23 provide the shielding design source terms for these components.

12.2.1.2.2 Residual Heat Removal System

The radioactive sources in the Residual Heat Removal (RHR) System are evaluated for the system operating in the reactor shutdown mode. In this mode, the system recirculates reactor coolant to remove reactor core decay heat. The system is described in Section 5.4.7. The system is operated from approximately four hours after shutdown until the end of the refueling period. The radiation source in the RHR system is based on the maximum estimated activity due to the iodine spike effect at four hours of decay following shutdown. The source terms listed in Table 12.2-24 are used for the shielding calculations for this system.

12.2.1.2.3 Reactor Core Isolation Cooling System

Components of the Reactor Core Isolation Cooling (RCIC) System that contain radiation sources during normal power operation are the RCIC turbine steam inlet and exhaust piping. The steam radioactivity, as discussed in Section 12.2.1.1.3, decayed for the appropriate transit time, is used for the shielding calculations for this system and is listed in Table 12.2-25.

12.2.1.2.4 High-Pressure Coolant Injection System

The radiation sources for the High Pressure Coolant Injection (HPCI) System are the HPCI turbine steam inlet and exhaust piping. The steam radioactivity, as discussed in Section 12.2.1.1.3, decayed for the appropriate transit time, is used for the shielding calculations for this system and is listed in Table 12.2-26.

12.2.1.2.5 Core Spray System

During testing, the Core Spray System components use condensate from the condensate storage tank or draw water from the suppression pool. Source terms based on suppression pool activities listed in Table 12.2-27 are used for shielding the core spray equipment.

12.2.1.2.6 Spent Fuel Storage and Transfer Sources

The predominant radiation sources in the spent fuel storage and transfer areas are the spent fuel assemblies and activated control rods. For shielding design purposes, the spent fuel pool is assumed to contain 3667 fuel assemblies. These spent fuel assemblies are assumed to have 72 hours of decay before being stored. Shielding design source terms for spent fuel assemblies are shown in Table 12.2-28. The shielding design source terms have been verified to bound the source term for the expected storage of spent fuel and irradiated components. The shielding design source terms will remain bounding up to the inclusion of 6241 storage locations in the spent

(Historical Information)

fuel storage area. The storage locations can hold fuel assemblies in an as-built condition or in a consolidated configuration. The spent fuel assemblies are assumed to have 72 hours of decay before being stored in their as-built condition and 14 days of decay before being stored in a 2 to 1 consolidated configuration.

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(Historical Information)

12.2.1.2.7 Fuel Pool Cooling and Cleanup System

Sources in the Fuel Pool Cooling and Cleanup (FPCC) System are primarily a result of a transfer of radioactive isotopes from the reactor coolant into the spent fuel pool during refueling operations. The reactor coolant activities for fission, corrosion, and activation products (Tables 12.2-7 through 12.2-11) are decayed for the amount of time required to remove the reactor vessel head following shutdown, are reduced by operation of the RWCU system filter demineralizers following shutdown, and are diluted by the total volumes of the water in the reactor vessel, refueling pool, and spent fuel pool. See Table 12.2-29. This activity then undergoes subsequent accumulation and decay on the FPCC filter demineralizers. The FPCC filter demineralizer resins are backwashed periodically into the waste sludge phase separator. Sources for FPCC heat exchangers and system piping are based on crud plateout on these components. Table 12.2-30 provides the FPCC filter demineralizer shielding design source terms, and Table 12.2-31 provides the shielding design source terms for the FPCC heat exchangers and associated piping.

12.2.1.2.8 Radiation Sources in the Traversing In-core Gamma Probe System

The radiation source data for the traversing in-core gamma probe (TIP) system is provided in Table 12.2-32.

12.2.1.2.9 Reactor Startup Sources

The reactor startup sources are shipped to the site in a specially designed shielding cask. The sources are transferred underwater from the cask and loaded into beryllium containers. This is then loaded into the reactor while remaining underwater. The sources remain within the reactor for their lifetime. Thus, there are no unique shielding requirements after reactor operation. Table 12.2-33 gives information on source strength and other important data.

(Historical Information)

12.2.1.2.10 Control Rod Drives (CRDs)

Shielding design source terms for the control rod drives (CRDs) after removal from the RPV are shown in Table 12.2-34.

12.2.1.2.11 Reactor Building Ventilation Systems

The Reactor Building Ventilation System (RBVS) exhaust air filters, as part of the Reactor Building Heating, Ventilation, and Air Conditioning (HVAC) Systems, contain sources of radioactivity. Table 12.2-35 shows the RBVS exhaust air filter shielding design source terms. The shielding design source terms for the primary containment prepurge filters are given in Table 12.2-36.

12.2.1.2.12 Reactor Vessel Steam Separator and Steam Dryer

Shielding design source terms for the steam separator and steam dryer are given in Table 12.2-37. These shielding design source terms are based on estimated crud plateout activity from operating plants.

12.2.1.2.13 Reactor Building Sumps

The concentrations of radioisotopes used for the shielding design for the Reactor Building Equipment and floor drain sumps are listed in Tables 12.2-38 and 12.2-39.

12.2.1.2.14 RPV Head and Spent Fuel Cask

The shielding design source terms for the RPV head washdown area are shown in Table 12.2-40. Shielding design source terms for the spent fuel shipping cask are shown in Table 12.2-41.

12.2.1.2.15 Reactor Building Source Descriptions

A listing and detailed description of the shielding design radiation sources for the Reactor Building are shown in Table 12.2-42.

12.2.1.3 Service and Radwaste Areas of the Auxiliary Building

The numerous plant radioactive wastes are collected and processed, safely inside the service and radwaste areas of the auxiliary building. Radioactive wastes are in liquid, gaseous, and solid phases. Each type of radwaste, and its associated radiation sources, is contained and handled in a specific manner to ensure maximum radiation protection and minimum plant contamination. Detailed system and equipment descriptions are provided in Sections 11.2, 11.3, and 11.4.

12.2.1.3.1 Liquid Radwaste System

The liquid radwastes are collected and contained inside the service and radwaste areas from the following systems: fuel pool cleanup, equipment drain, floor drain, chemical waste, condensate, decontamination, and reactor water cleanup. Further processing will be accomplished in the solidification and volume reduction system discussed in Section 12.2.1.3.3. The liquid radwaste system shielding design sources are radioisotopes, including fission and corrosion products, present in the reactor coolant. The components of the Liquid Radwaste System contain varying amounts of radioactivity, depending on the origin of the handled radwaste and the system and equipment design.

The concentrations of radioisotopes used for the shielding design for pipes, pumps, tanks, filters, demineralizers and evaporators, and equipment and floor drain sumps are listed in Tables 12.2-43 through 12.2-86. Shielding for each component of the Liquid Waste Management System is based on reactor coolant radioactivity concentrations given in Tables 12.2-7 through 12.2-11.

(Historical Information)

12.2.1.3.2 Off-gas Treatment System

Radioactive off-gases are contained and processed in the Off-gas Treatment System. The major radiation sources of this system, located in the service and radwaste areas of the Auxiliary Building, are found in the off-gas recombiner cells and the Off-gas Charcoal Treatment System.

The shielding design radiation source terms for the recombiner and charcoal treatment system components are based on the expected transit times for N^{16}, noble gases, and the formation and accumulation of noble gas daughter products. Eighty percent of the N^{16} and 100 percent noble gases are assumed to be removed from the main condensers by the steam jet air ejectors. These gases pass from the steam jet air ejector to the recombiner equipment and to the charcoal treatment system. The charcoal treatment system functions to filter, as well as to delay, the release of the radioactive offgases. Off-gas release to the environment is through the plant north vent. The shielding design source terms for the piping, recombiner components, and charcoal treatment system equipment are presented in Tables 12.2-87 through 12.2-97.

12.2.1.3.3 Deleted

(Historical Information)

12.2.1.3.4 Service and Radwaste Area Ventilation Systems

Components of the ventilation system inside the service and radwaste areas that contain sources of radioactivity are the radwaste exhaust and the Reactor Building vent system filters. Tables 12.2-106 and 12.2-35 show the exhaust air filter shielding design radiation source terms.

12.2.1.4 Turbine Building

12.2.1.4.1 Main Steam and Power Conversion Systems

Radiation sources for piping and equipment that contain main steam are based on the radioisotopes carried into the main steam from the reactor coolant. The sources include fission product gases and halogens, particulate fission and corrosion products, and gaseous activation products as discussed in Section 12.2.1.1.3. Steam density variations and steam transit times through equipment and pipes are factored into the shielding source term evaluation to account for volumetric dilution effects, radiological decay, and daughter product generation. Tables 12.2-108 through 12.2-115 show the shielding design radiation source terms for the following major components that contain main steam as the dominant source: moisture separators, crossaround piping, feedwater heaters, drain

coolers, steam seal evaporator, reactor feed pump turbines (RFPT), and the steam packing exhaust condenser.

12.2.1.4.2 Condensate and Feedwater Systems

The radiation sources in the condensate and feedwater systems are based on decayed main steam radioactivity. See Section 12.2.1.1.3. Eighty percent of the N^{16} and 100 percent of the noble gases are assumed to be removed from the condensate and feedwater systems by the main condenser air removal system. The radiation sources in the hotwell are shown in Table 12.2-116. They are negligible as radiation sources in the remainder of the condensate and feedwater systems. The hotwell is designed for a 3-minute holdup of condensate, and therefore N^{16} radioactivity at the condenser outlet is negligible. Particulate fission products, activated corrosion products, and the particulate daughter products from the decay of fission product gases in transit through the turbine and condenser are the inlet radiation sources to the condensate system. Tables 12.2-117 through 12.2-123 provide the shielding design radiation source terms for the condensate pumps and their associated piping, the condensate filter demineralizers, the resin mix and hold tank, the anion and cation regeneration tanks, the ultrasonic resin cleaner, and the shielding design radiation source terms for the feedwater system.

12.2.1.4.3 Steam Jet Air Ejector System

The steam jet air injectors (SJAEs) are the only major equipment of the plant Off-gas Treatment System located inside the Turbine Building. Shielding design radiation sources in the Off-gas Treatment System are developed from noble gases and other noncondensable gases removed from the main condenser, and the radioactivity entering with the extraction steam to the SJAEs. The extraction steam radioactivity entering is based on the main steam radioactivity, as described in Section 12.2.1.1.3, and decayed for the expected transit time to the SJAE system. Eighty percent of the N^{16} and 100 percent of the noble gases are assumed to be

removed from the condenser by the SJAE system. The specific activities and quantities of radioactivity, including particulate daughters, in the SJAE system condenser and the mechanical vacuum pump to be used for shielding design calculations, are shown in Tables 12.2-124 through 12.2-129

12.2.1.4.4 Turbine Building Drain Sumps

The shielding design source terms for the Turbine Building drain sumps are listed in Tables 12.2-130 through 12.2-132.

12.2.1.5 Shielding Design Sources Resulting from Design Basis Accidents

The shielding design for the control room and other vital plant areas is based on radiation sources resulting from design basis accidents (DBAs). Vital areas and post-accident radiation sources are identified in accordance with requirements set down in NUREG-0737, Item II.B.2, Reference 12.2-5. TID-14844, Reference 12.2-7, source terms are used in the post-accident shielding review. Section 12.3.2.2.6 provides the information on source activities, dilution volumes, and systems containing these activities, as well as the airborne activities in various plant areas. The initial inventory of radioisotopes for these sources is given in Table 12.2-134 through 12.2-136.

12.2.1.6 Stored Radioactivity

There are three sources of radioactivity which are not stored inside the plant structures.

The Low Level Radwaste Storage Facility (LLRSF) provides a safe, secure space for temporary storage of low level radwaste generated by both the Salem and Hope Creek Generating Stations when offsite disposal or storage is not available. The shielding is discussed in Section 12.3.2.2.11. The LLRSF source terms are shown in Table 12.2-145.

The second source of radioactivity not stored inside the plant structures is the condensate storage tank (CST). The CST contains low concentrations of radioisotopes from the condensate system and from the refueling water system. A dike is provided around the CST so that, in case of leakage or gross tank failure, the radioactive condensate is contained, and uncontrolled release is prevented. Special access barrier features prevent personnel from uncontrolled access to the slightly radioactive tank. The CST source terms are shown in Table 12.2-137.

Provisions have been made to recycle the water from the CST through the condensate demineralizers.

The third source of radioactivity not stored inside the plant structure is the Independent Spent Fuel Storage Installation (ISFSI). The ISFSI is operated under the general license provisions of 10 CFR 72, Subpart K. It provides additional on-site storage for HCGS spent nuclear fuel removed from the spent fuel pool stored in dry, seal welded stainless steel storage canisters. Each storage canister is located inside a ventilated concrete overpack at the ISFSI that provides structural protection, decay heat removal, and shielding. The storage canisters are considered leak-tight under all design basis normal, off-normal, and accident conditions as described in the spent fuel storage system FSAR. Therefore, there are no effluent releases from the dry storage systems at the ISFSI. Source terms and other design and operating information for the dry spent fuel storage system can be found in the storage system 10 CFR 72 FSAR. A discussion of the off-site dose contribution due to direct radiation from the ISFSI may be found in the 10 CFR 72.212 evaluation report for the ISFSI.

Under normal conditions, no other radiation sources are stored outside plant structures without prior approval of the radiation protection engineer. Spent fuel is stored in the spent fuel pool until it is placed in the spent fuel shipping cask for offsite transport. All radiation sources contained inside the plant structures are shielded to provide a dose rate of less than 0.5 mrem/h for all areas outside plant structures.

(Historical Information)

12.2.1.7 Special Sources

Special radioactive materials used in the radiochemistry laboratory, and sealed sources used for calibration, require special handling equipment and are shielded accordingly. A listing of these special material sources can be found in Table 12.2-33. Unsealed sources and radioactive samples are handled under conventional hoods that exhaust to the plant ventilation systems. Design features are discussed in Section 12.3.1.

12.2.2 Airborne Radioactive Material Sources

12.2.2.1 Sources of Airborne Radioactivity

The sources of airborne radioactivity found in the various areas of the plant are mostly from process leakage of the systems carrying radioactive gases, steam, and liquids. Depending on the type of system and its physical conditions, such as system pressures and temperatures, leakage is in the form of a gas, steam, liquid, or a mixture of these.

12.2.2.2 Emission of Airborne Radioactive Materials

Radioactive materials become airborne through a number of mechanisms. The most common emission mechanisms are spraying, splashing, flashing, evaporation, and diffusion.

12.2.2.3 Locations of Sources of Airborne Radioactivity

Practically all sources of airborne radioactivity are found in the Reactor and Turbine Buildings and the service and radwaste areas of the Auxiliary Building. Within these structures, the radioactivity is released in equipment cubicles, valve and piping galleries, sampling stations, radwaste handling and shipping areas, cleaning and decontamination areas, fuel transfer and pool storage areas, and repair shops. Insignificant amounts of airborne radioactivity could occur in places such as the radiation protection work area or the condensate storage tank area.

12.2.2.4 Control of Airborne Radioactivity

Ventilation and filtration are the most effective means of controlling airborne radioactive materials. Ventilation flow paths are designed so that air from areas of low potential airborne radioactivity flows into areas with higher potential for airborne radioactivity. Such a flow pattern ensures that radioactivity released in the source locations mentioned above has a low probability to escape into areas with high personnel occupancy requirements, such as corridors, working aisles, and operating floors. Levels of airborne radioactivity are continuously monitored by the area radiation monitor and, the airborne radiological monitoring systems, and they are also periodically checked through surveys of the plant by the radiation protection staff.

(Historical Information)

12.2.2.5 Methodology for Estimating the Concentration of Airborne Radioactive Material Within the Plant

To estimate the airborne radioactive material concentrations at locations within the plant, the following methodology is used:

1. Estimate the total airborne releases (in Ci/yr) for each of the plant enclosures.

2. Estimate a distribution for these releases among the various equipment areas of each enclosure based on operating data and engineering judgment.

3. Determine the annual exhaust flow from each equipment area.

4. Calculate the resultant airborne radionuclide concentration (Ci/cc) in each equipment area based on the release distribution (Ci/yr) and exhaust flow rate (cc/yr).

The following sections discuss each step in the procedure listed above in more detail.

12.2.2.5.1 Estimation of Total Airborne Releases Within the Plant During Normal Full Power Operations

The estimated quantities of airborne radioactive material produced in the plant enclosures are given in Table 12.2-138. These releases were based upon BWR gaseous and liquid effluents, mentioned in Reference 12.2-1, using a computerized mathematical model (BWR-GALE) for calculating the release of radiological materials in gaseous and liquid effluents. Assumptions applicable to the development of Table 12.2-138 from BWR-GALE are as follows:

1. The drywell and torus releases are taken to be the Containment Building releases calculated by BWR-GALE.

2. The releases from the HCGS Reactor Building correspond to the Auxiliary Building releases calculated by BWR-GALE.

3. Turbine Building releases from BWR-GALE are assumed to include any airborne radioactive material released in the individual rooms and cubicles of the structure.

4. The radwaste area releases from BWR-GALE also include offgas system releases.

5. Tritium releases from BWR-GALE are divided equally between the reactor and turbine buildings.

6. Since the BWR-GALE code for gaseous releases is based on actual operating plant data, releases for both normal operations and anticipated operational occurrences are included.

12.2.2.5.2 Distribution of Airborne Releases Within the Plant

In the approach taken to determine the anticipated distribution of gaseous effluents, it is assumed that all airborne radioactive material originates only within the equipment areas of the plant.

It is further assumed that a major percentage of the release is generated within a few specific areas of each enclosure, with the remainder coming from the "remaining equipment areas." Eighty percent of each enclosure's release is distributed, as described below, among the major contributing areas, and 20 percent is assigned to the remaining equipment areas category. Releases are assumed to be generated continuously throughout the year, except for the drywell, where a 30-day annual release period is used.

(Historical Information)

The basis for the selection and relative contributions of the major areas is the Electric Power Research Institute Report, EPRI NP-495, Reference 12.2-2. This report provides data on the important sources of I^{131} at operating BWRs and uses measured data to determine the relative release rate from each source. The relative release rates for all airborne radionuclides, except tritium from the Reactor Building, are assumed to be directly proportional to the I^{131} release rates. Since the spent fuel pool and the reactor vessel (when it is open during refueling) are the major sources of airborne tritium in the Reactor Building, tritium releases for that enclosure are assigned entirely to the refueling area.

Table 12.2-139 lists the major airborne contributors in each enclosure and the percentage of the total enclosure release assigned to each. Table 12.2-140 also provides the specific equipment areas of the plant associated with the major contributors and the applicable exhaust air flow rates. Note that only those equipment areas that have a significant potential for airborne radioactive material releases are included in the remaining equipment areas category.

12.2.2.5.3 Estimated Airborne Radioactive Material Concentrations Within the Plant

The airborne radionuclide concentrations for each equipment area are calculated for a specific area, by multiplying the appropriate enclosure release, Table 12.2-138, by the applicable release percentage for the area, Table 12.2-139, and dividing that by the area's annual exhaust flow, Table 12.2-140. The resultant concentrations are presented in Tables 12.2-141 through 12.2-144, which also include the fractions of the maximum permissible concentrations in air as defined in 10CFR20, Appendix B, Table I, Reference 12.2-3.

(Historical Information)

12.2.2.5.4 Changes to Source Data Since PSAR

Liquid, solid, and airborne radioactive material sources are not specified in the HCGS PSAR. Section 12.2 has been added to the HCGS FSAR in compliance with Regulatory Guide 1.70, Revision 3, Reference 12.2-4.

12.2.3 References

12.2-1 Nuclear Regulatory Commission, Office of Standard Development, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Boiling Water Reactors," NUREG-0016, Revision 1, January 1979.

12.2-2 Electric Power Research Institute, "Sources of Radioiodine at Boiling Water Reactors," EPRI NP-495, Project 274-1, Final Report, February 1978.

12.2-3 Nuclear Regulatory Commission, "Standards for Protection Against Radiation" "Code of Federal Regulations," Title 10, Part 20, November 1975.

12.2-4 Nuclear Regulatory Commission, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Regulatory Guide 1.70, Revision 3, November 1978.

12.2-5 Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

12.2-6 251 Standard Safety Analysis Report, General Electric, for BWR/6, Revision 22.

(Historical Information)

12.2-7 Technical Information Document, TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites by J.J. DiNunno, R.E. Baker, F.D. Anderson and R.L. Waterfield, March 23, 1962.

(Historical Information)

TABLE 12.2-1

BASIC REACTOR DATA[1]

1. Reactor rated thermal power (100 percent) 3339 megawatts

2. (DELETED)

3. Core power peaking factors:

   a. At core center:

   $$\left.\frac{P_{max}}{P_{ave}}\right|_{Z} \text{(axial)} \quad 1.5$$

   $$\left.\frac{P_{max}}{P_{ave}}\right|_{R} \text{(radial)} \quad 1.4$$

   b. At core boundary:

   $$\left.\frac{P_{max}}{P_{ave}}\right|_{Z} \text{(axial)} \quad 0.5$$

   $$\left.\frac{P_{max}}{P_{ave}}\right|_{R} \text{(radial)} \quad 0.7$$

(Historical Information)

TABLE 12.2-1 (Cont)

4. Core volume fractions:

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

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

6. Average water steam density above core:

   a. In the plenum region — 0.23 g/cc

   b. Above the plenum (homogenized) — 0.6 g/cc

7. Average steam density — 0.036 g/cc

(1) This table, which includes volume fractions, reactor power, and power distribution, represents the physical data required to form the reactor vessel model.

(Historical Information)

TABLE 12.2-2

REACTOR VESSEL AND SHIELD RADIAL GEOMETRY USED IN CALCULATIONS TO DETERMINE RADIAL FLUX DISTRIBUTION AT REACTOR CORE MIDPLANE

Region Description		Region Thickness (cm)	Cumulative Thickness (cm)	Material
1.	Active fuel zone	242.48	242.48	Core
2.	Water	15.49	257.97	Water
3.	Core shroud	5.08	263.05	Stainless steel
4.	Water	58.26	321.31	Water
5.	Pressure vessel liner	0.95	322.26	Stainless steel
6.	Pressure vessel	16.35	338.61	Carbon steel
7.	Air	64.45	403.06	Air
8.	Biological shield (inner liner)	1.91	404.97	Carbon steel
9.	Biological shield(1)	47.3	452.27	Magnetite-Ilmenite
10.	Biological shield (outer liner)	3.81	456.08	Carbon steel
11.	Air gap	168.76	624.84	Air

(Historical Information)

TABLE 12.2-2 (Cont)

Region Description		Region Thickness (cm)	Cumulative Thickness (cm)	Material
12.	Primary containment	182.88	807.72	Ordinary concrete

(1) Magnetite-Ilmenite aggregate mixture.

TABLE 12.2-3

MATERIAL COMPOSITION TO DETERMINE RADIAL FLUX DISTRIBUTIONS AT REACTOR CORE MIDPLANE

		Material (10^{24} atoms/cc)					
Element	Core	Water	SS-304	Steel	Carbon Concrete(1)	Air	Ordinary Concrete
H	1.836-02(2)	4.888-02	–	–	6.24-03	–	7.84-03
O	2.257-02	2.444-02	–	–	4.42-02	1.12-05	4.41-02
Mo	5.93-04	–	–	–	–	–	–
U^{235}	1.24-04	–	–	–	–	–	–
U^{238}	5.46-03	–	–	–	–	–	–
Mg	–	–	–	–	8.96-04	–	1.43-04
Al	–	–	–	–	1.23-03	–	2.39-03
Si	–	–	1.69-03	4.19-04	2.15-03	–	1.57-02
Ca	–	–	–	–	3.32-03	–	2.91-03
Ti		–	–		5.29-03	–	–
Mn	–	–	1.73-03	8.59-04	3.47-05	–	–
Fe	–	–	5.76-02	8.40-02	1.20-02	–	3.09-04
C	–	–	3.18-04	1.10-03	–	–	–
Cr	–	–	1.73-02	–	–	–	–
Ni	–	–	8.06-03	–	–	–	–
N	–	–	–	–	–	4.17-05	–
Na	–	–	–	–	–	–	1.05-03
K	–	–	–	–	–	–	6.91-04
S	–	–	–	–	–	–	5.30-05

(1) Mixture magnetite-ilmenite (Il-209)

(2) $1.836\text{-}02 = 1.836 \times 10^{-2}$

(Historical Information)

TABLE 12.2-4

CALCULATED GAMMA RAY AND NEUTRON FLUXES AT OUTSIDE SURFACE OF RPV AT CORE MIDPLANE(1)(2)(4)

Gamma Energy (Upper MeV)	Gamma Flux (γ/cm^2-s)(3)	Neutron Energy (Upper ev)	Neutron Flux (η/cm^2-s)
10.0	2.44+07	15.0+6	3.58+05
8.0	2.91+08	12.2+6	1.29+06
6.5	2.52+08	10.0+6	2.97+06
5.0	2.88+08	8.18+6	5.80+06
4.0	5.06+08	6.36+6	8.38+06
3.0	3.86+08	4.96+6	6.14+06
2.5	6.62+08	4.06+6	9.78+06
2.0	5.46+08	3.01+6	1.10+07
1.66	6.50+08	2.46+6	3.56+06
1.33	8.20+08	2.35+6	2.13+07
1.0	6.20+08	1.83+6	6.95+07
0.8	4+08	1.11+6	1.96+08
0.6	1.55+09	0.55+6	5.29+08
0.4	9.58+08	0.11+6	3.74+08
0.3	1.52+09	3.35+3	8.72+07
0.2	1.44+09	5.83+2	7.13+07
0.1	8.21+07	1.01+2	4.88+07
0.05	2.25+05	2.9+1	3.25+07
		1.07+1	3.23+07
		3.06	1.98+07
		1.12	1.36+07
		0.414	1.10+07

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(1) At 100 percent power

(2) Geometry, Table 12.2-2

(3) 2.44+07 = 2.44×10^7

(4) The values are based on 3293 MW_t. Multiply the flux values by 1.015 for uprated power of 3339 MW_t.

(Historical Information)

TABLE 12.2-5

CALCULATED GAMMA RAY AND NEUTRON FLUXES AT OUTSIDE SURFACE OF BIOLOGICAL SHIELD

AT CORE MIDPLANE(1)(2)(4)

Gamma Energy (Upper MeV)	Gamma Flux (γ/cm^2-s)(3)	Neutron Energy (Upper ev)	Neutron Flux (η/cm^2-s)
10.0	8.30+04	15.0+6	8.56+02
8.0	1.05+06	12.2+6	2.92+03
6.5	1.02+06	10.0+6	6.77+03
5.0	9.52+05	8.18+6	1.38+04
4.0	1.36+06	6.36+6	1.96+04
3.0	8.61+05	4.96+6	1.46+04
2.5	1.10+06	4.06+6	1.92+04
2.0	8.95+05	3.01+6	2.87+04
1.66	1.06+06	2.46+6	1.30+04
1.33	1.31+06	2.35+6	4.61+04
1.0	9.83+05	1.83+6	1.17+05
0.8	1.24+06	1.11+6	2.24+05
0.6	2.77+06	0.55+6	4.25+05
0.4	1.58+06	0.11+6	3.64+05
0.3	2.61+06	3.35+3	1.40+05
0.2	3.71+06	5.83+2	1.43+05
0.1	7.72+05	1.01+2	1.16+05
0.05	4.21+03	2.9+1	8.24+04
		1.07+1	9.96+04
		3.06	7.68+04
		1.12	6.77+04
		0.414	2.41+05

(1) At 100 percent power

(2) Geometry, Table 12.2-2

(3) $8.30 + 04 = 8.30 \times 10^4$

(4) The values are based on 3293 MW_t. Multiply the flux values by 1.015 for uprated power of 3339 MW_t.

(Historical Information)

TABLE 12.2-6

REACTOR CORE FISSION PRODUCT GAMMA SOURCES(1)(3)

Gamma Energy (Upper MeV)	Gamma Activity (MeV/sec)(2)
0.10	2.59+17
0.20	1.36+18
0.30	7.87+17
0.40	1.31+18
0.50	3.58+18
0.60	1.84+18
0.80	1.58+19
1.00	2.99+18
1.25	1.05+18
1.50	9.72+17
1.75	8.45+18
2.00	2.73+17
2.50	6.81+17
3.00	5.07+17
4.00	5.35+15
5.00	4.45+08
TOTAL	3.99+19

(1) Design basis:
- a) Reactor power at 3293 MWt
- b) Equilibrium core with 3 year burn
- c) 3 days after shutdown

(2) $2.59+17 = 2.59 \times 10^{17}$

(3) The values are based on 3293 MW_t. Multiply the flux values by 1.015 for uprated power of 3339 MW_t.

(Historical Information)

TABLE 12.2-7

NOBLE GAS SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Half-life	Steam Specific Activity[2] (μCi/g)	Reactor Water Specific Activity (μCi/g)
Kr-83m	1.86 h	9.2-03[3]	-
Kr-85m	4.4 h	1.7-02	-
Kr-85	10.74 yr	5.4-05	-
Kr-87	76.0 min	5.4-02	-
Kr-88	2.79 h	5.4-02	-
Kr-89	3.18 min	3.5-01	-
Kr-90	32.3 s	7.6-01	-
Kr-91	8.6 s	9.2-01	-
Kr-92	1.84 s	9.2-01	-
Kr-93	1.29 s	2.7-01	-
Kr-94	1.0 s	6.5-02	-
Kr-95	0.5 s	5.9-03	-
Xe-131m	11.96 day	3.8-05	-
Xe-133m	2.26 day	8.1-04	-
Xe-133	5.27 day	2.2-02	-
Xe-135m	15.7 min	7.0-02	-
Xe-135	9.16 h	5.9-02	-
Xe-137	3.82 min	4.1-01	-
Xe-138	14.2 min	2.4-01	-
Xe-139	40.0 s	7.6-01	-
Xe-140	13.6 s	8.1-01	-
Xe-141	1.72 s	6.5-01	-
Xe-142	1.22 s	2.0-01	-
Xe-143	0.96 s	3.2-02	-
Xe-144	9.0 s	1.5-03	-
Total		6.7+00	

(Historical Information)

TABLE 12.2-7 (Cont)

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(1) All fission product specific activities are based on an off-gas release rate of 0.5 Ci/s at 30 minutes decay.

(2) At the reactor nozzle

(3) $9.2\text{-}03 = 9.2 \times 10^{-3}$

(Historical Information)

TABLE 12.2-8

HALOGEN SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Half-life	Steam Specific Activity(2)(3) (□Ci/g)	Reactor Water Specific Activity (□Ci/g)
Br^{83}	2.40 h	1.50-03(4)	7.5-02
Br^{84}	31.8 min	2.7-03	1.4-01
Br^{85}	3.0 min	1.7-03	8.5-02
I^{131}	8.06 day	1.3-03	6.5-02
I^{132}	2.28 h	1.2-02	6.0-01
I^{133}	20.8 h	8.9-03	4.5-01
I^{134}	52.3 min	2.4-02	1.2+00
I^{135}	6.7 h	1.3-02	6.5-01
Total		6.5-02	3.3+00

(1) The Halogen fission product release rate has been normalized to 3500 □Ci/s I^{131} from defective fuel (t=0). This corresponds to a noble gas fission product release rate of 500,000 □Ci/s at t=30 minutes decay.

(2) At the reactor nozzle

(3) A 2 percent-by-weight carryover from reactor water to steam is assumed.

(4) $1.50\text{-}03 = 1.50 \times 10^{-3}$

(Historical Information)

TABLE 12.2-9

OTHER FISSION PRODUCT SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Half-life	Steam Specific Activity[2][3] (μCi/g)	Reactor Water Specific Activity (μCi/g)
Sr-89	50.8 day	1.6-05[4]	1.6-02
Sr-90	28.9 yr	1.2-06	1.2-03
Sr-91	9.67 h	3.5-04	3.5-01
Sr-92	2.69 h	5.5-04	5.5-01
Zr-95	65.5 day	2.0-07	2.0-04
Zr-97	16.8 h	1.6-07	1.6-04
Nb-95	35.1 day	2.1-07	2.1-04
Mo-99	66.6 h	1.1-04	1.1-01
Tc-99m	6.00 h	1.4-03	1.4+00
Tc-101	14.2 min	7.0-04	7.0-01
Ru-103	39.8 day	9.5-08	9.5-05
Ru-106	368 day	1.3-08	1.3-05
Te-129m	34.1 day	2.0-07	2.0-04
Te-132	78.0 h	2.5-04	2.5-01
Cs-134	2.06 yr	8.0-07	8.0-04
Cs-136	13.0 day	5.5-07	5.5-04
Cs-137	30.2 yr	1.2-06	1.2-04
Cs-138	32.3 min	9.5-04	9.5-01
Ba-139	83.2 min	8.0-04	8.0-01
Ba-140	12.8 day	4.5-05	4.5-02
Ba-141	18.3 min	8.5-04	8.5-01
Ba-142	10.7 min	8.5-04	8.5-01
Ce-141	32.53 day	2.0-07	2.0-04
Ce-143	33.0 h	1.8-07	1.8-04
Ce-144	284.4 day	1.8-07	1.8-04
Pr-143	13.58 day	1.9-07	1.9-04

(Historical Information)

TABLE 12.2-9 (Cont)

Isotope	Half-life	Steam Specific Activity(2)(3) (μCi/g)	Reactor Water Specific Activity (μCi/g)
Nd-147	11.06 day	7.0-08	7.0-05
Np-239	2.35 day	1.2-03	1.2+00
Total		8.1-03	8.1+00

(1) All fission product specific activities are based on an offgas release rate of 0.5 Ci/s at 30 minutes decay.

(2) At the reactor nozzle

(3) A 0.1 percent-by-weight carryover from reactor water to steam is assumed.

(4) 1.6-05 = 1.6×10^{-5}.

TABLE 12.2-10

COOLANT ACTIVATION PRODUCT SHIELDING DESIGN SOURCE TERMS

Isotope	Half-life	Steam Specific Activity(1) (μCi/g)	Reactor Water Specific Activity(2) (μCi/g)
N^{13}	9.99 min	6.5-03(3)(4)(5)	4.0-02
N^{16}	7.13 s	5.0+01(4)(5)	4.0+01
N^{17}	4.14 s	1.6-02(4)(5)	6.3-03
O^{19}	26.8 s	8.0-01	6.9-01
F^{18}	109.8 min	4.0-03	4.0-03
Total		5.1+01	4.1+01

(1) At the reactor nozzle

(2) At the recirculation outlet

(3) $6.5\text{-}03 = 6.5 \times 10^{-3}$

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-11

NONCOOLANT ACTIVATION PRODUCT SHIELDING DESIGN SOURCE TERMS

Isotope	Half-life	Steam Specific Activity[1][2] (μCi/g)	Reactor Water Specific Activity (μCi/g)
Na-24	15.0 h	2.0-06[3]	2.0-03
P-32	14.31 day	2.0-08	2.0-05
Cr-51	27.8 day	5.0-07	5.0-04
Mn-54	313.0 day	4.0-08	4.0-05
Mn-56	2.58 h	5.0-05	5.0-02
Co-58	71.4 day	5.0-06	5.0-03
Co-60	5.25 yr	5.0-07	5.0-04
Fe-59	45.0 day	8.0-08	8.0-05
Ni-65	2.55 h	3.0-07	3.0-04
Zn-65	243.7 day	2.0-09	2.0-06
Zn-69m	13.7 h	3.0-08	3.0-05
Ag-110m	253.0 day	6.0-08	6.0-05
W-187	23.9 h	3.0-06	3.0-03
Total		6.2-05	6.2-02

(1) At the reactor nozzle

(2) A 0.1 percent-by-weight carryover from reactor water to steam is assumed

(3) 2.0-06 = 2.0×10^{-6}

(Historical Information)

TABLE 12.2-12

DRYWELL EQUIPMENT DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	6.78-02
Br-84	8.66-02
I-131	6.60-02
I-132	5.68-01
I-133	4.51-01
I-134	8.90-01
I-135	6.33-01
Sr-89	1.63-02
Sr-91	3.45-01
Sr-92	5.04-01
Mo-99	1.11-01
Tc-99m	1.36+00
Tc-101	2.63-01
Te-132	2.53-01
Cs-138	5.91-01
Ba-139	6.65-01
Ba-140	4.57-02
Ba-141	3.83-01
Ba-142	2.62-01
Np-239	1.21+00
Mn-56	4.55-02
La-142	6.19-02
La-141	3.53-02
Y-92	4.05-02
Y-91m	5.71-02
Total	9.03+00

(Historical Information)

TABLE 12.2-12 (Cont)

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(1) Design basis:

a. Sump volume = 785 gallons

b. Input rate of 5.36 gpm for 50 minutes

c. Input activity 1.0 x primary coolant activity, Tables 12.2-8 through 12.2-11.

(2) 6.78-02 = 6.78×10^{-2}

(3) Isotopes with an activity less than 1.00-02 Ci are not listed.

(Historical Information)

TABLE 12.2-13

DRYWELL FLOOR DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	7.40-04
Br-84	1.24-03
Br-85	2.66-04
I-131	6.62-04
I-132	6.00-03
I-133	4.57-03
I-134	1.12-02
I-135	6.55-03
Sr-89	1.63-04
Sr-91	3.54-03
Sr-92	5.45-03
Mo-99	1.12-03
Tc-99m	1.41-02
Tc-101	5.21-03
Te-132	2.55-03
Cs-138	8.41-03
Ba-139	7.71-03
Ba-140	4.59-04
Ba-141	6.76-03
Ba-142	5.84-03
Np-239	1.22-02
Mn-56	4.95-04
La-142	3.24-04
La-141	1.46-04
Y-92	1.19-04
Y-91m	1.83-04
Total	1.06-01

(Historical Information)

TABLE 12.2-13 (Cont)

---

(1) Design basis:

a. Sump volume = 785 gallons

b. Input rate of 20 gpm for 14 minutes

c. Input activity = 0.001 x primary coolant activity, Tables 12.2-8 through 12.2-11.

(2) 7.40-04 = 7.40×10^{-4}

(3) Isotopes with an activity less than 1.00-04 Ci are not listed.

(Historical Information)

TABLE 12.2-14

RWCU SYSTEM RECIRCULATION PUMPS SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)[2][3]
N-13	2.74-08
N-16	1.55-07
O-19	1.39-07
F-18	2.90-09
Br-83	5.45-08
Br-84	1.00-07
Br-85	5.04-08
I-131	4.74-08
I-132	4.37-07
I-133	3.28-07
I-134	8.66-07
I-135	4.74-07
Sr-89	1.17-08
Sr-91	2.55-07
Sr-92	4.00-07
Mo-99	8.03-08
Tc-99m	1.02-06
Tc-101	4.89-07
Te-132	1.82-07
Cs-138	6.80-07
Ba-139	5.80-07
Ba-140	3.28-08
Ba-141	5.99-07
Ba-142	5.86-07
Np-239	8.76-07
Na-24	1.46-09
Cr-51	3.65-10
Mn-56	3.64-08
Co-58	3.65-09

(Historical Information)

TABLE 12.2-14 (Cont)

Isotope	Specific Activity (Ci/cc)(2)(3)
W-187	2.19-09
La-142	4.05-09
La-141	1.64-09
Y-92	1.18-09
Y-91m	1.86-09
Xe-135m	5.57-09
Total	8.54-06

---

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11. Pumps are assumed to be 54 seconds downline from the reactor recirculation loop

(2) 2.74-08 = 2.74×10^{-8}

(3) Isotopes with specific activities less than 1.00-10 Ci/cc are not listed

(Historical Information)

TABLE 12.2-15

RWCU SYSTEM REGENERATIVE HEAT EXCHANGER -

FIRST STAGE SHIELDING DESIGN SOURCE TERMS[1]

	Activity (Ci)[2][3]	
Isotope	Tube Side	Shell Side
N-13	4.15-03	4.12-03
N-16	3.35-03	-
O-19	1.33-02	-
Br-83	8.44-03	1.07-03
Br-84	1.54-02	1.84-03
Br-85	7.22-03	-
I-131	7.36-03	-
I-132	6.77-02	8.67-03
I-133	5.09-02	6.59-03
I-134	1.34-01	1.65-02
I-135	7.34-02	9.46-03
Sr-89	1.81-03	-
Sr-91	3.96-02	5.11-03
Sr-92	6.19-02	7.90-03
Mo-99	1.24-02	1.61-03
Tc-99m	1.58-01	2.04-02
Tc-101	7.45-02	7.99-03
Ce-132	2.83-02	3.67-03
Cs-138	1.05-01	5.00-02
Ba-139	8.96-02	1.13-02
Ba-140	5.09-03	-
Ba-141	9.17-02	1.03-02
Ba-142	8.90-02	9.06-03
Np-239	1.36-01	1.76-02
Mn-56	5.63-03	-
Xe-135	-	4.97-02

(Historical Information)

TABLE 12.2-15 (Cont)

Isotope	Activity (Ci)(2)(3) Tube Side	Shell Side
Xe-135m	1.18-03	7.15-01
Xe-133	-	1.04-02
Total	1.29+00	9.89-01

---

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. Tube side is reactor water with 71 seconds decay accumulated for 6.9 seconds at 342 gpm at 0.76 g/cc; shell side is filtered and demineralized reactor water with 300 seconds decay accumulated for 8.7 seconds at 310 gpm at 0.86 g/cc

b. Shell side assumes a decontamination factor (DF) of 10 across the filter-demineralizer (except for Cs and Rb, which have a DF of 2, and gases, which have a DF of 1)

c. Shell side includes noble gases generated by particulate parents accumulating on filter demineralizer resins

(2) $4.15\text{-}03 = 4.15 \times 10^{-3}$

(3) Isotopes with activities less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-16

RWCU SYSTEM REGENERATIVE HEAT EXCHANGER -

SECOND STAGE SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3] Tube Side	Shell Side
N-13	4.41-03	4.40-03
N-16	1.78-03	-
O-19	1.20-02	-
Br-83	9.04-03	1.13-03
Br-84	1.65-02	1.95-03
Br-85	7.53-03	-
I-131	7.88-03	1.01-03
I-132	7.25-02	9.15-03
I-133	5.45-02	6.95-03
I-134	1.43-01	1.74-02
I-135	7.86-02	9.98-03
Mn-56	6.03-03	-
Sr-89	1.94-03	-
Sr-91	4.24-02	5.39-03
Sr-92	6.63-02	8.34-03
Mo-99	1.33-02	1.70-03
Tc-99m	1.69-01	2.15-02
Tc-101	7.94-02	8.49-03
Te-132	3.03-02	3.87-03
Cs-138	1.12-01	5.29-02
Ba-139	9.59-02	1.19-02
Ba-140	5.46-03	-
Ba-141	9.78-02	1.09-02
Ba-142	9.46-02	9.65-03
Np-239	1.45-01	1.86-02
W-187	3.64-04	-
La-142	1.00-03	-
Xe-135	-	5.22-02

(Historical Information)

TABLE 12.2-16 (Cont)

Isotope	Activity (Ci)[(2)(3)] Tube Side	Shell Side
Xe-135m	1.39-03	7.59-01
Xe-133	-	1.10-02
Kr-83m	-	1.49-02
Total	1.37+00	1.05+00

---

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. The tube side is reactor water with 78 seconds decay accumulated for 7.4 seconds at 317 gpm at 0.82 g/cc; the shell side is filtered and demineralized reactor water with 291 seconds decay accumulated for 9.1 seconds at 294 gpm at 0.92 g/cc

b. The shell side assumes a DF of 10 across the filter demineralizer (except for Cs and Rb, which have a DF of 2, and gases, which have a DF of 1)

c. The shell side includes noble gases generated by particulate parents accumulating on filter-demineralizer resins

(2) $4.41\text{-}03 = 4.41 \times 10^{-3}$

(3) Isotopes with activities less than 1.00-04 Ci are not listed

(Historical Information)

TABLE 12.2-17

RWCU SYSTEM REGENERATIVE HEAT EXCHANGER -

THIRD STAGE SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3] Tube Side	Shell Side
N-13	4.73-03	4.73-03
O-19	-	1.09-02
Br-83	1.21-03	9.77-03
Br-84	2.10-03	1.78-02
Br-85	-	7.91-03
I-131	1.07-03	8.52-03
I-132	9.74-03	7.84-02
I-133	7.39-03	5.90-02
I-134	1.85-02	1.54-01
I-135	1.06-02	8.50-02
Mn-56	-	6.51-03
Sr-89	-	2.10-03
Sr-91	5.73-03	4.58-02
Sr-92	8.87-03	7.17-02
Mo-99	1.81-03	1.44-02
Tc-99m	2.29-02	1.83-01
Tc-101	9.10-03	8.53-02
Te-132	4.11-03	3.28-02
Cs-138	5.65-02	1.21-01
Ba-139	1.27-02	1.04-01
Ba-140	-	5.90-03
Ba-141	1.16-02	1.05-01
Ba-142	1.04-02	1.02-01
Np-239	1.97-02	1.57-01
La-142	-	1.18-03
Xe-135	5.53-02	-
Xe-135m	8.12-01	1.63-03

(Historical Information)

TABLE 12.2-17 (Cont)

Isotope	Activity (Ci)(2)(3) Tube Side	Shell Side
Xe-133	1.17-02	-
Kr-83m	1.59-02	-
Total	1.12+00	1.48+00

---

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. The tube side is reactor water with 85 seconds decay accumulated for 8 seconds at 292 gpm at 0.89 g/cc; the shell side is filtered and demineralized reactor water with 281 seconds decay accumulated for 9.7 seconds at 278 gpm at 0.97 g/cc

b. The shell side assumes a DF of 10 across the filter demineralizer (except for Cs and Rb, which have a DF of 2, and gases, which have a DF of 1)

c. The shell side includes noble gases generated by particulate parents accumulating on filter demineralizer resins

(2) $4.73\text{-}03 = 4.73 \times 10^{-3}$

(3) Isotopes with activities less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-18

RWCU SYSTEM NONREGENERATIVE HEAT EXCHANGER -

FIRST STAGE TUBE SIDE SHIELDING DESIGN SOURCE TERMS(1)

Isotope(2)	Activity (Ci)(3)(4)
N-13	8.14-04
O-19	1.47-03
Br-83	1.70-03
Br-84	3.08-03
Br-85	1.32-03
I-131	1.48-03
I-132	1.36-02
I-133	1.03-02
I-134	2.68-02
I-135	1.48-02
Sr-89	3.65-04
Sr-91	7.98-03
Sr-92	1.25-02
Mo-99	2.51-03
Tc-99m	3.19-02
Tc-101	1.47-02
Te-132	5.71-03
Cs-138	2.09-02
Ba-139	1.80-02
Ba-140	1.03-03
Ba-141	1.82-02
Ba-142	1.75-02
Np-239	2.74-02
Mn-56	1.13-03
Co-58	1.14-04
La-142	2.28-04

(Historical Information)

TABLE 12.2-18 (Cont)

Isotope(2)	Activity (Ci)(3)(4)
Y-91m	1.07-04
Xe-135m	3.16-04
Total	2.57-01

---

(1) Based on shielding design source terms given in Tables 12.2-9 through 12.2-11; reactor water with 99 seconds decay accumulated for 13.6 seconds at 278 gpm at 0.96 g/cc

(2) Tube side only; shell side is demineralized feedwater

(3) 8.14-04 = 8.14 x 10^{-4}

(4) Isotopes with activities less than 1.00-04 Ci are not listed

(Historical Information)

TABLE 12.2-19

RWCU SYSTEM NONREGENERATIVE HEAT EXCHANGER -

SECOND STAGE TUBE SIDE SHIELDING DESIGN SOURCE TERMS[1]

Isotope[2]	Activity (Ci)[3][4]
N-13	8.19-04
O-19	1.10-03
Br-83	1.73-03
Br-84	3.14-03
Br-85	1.29-03
I-131	1.52-03
I-132	1.39-02
I-133	1.05-02
I-134	2.73-02
I-135	1.51-02
Sr-89	3.73-04
Sr-91	8.15-03
Sr-92	1.27-02
Mo-99	2.57-03
Tc-99m	3.26-02
Tc-101	1.49-02
Te-132	5.83-03
Cs-138	2.13-02
Ba-139	1.84-02
Ba-140	1.05-03
Ba-141	1.84-02
Ba-142	1.76-02
Np-239	2.80-02
Mn-56	1.16-03
Co-58	1.17-04
La-142	2.61-04
La-141	1.07-04
Nb-97m	2.53-06

(Historical Information)

TABLE 12.2-19 (Cont)

Isotope(2)	Activity (Ci)(3)(4)
Y-91m	1.23-04
Xe-135m	3.64-04
Total	2.61-01

---

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11; reactor water with 112 seconds decay accumulated for 13.8 seconds at 273 gpm at 0.98 g/cc

(2) Tube side only; shell side is demineralized feedwater

(3) 8.19-04 = 8.19×10^{-4}

(4) Isotopes with activities less than 1.00-06 Ci are not listed

(Historical Information)

TABLE 12.2-20

RWCU SYSTEM FILTER DEMINERALIZER SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	7.97+00
Br-84	3.23+00
I-131	2.46+02
I-132	7.13+02
I-133	4.02+02
I-134	4.56+01
I-135	1.92+02
Sr-89	7.66+01
Sr-90	6.00+00
Sr-91	1.49+02
Sr-92	6.58+01
Nb-95	1.05+00
Mo-99	2.66+02
Tc-99m	5.99+02
Tc-101	6.96+00
Te-132	6.59+02
Cs-134	3.99+00
Cs-136	2.33+00
Cs-137	6.00+00
Cs-138	2.22+01
Ba-139	4.86+01
Ba-140	1.88+02
Ba-141	1.10+01
Ba-142	6.56+00
Ce-141	2.40+00
Np-239	2.59+03
Na-24	1.32+00
Cr-51	2.30+00
Mn-56	5.64+00

(**Historical Information**)

TABLE 12.2-20 (Cont)

Isotope	Activity (Ci)[(2)(3)]
Co-58	2.42+01
Co-60	2.50+00
W-187	3.14+00
La-142	6.88+00
La-141	1.13+01
La-140	1.30+02
Y-92	6.59+01
Y-91	1.05+01
Y-91m	8.83+01
Y-90	3.30+00
Total	6.68+03

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. Filter demineralizer is assumed to be 147 seconds downline from the reactor recirculation loop

b. Filter demineralizer resins are assumed to accumulate all nongaseous isotopes in reactor water for 6.8 days at 135 gpm at 0.99 g/cc

c. Backwash volume for the receiving tank is assumed to be 1360 gallons

(2) $7.97+00 = 7.97 \times 10^{0}$

(3) Isotopes with activities less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-21

RWCU SYSTEM FILTER DEMINERALIZER HOLDING PUMPS

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
N-13	2.79-08
F-18	3.88-09
Br-83	7.34-09
Br-84	1.26-08
Br-85	2.57-09
I-131	6.52-09
I-132	5.93-08
I-133	4.50-08
I-134	1.12-07
I-135	6.46-08
Sr-89	1.61-09
Sr-91	3.49-08
Sr-92	5.40-08
Mo-99	1.10-08
Tc-99m	1.40-07
Tc-101	5.42-08
Te-132	2.51-08
Cs-138	4.26-07
Ba-139	7.68-08
Ba-140	4.51-09
Ba-141	6.98-08
Ba-142	6.48-08
Np-239	1.20-07
Na-24	2.00-10
Mn-56	4.90-09
Co-58	5.02-10
La-142	2.78-09
La-141	1.19-09

(Historical Information)

TABLE 12.2-21 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)
Y-91m	1.18-09
Xe-135	3.53-07
Xe-135m	3.96-06
Xe-133m	4.03-09
Xe-133	7.03-08
Kr-83m	9.33-08
Total	5.93-06

---

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. Pumps are assumed to be 258 seconds downline from the reactor recirculation loop

b. A DF of 10 is assumed across the filter demineralizer (except for Cs and Rb, which have a DF of 2, and gases that have a DF of 1)

c. Includes noble gases generated by particulate parents accumulating on filter demineralizer resins

(2) 2.79-08 = 2.79×10^{-8}

(3) Density = 1 g/cc

(Historical Information)

TABLE 12.2-22

RWCU BACKWASH RECEIVER TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
I-131	4.59+02
I-132	1.19+03
I-133	5.02+02
I-134	4.49+01
I-135	1.95+02
Sr-89	1.52+02
Sr-90	1.20+01
Sr-91	1.57+02
Sr-92	6.53+01
Mo-99	4.39+02
Tc-99m	7.66+02
Te-132	1.12+03
Cs-137	1.20+01
Cs-138	2.16+01
Ba-139	4.81+01
Ba-140	3.59+02
Ba-141	1.05+01
Ba-142	6.05+00
Np-239	4.16+03
Co-58	4.80+01
La-141	1.13+01
La-140	2.85+02
Y-92	6.60+01
Y-91	2.18+01
Y-91m	9.35+01
Total	1.04+04

(Historical Information)

TABLE 12.2-22 (Cont)

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(1) Design basis:

a. Tank volume = 3,000 gallons

b. Two consecutive RWCU filter/demineralizer backwash (one with 1.7 days decay and one with no decay)

c. 2 backwash volume = 2720 gallons.

(2) 4.59+02 = 4.59×10^2

(3) Isotopes with an activity less than 1.00+00 Ci are not listed.

(Historical Information)

TABLE 12.2-23

RWCU BACKWASH TRANSFER PUMP SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	1.05-06
I-131	6.06-05
I-132	1.57-04
I-133	6.63-05
I-134	5.90-06
I-135	2.58-05
Sr-89	2.01-05
Sr-90	1.59-06
Sr-91	2.07-05
Sr-92	8.61-06
Mo-99	5.81-05
Tc-99m	1.01-04
Te-132	1.48-04
Cs-134	1.05-06
Cs-137	1.59-06
Cs-138	2.95-06
Ba-139	6.35-06
Ba-140	4.74-05
Ba-141	1.39-06
Np-239	5.50-04
Co-58	6.34-06
La-141	1.49-06
La-140	3.77-05
Y-92	8.72-06
Y-91	2.88-06
Y-91m	1.24-05
Y-90	1.01-06
Total	1.36-03

(Historical Information)

TABLE 12.2-23 (Cont)

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(1) Design basis:

a. RWCU backwash volume = 1000 gallons per batch

b. No decay in the RWCU backwash receiver tank

c. Density = 1.0 g/cc

(2) 1.05-06 = 1.05×10^{-6}

(3) Isotopes with specific activity less than 1.00-06 Ci/g are not listed

(Historical Information)

TABLE 12.2-24

RESIDUAL HEAT REMOVAL SYSTEM SHIELDING DESIGN SOURCE TERMS(1)(4)

Isotope	Specific Activity (Ci/g)(2)(3)
I-131	4.33-07
I-132	3.55-06
I-133	3.73-07
I-134	2.05-07
I-135	2.27-07
Sr-89	3.25-05
Sr-90	1.14-06
Sr-91	4.03-06
Sr-92	1.83-06
Zr-95	4.63-05
Zr-97	4.73-06
Nb-95	4.87-05
Tc-99m	1.84-07
Te-132	2.28-05
Cs-137	2.39-07
Ba-139	7.42-07
Ba-140	2.71-05
Ce-141	3.01-05
Ce-143	6.09-06
Pr-143	2.60-05
Np-239	2.28-07
La-142	5.74-07
La-141	1.95-06
La-140	2.88-05
Nb-95m	6.12-07
Y-92	3.79-06
Y-91	3.69-05
Y-91m	2.63-06
Xe-135	2.43-07

(Historical Information)

TABLE 12.2-24 (Cont)

RESIDUAL HEAT REMOVAL SYSTEM SHIELDING DESIGN SOURCE TERMS(1)(4)

Isotope	Specific Activity (Ci/g)(2)(3)
Xe-135m	1.16-07
Xe-133	4.08-07
Total	3.34-04

(1) Design basis:

a. Primary coolant activity normalized to 0.1 Ci/s Table 12.2-7 noble gas release at 30 minutes decay and 700 µCi/s iodine release rate, Table 12.2-8.

b. 4-hour decay prior to system operation

c. Fission product spiking factors from Reference 12.2-6.

(2) $4.33\text{-}07 = 4.33 \times 10^{-7}$

(3) Isotopes with specific activities less than 1.0-07 Ci/g are not listed

(4) The values are based on 3323 MW_t. The impact of uprating to 3339 MW_t is insignificant.

(Historical Information)

TABLE 12.2-25

RCIC SYSTEM SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)[2][3]
N-16	9.53-07[4][5]
O-19	2.98-08
Kr-87	2.50-09
Kr-88	2.50-09
Kr-89	1.57-08
Kr-90	2.93-08
Kr-91	2.23-08
Kr-92	5.14-09
Xe-133	1.02-09
Xe-135m	3.23-09
Xe-135	2.74-09
Xe-137	1.85-08
Xe-138	1.11-08
Xe-139	3.04-08
Xe-140	2.53-08
I-134	1.11-09
Cs-141	1.71-09
Cs-140	2.83-09
Rb-93	1.94-09
Rb-92	1.01-08
Rb-91	2.64-09
Rb-90	1.14-09
Total	1.17-06

(Historical Information)

TABLE 12.2-25 (Cont)

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(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. Inlet steam to RCIC turbine, operating at maximum conditions; 34,100 lb/h, 1250 psig, 575°F

b. Credit of 9.51 seconds decay in transit from the RPV steam nozzles

c. Density = 0.0464 g/cc

(2) 8.97-07 = 8.97 x 10^{-7}

(3) Isotopes with specific activities less than 1.0-09 Ci/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-26

HPCI SYSTEM SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)(2)(3)
N-16	1.26-06[4][5]
O-19	3.19-08
Kr-87	2.50-09
Kr-88	2.50-09
Kr-89	1.59-08
Kr-90	3.09-08
Kr-91	2.73-08
Kr-92	9.96-09
Kr-93	1.41-09
Xe-133	1.02-09
Xe-135m	3.23-09
Xe-135	2.74-09
Xe-137	1.87-08
Xe-138	1.11-08
Xe-139	3.19-08
Xe-140	2.86-08
Xe-141	2.31-09
I-134	1.11-09
Cs-142	1.66-09
Cs-141	1.74-09
Cs-140	2.10-09
Rb-93	2.40-09
Rb-92	1.14-08
Rb-91	2.02-09
Total	1.51-06

(Historical Information)

TABLE 12.2-26 (Cont)

Isotope	Specific Activity (Ci/cc)[(2) (3)]

(1) Based on shielding design source terms given in Tables 12.2-7 through 12.2-11:

a. Inlet steam to HPCI turbine, operating at maximum conditions; 235,000 lb/h, 1250 psig, 575°F

b. Credit of 6.3 seconds decay in transit from the RPV steam nozzles.

c. Density = 0.0464 g/cc

(2) 1.26-06 = 1.26×10^{-6}

(3) Isotopes with specific activities less than 1.0-09 Ci/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-27

CORE SPRAY SYSTEM SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/cc)[2][3]
I-131	2.84-05
I-132	1.08-06
I-135	4.26-07
Sr-89	1.31-05
Sr-90	9.89-05
Zr-95	1.15-07
Mo-99	3.32-07
Tc-99m	3.51-07
Te-132	1.01-06
Cs-134	7.32-06
Cs-137	1.08-04
Ba-140	2.79-06
Ce-141	2.26-07
Ce-144	5.90-07
Np-239	2.61-06
Mn-54	1.40-07
Co-58	3.19-06
Co-60	1.20-05
Ag-110m	1.74-07
Pr-144	5.90-07
La-140	3.11-06
Nb-95	1.94-07
Y-91	2.09-06
Y-90	9.91-05
Total	3.86-04

(Historical Information)

TABLE 12.2-27 (Cont)

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(1) Design basis:

a) Suppression pool water activity at 40 years operation

b) HPCI pump turbine exhaust - 18 (30 minutes @) system tests per year

c) RCIC pump turbine exhaust - 18 (30 minutes @) system tests per year

d) Main steam safety valve relieves - Once per year for 12 minutes

(2) 2.84-05 = 2.84×10^{-5}

(3) Isotopes with specific activity less than 1.0-07 μCi/cc are not listed

TABLE 12.2-28

SPENT FUEL ASSEMBLY SHIELDING DESIGN SOURCE TERMS[1]

Gamma Energy (Upper MeV)	Gamma Activity At 3 Days After Shutdown (Mev/s)[2]	Gamma Activity at 3 Months After Shutdown (Mev/s)
0.10	3.70+14	1.06+13
0.20	2.00+15	2.81+14
0.30	1.15+15	9.36+12
0.40	2.01+15	8.54+12
0.50	5.48+15	8.02+14
0.60	2.95+15	6.62+14
0.80	2.38+16	7.41+15
1.00	4.42+15	2.06+14
1.25	1.69+15	2.02+14
1.50	1.55+15	1.59+14
1.75	1.16+16	1.25+14
2.00	3.75+14	1.32+13
2.50	9.34+14	1.02+14
3.00	6.98+14	8.07+12
4.00	7.45+12	2.65+11
5.00	5.59+05	0
Total	5.91+16	1.00+16

(1) Design basis:
   a) Reactor power at 3440 MWt
   b) Maximum exposed fuel assembly with 54 months burn
   c) Core peaking factor of 1.4

(2) 3.70+14 = 3.70×10^{14}

TABLE 12.2-29

FUEL POOL COOLING AND CLEANUP SYSTEM
SHIELDING DESIGN SOURCE TERMS

Isotope	Specific Activity (Ci/gm)(2)(3)
I--131	5.87-12
I--133	1.63-13
Sr--89	6.24-10
Sr--90	2.34-11
Zr--95	9.03-10
Zr--97	8.38-13
Nb--95	9.09-10
Te-129m	1.43-12
Te-132	1.68-10
Cs-136	1.41-11
Cs-137	1.78-10
Ba-140	4.27-10
Ce-141	5.57-10
Ce-143	8.52-12
Pr-143	4.17-10
Np-239	5.03-13
Co--58	4.38-13
La-140	8.03-11
Nb-95m	4.96-12
Y---91	7.14-10
Total	5.04-09

(1) Design basis:
- a) Total refueling water volume - 1,106,000 gallons
- b) Refueling water surface dose rate of 2.5 mrem/h (approximately 120 h after shutdown)

(Historical Information)

TABLE 12.2-29 (Cont)

c) Input activity is the fission spike source at 4 h after shutdown (Table 12.2-24)

(2) 5.87-12 = 5.87×10^{-12}

(3) Isotopes with specific activity less than 1.00-13 Ci/g are not listed

(Historical Information)

TABLE 12.2-30

FUEL POOL FILTER DEMINERALIZER SHIELDING

DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
I--131	7.24-02
I--132	2.20+00
I--133	5.80-03
Sr--89	6.92+00
Sr--90	2.55-01
Sr--91	2.82-03
Zr--95	9.97+00
Zr--97	4.01-02
Te-129m	1.60-02
Te-129	9.63-03
Te-132	2.49+00
Cs-136	3.85-02
Cs-137	4.48-01
Ba-140	5.03+00
Ce-141	6.24+00
Ce-143	1.95-01
Np-239	8.43-03
Pr-143	4.88+00
Co--58	4.82-03
Nb-95m	9.04-02
Nb-95	1.03+01
Y-91	7.90+00
Y-91m	1.63-03
Y-90	3.49-02
Total	5.97+01

(Historical Information)

TABLE 12.2-30 (Cont)

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(1) Design basis:

a) Assume all activity in refueling water accumulated in one fuel pool filter demineralizer

b) Refueling water specific activity of 1.4-02 μCi/cc after refueling flooding

c) Total refueling water volume - 1.106E6 gallons

(2) 7.24-02 = 7.24×10^{-2}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-31

FUEL POOL HEAT EXCHANGERS SHIELDING

DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2]
Na-24	1.69-02
P-32	1.69-04
Cr-51	4.22-03
Mn-54	3.38-04
Mn-56	4.22-01
Co-58	4.22-02
Co-60	4.22-03
Fe-59	6.75-04
Ni-65	2.54-03
Zn-65	1.69-05
Zn-69m	2.54-04
Ag-110m	5.06-04
W-187	2.54-02
Total	5.19-01

(1) Design basis:

a. Based on operating experiences

b. Corrosion products (Table 12.2-11) were normalized to yield a contact dose rate of 175 mrem/h for a fuel pool heat exchanger

(2) $1.69\text{-}02 = 1.69 \times 10^{-2}$

(Historical Information)

TABLE 12.2-32

TRAVERSING IN-CORE GAMMA PROBE SHIELDING DESIGN SOURCE TERMS[1]

Energy (Mev)	Gamma Activity (Mev/s)[2]
0.4	3.39+08
0.8	5.01+09
1.3	5.54+07
2.0	1.06+07

---

(1) Design basis:

a) Source volume - 0.211 in. od x 2.549 in. long

b) Irradiated for 2 h at 30-day intervals at 1.0+14 nv for 3 years

c) Normalized to yield 0.104 rem/h at 1 meter with one day decay

(2) 3.39+08 = 3.39×10^8

(Historical Information)

TABLE 12.2-33

SPECIAL MATERIAL SOURCES

Use or Type	Isotopes	Source Strength
For power range detectors	U234 - 0.83 mg[1] U235 - 0.22 mg U238 - 0.02 mg	5.1 μCi - Total
For source range detectors	U235 - 2.72 mg[1] U238 - 0.20 mg	0.0059 μCi - Total
For intermediate range detectors	U235 - 0.75 mg[1] U238 - 0.05 mg	0.0016 μCi - Total
Process monitoring	Cs137	9 μCi
	Ba133	9 μCi
	Sr90	0.1 μCi
	Co60	18-22 μCi
	U234/238	6.6×10^{-5} mCi
	C136	9 mCi
	Th232	0.5 mCi
Area monitoring	Sr90	0.1μCi
Startup (neutron sources)	Sb124	16,800-19,600 Ci
Calibration	Cs137	500 mCi 5 mCi 1 mCi

(1) Per detector

(Historical Information)

TABLE 12.2-34

CONTROL ROD DRIVE MECHANISM SHIELDING

DESIGN SOURCE TERMS(1)

Isotopes	Activity (Ci)(2) CRD Filter	CRD Body
Na-24	2.32-02	1.31-02
P-32	2.32-04	1.31-04
Cr-51	5.79-03	3.27-03
Mn-54	4.63-04	2.61-04
Mn-56	5.79-01	3.27-01
Co-58	5.79-02	3.27-02
Co-60	5.79-03	3.27-03
Fe-59	9.26-04	5.22-04
Ni-65	3.47-03	1.96-03
Zn-65	2.32-05	1.31-05
Zn-69m	3.47-04	1.96-04
Ag-110m	6.95-04	3.93-04
W-187	3.47-02	1.96-02
Total	7.12-01	4.02-01

---

(1) Design basis:

a. Based on operating experiences

b. CRD filter source terms are corrosion products (Table 12.2-11) normalized to yield a contact dose rate of 50 rem/h

c. CRD body source terms are corrosion products (Table 12.2-11) normalized to yield a contact dose rate of 150 mrem/h

(2) 2.32-02 = 2.32×10^{-2}

(Historical Information)

TABLE 12.2-35

REACTOR BUILDING VENT SYSTEM FILTER SHIELDING

DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
I-131	2.52-03
I-132	2.72-04
I-133	1.83-03
I-134	2.09-04
I-135	8.71-04
Sr-89	3.73-05
Sr-09	1.49-05
Zr-95	4.53-04
Mo-99	4.01-04
Tc-99	3.49-04
Cs-134	3.06-03
Cs-136	3.30-05
Cs-137	4.96-03
Ce-141	1.55-04
Cr-51	1.34-04
Mn-54	1.45-03
Co-58	1.62-04
Co-60	4.74-03
Fe-59	9.77-05
Zn-65	3.38-03
Rh-103m	1.85-04
La-140	6.16-04
Nb-95	1.27-03
Ru-103	1.89-04
Sb-124	2.81-05

(Historical Information)

TABLE 12.2-35 (Cont)

Isotope	Activity (Ci)(2)(3)
Ba-140	6.16-04
Y-90	1.47-05
Total	3.11-02

---

(1) Design basis:
   a. Based on release rates given by NUREG-0016, Ref. 12.2-1
   b. Iodine concentrations are normalized to correspond to a 500,000 μCi/s noble gas release at 30 minutes decay
   c. Particulate release rates are increased by a factor of 5
   d. 100% efficiency for particulates and particulate iodines and accumulated for one year

(2) $2.52\text{-}03 = 2.52 \times 10^{-3}$

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-36

PRIMARY CONTAINMENT PREPURGE FILTER SHIELDING

DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
I-131	8.19-03
I-132	8.88-04
I-133	5.97-03
I-134	6.80-04
I-135	2.84-03
Sr-89	9.20-05
Sr-90	1.02-05
Zr-95	2.49-04
Mo-99	1.15-03
Tc-99m	9.96-04
Ru-103	1.49-04
Cs-134	5.10-03
Cs-136	4.30-05
Cs-137	7.53-03
Cs-138	5.80-04
Ba-140	1.14-03
Ba-141	1.66-04
Ru-130m	1.46-04
Cr-51	1.74-04
Mn-54	1.32-03
Co-58	1.22-04
Co-60	6.49-03
Fe-59	1.04-04
Zn-65	3.71-03
Rb-88	2.88-03
Rb-89	3.47-05
La-140	1.14-03

(Historical Information)

TABLE 12.2-36 (Cont)

Isotope	Activity (Ci)(2)(3)
Nb-95	2.77-03
Y-90	1.00-05
Total	5.47-02

---

(1) Design basis:

a. Based on release rates given by NUREG-0016, Ref. 12.2-1

b. Iodine concentrations are normalized to correspond to a 500,000 μCi/s noble gases release at 30 minutes decay

c. Particulates release rates are increased by a factor of 5

d. 100 percent efficiency for all isotopes and accumulated for one year

(2) $8.19\text{-}03 = 8.19 \times 10^{-3}$

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-37

RPV STEAM SEPARATOR AND DRYER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
P-32	6.62-01
Cr-51	3.37+01
Mn-54	5.31+00
Co-58	5.31+02
Co-60	7.03+01
Fe-59	7.20+00
Zn-65	2.61-01
Ag-110m	7.85+00
Ag-110	1.02-01
Total	6.56+02

(1) Design basis:

a) Based on operating experiences

b) Corrosion products (Table 12.2-11) were normalized to yield a contact dose rate of 10 rem/h on top of steam dryer

(2) $6.62\text{-}01 = 6.62 \times 10^{-1}$

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-38

REACTOR BUILDING EQUIPMENT DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	3.25-02
Br-84	1.89-02
Br-85	1.09-03
I-131	4.84-02
I-132	3.33-01
I-133	3.14-01
I-134	2.57-01
I-135	3.95-01
Sr-89	1.20-02
Sr-91	2.26-01
Sr-92	2.52-01
Mo-99	8.07-02
Tc-99m	8.47-01
Tc-101	4.18-02
Te-132	1.84-01
Cs-138	1.30-01
Ba-139	2.49-01
Ba-140	3.36-02
Ba-141	6.52-02
Ba-142	3.98-02
Np-239	8.77-01
Na-24	1.36-03
Mn-56	2.24-02
Co-58	3.75-03
W-187	2.12-03
La-142	3.31-02
La-141	3.21-02
La-140	1.20-03

(Historical Information)

TABLE 12.2-38 (Cont)

Isotope	Activity (Ci)[2][3]
Y-92	9.31-02
Y-91m	9.93-02
Total	4.73+00

---

(1) Design basis:

a. Sump volume = 660 gallons

b. Input rate of 2.6 gpm for 254 minutes

c. Input activity = 0.30 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

(2) 3.25-02 = 3.25×10^{-2}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-39

REACTOR BUILDING FLOOR DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	2.82-04
Br-84	2.30-04
Br-85	1.39-05
I-131	3.31-04
I-132	2.57-03
I-133	2.21-03
I-134	2.85-03
I-135	2.96-03
Sr-89	8.17-05
Sr-91	1.65-03
Sr-92	2.14-03
Mo-99	5.56-04
Tc-99m	6.37-03
Tc-101	5.35-04
Te-132	1.26-03
Cs-138	1.58-03
Ba-139	2.45-03
Ba-140	2.30-04
Ba-141	8.31-04
Ba-142	5.11-04
Np-239	6.05-03
Mn-56	1.92-04
Co-58	2.56-05
W-187	1.49.05
La-142	3.00-04
La-141	2.31-04

(Historical Information)

TABLE 12.2-39 (Cont)

Isotope	Activity (Ci)[2][3]
Y-92	4.46-04
Y-91M	5.43-04
Total	3.75-02

(1) Design basis:

a. Sump volume = 1350 gallons

b. Input rate of 10 gpm for 135 minutes

c. Input activity = 0.001 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

(2) 2.82-04 = 2.82 x 10^{-4}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-40

RPV HEAD WASHDOWN AREA SHIELDING DESIGN SOURCE TERMS[1]

Energy (MeV)	Gamma Activity (Mev/s)[2]
0.1	7.41+8
0.4	2.50+9
0.8	1.35+11
1.3	6.10+11
1.7	9.05+10
2.18	1.98+5
2.5	0.00
2.8	0.00

---

(1) Design basis:

a) Source terms were based on decontamination solution activities (HCGS PSAR, Table 9.2-2) and were then normalized to a contact dose rate of 1.0 rem/h underneath the RPV head

b) Source volume: A disk 11 ft radius by 10 cm deep = 3.532+6 cc

(2) $7.41E8 = 7.41 \times 10^8$

(Historical Information)

TABLE 12.2-41

SPENT FUEL CASK SHIELDING DESIGN SOURCE TERMS[1]

Energy (MeV)	Gamma Activity (Mev/s)[2]
0.8	2.59+14
1.2	6.44+13
1.6	9.46+12
2.0	7.09+12
2.4	1.53+12
2.8	5.74+10
3.5	2.13+10
5.0	1.39+10

(1) Design basis:

a) Source volume = 2.6706+7 cc (152.4 cm radius by 366 cm height)

b) Based on spent fuel decayed 90 days, with the contact dose rate on the lead cask (thickness = 3 in.) normalized to 200 mrem/h.

(2) 2.59+14 = 2.59×10^{14}

(Historical Information)

TABLE 12.2-42

REACTOR BUILDING SHIELDING DESIGN RADIATION SOURCE DESCRIPTION

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-42 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

TABLE 12.2-42 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-43

RADWASTE AREA EQUIPMENT DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	4.50-03
Br-84	1.86-03
I-131	2.00-02
I-132	9.50-02
I-133	1.07-01
I-134	2.60-02
I-135	9.25-02
Sr-89	5.07-03
Sr-91	6.20-02
Sr-92	3.70-02
Mo-99	3.19-02
Tc-99m	2.03-01
Tc-101	4.09-03
Te-132	7.35-02
Cs-138	1.28-02
Ba-139	2.77-02
Ba-140	1.40-02
Ba-141	6.38-03
Ba-142	3.90-03
Np-239	3.42-01
Mn-56	3.20-03
Co-58	1.59-03
La-142	3.90-03
La-141	6.13-03
La-140	2.02-03

TABLE 12.2-43 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-92	3.41-02
Y-91m	3.53-02
Total	1.26+00

---

(1) Design basis:

a. Sump volume = 841 gallons

b. Input rate of 0.764 gpm for 1101 minutes

c. Input activity = 0.10 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

(2) 4.50-03 = 4.50×10^{-3}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-44

RADWASTE AREA HIGH CONDUCTIVITY DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	3.71-04
Br-84	1.52-04
I-131	3.83-03
I-132	1.51-02
I-133	1.47-02
I-134	2.13-03
I-135	8.82-03
Sr-89	1.01-03
Sr-91	6.66-03
Sr-92	3.06-03
Mo-99	5.61-03
Tc-99m	2.13-02
Tc-101	3.35-04
Te-132	1.31-02
Cs-138	1.05-03
Ba-139	2.27-03
Ba-140	2.73-03
Ba-141	5.23-04
Ba-142	3.20-04
Np-239	5.88-02
Mn-56	2.64-04
Co-58	3.16-04
W-187	1.07-04
La-142	3.20-04
La-141	5.23-04
La-140	8.41-04
Y-92	3.06-03

(Historical Information)

TABLE 12.2-44 (Cont)

Isotope	Activity (Ci)[(2)(3)]
Y-91	1.04-04
Y-91m	3.92-03
Total	1.72-01

---

(1) Design basis:

a. Sump volume = 842 gallons

b. Input rate of 0.313 gpm for 2689 minutes

c. Input activity = 0.02 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

(2) 3.71-04 = 3.71 x 10^{-4}

(3) Isotopes with an activity less than 1.00-04 Ci are not listed

(Historical Information)

TABLE 12.2-45

RADWASTE AREA FLOOR DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	4.10-05
Br-84	1.70-05
I-131	2.00-04
I-132	9.26-04
I-133	1.04-03
I-134	2.37-04
I-135	8.67-04
Sr-89	5.06-05
Sr-91	5.89-04
Sr-92	3.38-04
Mo-99	3.16-04
Tc-99m	1.91-03
Tc-101	3.72-05
Te-132	7.28-04
Cs-138	1.16-04
Ba-139	2.52-04
Ba-140	1.40-04
Ba-141	5.81-05
Ba-142	3.55-05
Np-239	3.38-03
Mn-56	2.92-05
Co-58	1.59-05
La-142	3.55-05
La-141	5.64-05
La-140	2.19-05

(Historical Information)

TABLE 12.2-45 (Cont)

Isotope	Activity (Ci)[(2)(3)]
Y-92	3.18-04
Y-91m	3.37-04
Total	1.20-02

---

(1) Design basis:

a. Sump volume = 841 gallons

b. Input rate of 0.695 gpm for 1210 minutes

c. Input activity = 0.001 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

(2) 4.10-05 = 4.10×10^{-5}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-46

FLOOR DRAIN COLLECTOR TANK

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	1.17-02
Br-84	4.86-03
I-131	4.08-02
I-132	2.06-01
I-133	2.30-01
I-134	6.82-02
I-135	2.19-01
Sr-89	1.03-02
Sr-91	1.41-01
Sr-92	9.51-02
Mo-99	6.58-02
Tc-99m	4.75-01
Tc-101	1.07-02
Te-132	1.51-01
Cs-138	3.34-02
Ba-139	7.24-02
Ba-140	2.85-02
Ba-141	1.67-02
Ba-142	1.02-02
Np-239	7.08-01
Mn-56	8.26-03
Co-58	3.21-03
W-187	1.58-03
La-142	1.02-02
La-141	1.53-02
La-140	3.23-03

(Historical Information)

TABLE 12.2-46 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-92	8.02-02
Y-91m	7.89-02
Total	2.81+00

---

(1) Design basis:

a. Tank volume = 17,000 gallons

b. Input rate of 20 gpm for 425 minutes

c. Input activity = 0.01 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11.

(2) 1.17-02 = 1.17 x 10^{-2}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-47

FLOOR DRAIN COLLECTOR AND FILTER HOLDING PUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (Ci)(2)(3)
Br-83	3.20-10
Br-84	1.51-10
I-131	6.42-10
I-132	3.89-09
I-133	4.00-09
I-134	2.11-09
I-135	4.61-09
Sr-89	1.60-10
Sr-90	1.20-11
Sr-91	2.75-09
Sr-92	2.54-09
Mo-99	1.06-09
Tc-99m	9.89-09
Tc-101	3.23-10
Te-132	2.42-09
Cs-137	1.20-11
Cs-138	1.04-09
Ba-139	2.19-09
Ba-140	4.47-10
Ba-141	5.20-10
Ba-142	3.18-10
Np-239	1.15-08
Na-24	1.71-11
Mn-56	2.23-10
Co-58	4.99-11
W-187	2.71-11
La-142	3.03-10
La-141	3.61-10

(Historical Information)

TABLE 12.2-47 (Cont)

Isotope	Specific Activity (Ci)[(2)(3)]
La-140	2.62-11
Y-92	1.42-09
Y-91m	1.39-09
Total	5.47-08

---

(1) Design basis:

a. Density = 1 g/cc

b. Output of floor drain collector tank, Table 12.2-46

(2) 3.20-10 = 3.20×10^{-10}

(3) Isotopes with activity less than 1.00-11 Ci/g are not listed

(Historical Information)

TABLE 12.2-48

FLOOR DRAIN FILTER

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	6.59-04
Br-84	3.74-04
Br-85	2.15-05
I-131	1.00-03
I-132	6.84-03
I-133	6.50-03
I-134	5.11-03
I-135	8.11-03
Sr-89	2.49-04
Sr-90	1.87-05
Sr-91	4.67-03
Sr-92	5.12-03
Mo-99	1.67-03
Tc-99m	1.74-02
Tc-101	8.26-04
Te-132	3.82-03
Cs-134	1.25-05
Cs-137	1.87-05
Cs-138	2.57-03
Ba-139	4.99-03
Ba-140	6.97-04
Ba-141	1.29-03
Ba-142	7.88-04
Np-239	1.82-02
Na-24	2.81-05
Mn-56	4.54-04
Co-58	7.78-05
W-187	4.38-05

(Historical Information)

TABLE 12.2-48 (Cont)

Isotope	Activity (Ci)[2][3]
La-142	6.67-04
La-141	6.59-04
La-140	2.60-05
Y-92	1.97-03
Y-91m	2.08-03
Total	9.46-02

---

(1) Design basis:

a. Average input activity to filter is the normalized primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11, of 0.003 μCi/cc

b. Total process flow = 46,800 gallons per batch at 176 gpm

c. 100% accumulation efficiency for all isotopes

(2) 6.59-04 = 6.59×10^{-4}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-49

FLOOR DRAIN DEMINERALIZER

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	3.81-05
Br-84	1.56-05
I-131	2.34-03
I-132	4.36-03
I-133	1.93-03
I-134	2.19-04
I-135	9.15-04
Sr-89	1.20-03
Sr-90	1.05-04
Sr-91	7.13-04
Sr-92	3.14-04
Zr-95	1.54-05
Nb-95	1.79-05
Mo-99	1.54-03
Tc-99m	3.13-03
Tc-101	3.44-05
Te-129m	1.37-05
Te-132	4.07-03
Cs-134	6.93-05
Cs-136	2.74-05
Cs-137	1.05-04
Cs-138	1.07-04
Ba-139	2.33-04
Ba-140	2.16-03
Ba-141	5.37-05
Ba-142	3.28-05
Ce-141	3.56-05
Ce-144	1.53-05

(Historical Information)

TABLE 12.2-49 (Cont)

Isotope	Activity (Ci)(2)(3)
Pr-143	1.03-05
Np-239	1.42-02
Cr-51	3.26-05
Mn-56	2.72-05
Co-58	3.89-04
Co-60	4.36-05
W-187	1.51-05
Pr-144	1.53-05
La-142	3.28-05
La-141	5.37-05
La-140	2.01-03
Y-92	3.14-04
Y-91	1.78-04
Y-91m	4.21-04
Y-90	8.99-05
Total	4.16-02

---

(1) Design basis:

a. Average input activity to the demineralizer is the normalized primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11, of 0.003 μCi/cc

b. Input rate of 7.333 gpm for 25 days

c. 100 percent accumulation efficiency for all isotopes

(2) 3.81-05 = 3.81×10^{-5}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-50

FLOOR DRAIN SAMPLING TANK

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	1.17-04
Br-84	4.86-05
I-131	4.10-04
I-132	2.06-03
I-133	2.30-03
I-134	6.82-04
I-135	2.19-03
Sr-89	1.03-04
Sr-91	1.41-03
Sr-92	9.51-04
Mo-99	6.58-04
Tc-99m	4.75-03
Tc-101	1.08-04
Te-132	1.51-03
Cs-134	2.57-04
Cs-136	1.74-04
Cs-137	3.59-04
Cs-138	1.67-02
Ba-139	7.24-04
Ba-140	2.85-04
Ba-141	1.67-04
Ba-142	1.02-04
Np-239	7.08-03
Mn-56	8.25-05
Co-58	3.21-05
W-187	1.58-05
La-142	1.02-04
La-141	1.53-04

(Historical Information)

TABLE 12.2-50 (Cont)

Isotope	Activity (Ci)(2)(3)
La-140	3.23-05
Y-92	8.02-04
Y-91m	7.89-04
Total	4.55-02

---

(1) Design basis:
  a. Tank volume = 17,000 gallons
  b. Input rate of 20 gpm for 850 minutes
  c. Input activity = 1 percent primary coolant activity,
  d. Filter - DF = 1 for all isotopes, demineralizer - DF = 100 for anions and DF = 2 for Cs and Rb

(2) 1.17-04 = 1.17×10^{-4}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed.

(**Historical Information**)

TABLE 12.2-51

FLOOR DRAIN SAMPLING PUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci)[2][3]
Br-83	1.81-12
Br-84	7.54-13
I-131	6.32-12
I-132	3.19-11
I-133	3.56-11
I-134	1.06-11
I-135	3.40-11
Sr-89	1.59-12
Sr-90	1.20-13
Sr-91	2.19-11
Sr-92	1.47-11
Mo-99	1.02-11
Tc-99m	7.36-11
Tc-101	1.66-12
Te-132	2.34-11
Cs-134	3.99-12
Cs-136	2.70-12
Cs-137	5.98-12
Cs-138	2.59-10
Ba-139	1.12-11
Ba-140	4.12-12
Ba-141	2.59-12
Ba-142	1.58-12
Np-239	1.10-10
Na-24	1.46-13
Mn-56	1.28-12
Co-58	4.97-13
W-187	2.45-13

(Historical Information)

TABLE 12.2-51 (Cont)

Isotope	Specific Activity (Ci)[2][3]
La-142	1.58-12
La-141	2.37-12
La-140	5.00-13
Y-92	1.24-11
Y-91m	1.22-11
Total	7.01-10

---

(1) Design basis:

a. Density = 1 gm/cc

b. Output of floor drain sampling tank Table 12.2-50

c. Filter - DF = 1 for all isotopes, demineralizer - DF = 100 for anions and DF = 2 for Cs and Rb

(2) 1.81-12 = 1.81×10^{-12}

(3) Isotopes with specific activity less than 1.00-13 Ci/g are not listed

(Historical Information)

TABLE 12.2-52

DETERGENT DRAIN TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	3.75-06
Br-84	1.54-06
I-131	2.15-05
I-132	9.62-05
I-133	1.06-04
I-134	2.16-05
I-135	8.27-05
Sr-89	5.49-06
Sr-91	5.77-05
Sr-92	3.09-05
Mo-99	3.37-05
Tc-99m	1.85-04
Tc-101	3.39-06
Te-132	7.78-05
Cs-138	1.06-05
Ba-139	2.30-05
Ba-140	1.51-05
Ba-141	5.30-06
Ba-142	3.24-06
Np-239	3.59-04
Mn-56	2.67-06
Co-58	1.72-06
La-142	3.24-06
La-141	5.22-06
La-140	2.76-06
Y-92	3.00-05
Y-91m	3.34-05
Total	1.23-03

(Historical Information)

TABLE 12.2-52 (Cont)

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(1) Design basis:

a. Tank volume = 1000 gallons

b. Input rate of 0.695 gpm for 1 day

c. Input activity is the normalized primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11, of 0.00104 μCi/cc

(2) 3.75-06 = 3.75×10^{-6}

(3) Isotopes with an activity less than 1.00-06 Ci are not listed

(Historical Information)

TABLE 12.2-53

DETERGENT DRAIN PUMP SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci)[2][3]
Br-83	6.84-12
Br-84	1.28-11
Br-85	7.76-12
I-131	5.93-12
I-132	5.47-11
I-133	4.10-11
I-134	1.10-10
I-135	5.93-11
Sr-89	1.46-12
Sr-91	3.19-11
Sr-92	5.02-11
Mo-99	1.00-11
Tc-99m	1.28-10
Tc-101	6.39-11
Te-132	2.28-11
Cs-138	8.67-11
Ba-139	7.30-11
Ba-140	4.11-12
Ba-141	7.76-11
Ba-142	7.76-11
Np-239	1.10-10
Mn-56	4.56-12
Total	1.04-09

(Historical Information)

TABLE 12.2-53 (Cont)

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(1) Design basis:

a. Activity is the normalized primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11, of 0.00104 Ci/cc

b. Lab and personnel decontamination drains

(2) 6.84-12 = 6.84×10^{-12}

(3) Isotopes with specific activity less than 1.0-12 Ci/g are not listed

(Historical Information)

TABLE 12.2-54

DETERGENT DRAIN FILTER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	3.76-06
Br-84	1.54-06
I-131	2.41-04
I-132	4.31-04
I-133	1.90-04
I-134	2.16-05
I-135	9.02-05
Sr-89	1.37-04
Sr-90	1.24-05
Sr-91	7.03-05
Sr-92	3.10-05
Zr-95	1.78-06
Nb-95	2.10-06
Mo-99	1.52-04
Tc-99m	3.09-04
Tc-101	3.39-06
Te-129m	1.55-06
Te-132	4.03-04
Cs-134	8.17-06
Cs-136	2.93-06
Cs-137	1.24-05
Cs-138	1.06-05
Ba-139	2.30-05
Ba-140	2.31-04
Ba-141	5.30-06
Ba-142	3.24-06
Ce-141	4.02-06
Ce-144	1.80-06
Pr-143	1.10-06

TABLE 12.2-54 (Cont)

Isotope	Activity (Ci)(2)(3)
Np-239	1.40-03
Cr-51	3.65-06
Mn-56	2.68-06
Co-58	4.50-05
Co-60	5.16-06
W-187	1.49-06
Pr-144	1.80-06
La-142	3.24-06
La-141	5.30-06
La-140	2.22-04
Y-92	3.10-05
Y-91	2.06-05
Y-91m	4.15-05
Y-90	1.09-05
Total	4.20-03

---

(1) Design basis:

a. Input activity is the normalized primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11, of 0.00104 μCi/cc

b. Process flow of 1000 gal/day for 30 days

c. 100% accumulation efficiency for all isotopes

(2) 3.76-06 = 3.76×10^{-6}

(3) Isotopes with an activity less than 1.00-06 Ci are not listed

(Historical Information)

TABLE 12.2-55

RWCU PHASE SEPARATOR SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
I-131	1.75+03
I-132	2.19+03
I-133	5.36+02
I-134	4.49+01
I-135	1.95+02
Sr-89	1.91+03
Sr-90	2.16+02
Sr-91	1.58+02
Sr-92	6.53+01
Zr-95	2.58+01
Nb-95	3.36+01
Mo-99	7.68+02
Tc-99m	1.08+03
Ru-103	1.01+01
Te-129m	1.96+01
Te-132	2.17+03
Cs-134	1.40+02
Cs-136	2.70+01
Cs-137	2.16+02
Cs-138	2.16+01
Ba-139	4.80+01
Ba-140	2.07+03
Ba-141	1.05+01
Ce-141	5.04+01
Ce-144	2.99+01
Pr-143	1.02+01
Np-239	6.56+03
Cr-51	4.34+01
Co-58	6.62+02

(Historical Information)

TABLE 12.2-55 (Cont)

Isotope	Activity (Ci)(2)(3)
Co-60	8.91+01
Pr-144	2.99+01
La-141	1.13+01
La-140	2.21+03
Te-129	1.25+01
Y-92	6.60+01
Y-91	3.00+02
Y-91m	9.38+01
Y-90	2.09+02
Total	2.42+04

---

(1) Design basis:

a. RWCU phase separator sludge volume = 870 gallons at 15 weight percent solids

b. Total number of backwash = 36 with various decay times (6.8 days to 54.4 days) for 60 days

(2) $1.75+03 = 1.75 \times 10^3$

(3) Isotopes with an activity less than 1.00+01 Ci are not listed

(Historical Information)

TABLE 12.2-56

RWCU PHASE SEPARATOR DECANT PUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci)[2][3]
Br-83	1.50-10
Br-84	2.80-10
Br-85	1.70-10
I-131	1.30-10
I-132	1.20-09
I-133	9.00-10
I-134	2.40-09
I-135	1.30-09
Sr-89	3.20-11
Sr-91	7.00-10
Sr-92	1.10-09
Mo-99	2.20-10
Tc-99m	2.80-09
Tc-101	1.40-09
Te-132	5.00-10
Cs-138	1.90-09
Ba-139	1.60-09
Ba-140	9.00-11
Ba-141	1.70-09
Ba-142	1.70-09
Np-239	2.40-09
Mn-56	1.00-10
Co-58	1.00-11
Total	2.28-08

(Historical Information)

TABLE 12.2-56 (Cont)

---

(1) Design basis:

a. Activity of decant stream = 0.002 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

b. No N-16 or noble gases are included

c. Density = 1.0 g/cc.

(2) 1.50-09 = 1.50×10^{-9}

(3) Isotopes with specific activity less than 1.00-11 Ci/g are not listed

(Historical Information)

TABLE 12.2-57

RWCU SLUDGE DISCHARGE PUMP

SHIELDING DESIGN SOURCE TERMS (1)

Isotope	Specific Activity (Ci) (2)(3)
I-131	1.77-04
I-132	2.21-04
I-133	5.41-05
I-134	4.54-06
I-135	1.97-05
Sr-89	1.93-04
Sr-90	2.18-05
Sr-91	1.60-05
Sr-92	6.60-06
Zr-95	2.61-06
Nb-95	3.39-06
Mo-99	7.76-05
Tc-99m	1.09-04
Ru-103	1.02-06
Te-129m	1.98-06
Te-132	2.19-04
Cs-134	1.41-05
Cs-136	2.73-06
Cs-137	2.18-05
Cs-138	2.18-06
Ba-139	4.85-06
Ba-140	2.09-04
Ba-141	1.06-06
Ce-141	5.09-06
Ce-144	3.02-06
Pr-143	1.03-06
Nd-147	2.79-07
Np-239	6.63-04

(Historical Information)

TABLE 12.2-57 (Cont)

Isotope	Specific Activity (Ci)[(2)(3)]
Cr-51	4.38-06
Co-58	6.69-05
Co-60	9.00-06
Pr-144	3.02-06
La-141	1.14-06
La-140	2.23-04
Te-129	1.26-06
Y-92	6.67-06
Y-91	3.03-05
Y-91m	9.47-06
Y-90	2.11-05
Total	2.44-03

---

(1) Design basis:

a. RWCU phase separator, Table 12.2-55, batch volume diluted to 5 weight percent solid prior to transfer to centrifuge feed tank

b. Batch size of 36 backwash for 60 days

c. Density = 1.0 g/cc

(2) 1.77-04 = 1.77×10^{-4}

(3) Isotopes with specific activity less than 1.00-07 Ci/g are not listed

(Historical Information)

TABLE 12.2-58

WASTE COLLECTOR TANK SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	1.18+00
Br-84	4.86-01
I-131	7.05+00
I-132	3.12+01
I-133	3.43+01
I-134	6.82+00
I-135	2.63+01
Sr-89	1.80+00
Sr-90	1.36-01
Sr-91	1.85+01
Sr-92	9.75+00
Mo-99	1.10+01
Tc-99m	5.89+01
Tc-101	1.07+00
Te-132	2.54+01
Cs-137	1.36-01
Cs-138	3.34+00
Ba-139	7.25+00
Ba-140	4.96+00
Ba-141	1.67+00
Ba-142	1.02+00
Np-239	1.17+02
Na-24	1.35-01
Mn-56	8.43-01
Co-58	5.64-01
W-187	2.42-01
La-142	1.02+00
La-141	1.65+00
La-140	9.37-01

(Historical Information)

TABLE 12.2-58 (Cont)

Isotope	Activity (Ci)[(2)(3)]
Y-92	9.51+00
Y-91	1.38-01
Y-91m	1.07+01
Total	3.95+02

---

(1) Design basis:

a. Source volume = 30,000 gallons

b. Input rate of 20 gpm for 1500 minutes

c. Input activity = primary coolant activity Tables 12.2-8, 12.2-9, and 12.2-11

d. Density = 1.0 g/cc

(2) 1.18 + 00 = 1.18×10^0

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-59

WASTE SURGE TANK SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	1.19+00
Br-84	4.86-01
I-131	1.65+01
I-132	6.10+01
I-133	5.28+01
I-134	6.82+00
I-135	2.84+01
Sr-89	4.46+00
Sr-90	3.40-01
Sr-91	2.19+01
Sr-92	9.77+00
Mo-99	2.29+01
Tc-99m	7.32+01
Tc-101	1.07+00
Te-132	5.43+01
Cs-134	2.27-01
Cs-136	1.46-01
Cs-137	3.40-01
Cs-138	3.34+00
Ba-139	7.24+00
Ba-140	1.19+01
Ba-141	1.67+00
Ba-142	1.02+00
Ce-141	1.37-01
Np-239	2.37+02
Na-24	1.85-01
Cr-51	1.37-01
Mn-56	8.44-01
Co-58	1.40+00

(Historical Information)

TABLE 12.2-59 (Cont)

Isotope	Activity (Ci)(2)(3)
Co-60	1.42-01
W-187	3.93-01
La-142	1.02+00
La-141	1.67+00
La-140	4.70+00
Y-92	8.77+00
Y-91	5.15-01
Y-91m	1.29+01
Total	6.51+02

(1) Design basis:

a. Source volume = 75,000 gallons

b. Input rate of 20 gpm for 3750 minutes

c. Input activity = primary coolant activity Tables 12.2-8, 12.2-9, and 12.2-11

d. Density = 1.0 g/cc

(2) 1.19 + 00 = 1.19×10^{0}

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-60

WASTE COLLECTOR AND SURGE PUMP SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (Ci)(2)(3)
Br-83	7.50-08
Br-84	1.40-07
Br-85	8.50-08
I-131	6.50-08
I-132	6.00-07
I-133	4.50-07
I-134	1.20-06
I-135	6.50-07
Sr-89	1.60-08
Sr-90	1.20-09
Sr-91	3.50-07
Sr-92	5.50-07
Mo-99	1.10-07
Tc-99m	1.40-06
Tc-101	7.00-07
Te-129m	1.64-01
Te-132	2.50-07
Cs-137	1.20-09
Cs-138	9.50-07
Ba-139	18.00-07
Ba-140	4.50-08
Ba-141	8.50-07
Ba-142	8.50-07
Np-239	1.20-06
Na-24	2.00-09
Mn-56	5.00-08
Co-58	5.00-09
W-187	3.00-09
Total	1.14-05

(1) Design basis:

a. Source activity is equal to primary coolant activity Table 12.2-8, 12.2-9 and 12.2-11

b. Primary coolant equilibrium activity with no decay

(2) 7.50-08 = 7.50×10^{-8}

(3) Isotopes with specific activity less than 1.00-09 Ci/g are not listed

(Historical Information)

TABLE 12.2-61

WASTE FILTER SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	8.74-01
Br-84	3.59-01
I-131	5.40+00
I-132	2.38+01
I-133	2.60+01
I-134	5.03+00
I-135	1.96+01
Sr-89	2.77+00
Sr-90	1.05-01
Sr-91	1.38+01
Sr-92	7.20+00
Mo-99	8.40+00
Tc-99m	4.40+01
Tc-101	7.90-01
Te-132	1.94+01
Cs-137	1.05-01
Cs-138	2.47+00
Ba-139	5.35+00
Ba-140	3.81+00
Ba-141	1.23+00
Ba-142	7.54-01
Np-239	8.95+01
Mn-56	6.22-01
Co-58	4.33-01
W-187	1.84-01
La-142	7.54-01
La-141	1.22+00
La-140	7.45-01
Y-92	7.05+00

(Historical Information)

TABLE 12.2-61 (Cont)

Isotope	Activity (Ci)[(2)(3)]
Y-91	1.09-01
Y-91m	8.02+00
Total	3.00+02

---

(1) Design basis:

a. Average input activity to filter is the primary coolant activity normalized to 0.924 μCi/g, Tables 12.2-8, 12.2-9, and 12.2-11

b. Process rate of 182 gpm for 26 hours

c. 100 percent accumulation efficiency for all isotopes

(2) 8.74-01 = 8.74×10^{-1}

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-62

WASTE DEMINERALIZER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	2.77-01
Br-84	1.14-01
I-131	1.90+01
I-132	3.18+01
I-133	1.40-01
I-134	1.59+00
I-135	6.65+00
Sr-89	1.54+01
Sr-90	1.59+00
Sr-91	5.19+00
Sr-92	2.28+00
Zr-95	2.04-01
Nb-95	2.56-01
Mo-99	1.12+01
Tc-99m	2.28+01
Tc-101	2.50-01
Te-129m	1.64-01
Te-132	2.98+01
Cs-134	1.03+00
Cs-136	2.57-01
Cs-137	1.59+00
Cs-138	7.81-01
Ba-139	1.69+00
Ba-140	1.99+01
Ba-141	3.91-01
Ba-142	2.39-01
Ce-141	4.21-01
Ce-144	2.24-01
Np-239	1.04+02

(Historical Information)

TABLE 12.2-62 (Cont)

Isotope	Activity (Ci)(2)(3)
Cr-51	3.71-01
Mn-56	1.97-01
Co-58	5.20+00
Co-60	6.56-01
W-187	1.10-01
Pr-144	2.24-01
La-142	2.39-01
La-141	3.91-01
La-140	1.97+01
Te-129	1.05-01
Y-92	2.28+00
Y-91	2.36+00
Y-91m	3.06+00
Y-90	1.48+00
Total	3.30+02

---

(1) Design basis:
   a. Resin volume = 1421 gallons
   b. Average input activity to demineralizer is the primary coolant activity normalized to 0.919 μCi/g, Tables 12.2-8, 12.2-9, and 12.2-11
   c. Process rate of 58 gpm for 52 days
   d. 100 percent accumulation efficiency for all isotopes

(2) $2.77\text{-}01 = 2.77 \times 10^{-1}$

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-63

WASTE SAMPLE TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	5.71-03
Br-84	1.20-03
I-131	4.26-02
I-132	1.82-01
I-133	2.02-01
I-134	2.29-02
I-135	1.47-01
Sr-89	1.09-02
Sr-91	1.06-01
Sr-92	4.82-02
Mo-99	6.12-02
Tc-99m	3.31-01
Tc-101	1.30-03
Te-132	1.53-01
Cs-134	2.75-02
Cs-136	1.84-02
Cs-137	4.13-02
Cs-138	4.16-01
Ba-139	2.98-02
Ba-140	3.00-02
Ba-141	2.57-03
Np-239	7.03-01
Mn-56	4.12-03
Co-58	3.42-03
W-187	1.43-03
La-142	4.81-03
La-141	9.17-03
La-140	6.03-03
Y-92	5.68-02

(Historical Information)

TABLE 12.2-63 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-91m	6.42-02
Total	2.73-00

---

(1) Design basis:
  a. Source volume = 18,000 gallons
  b. Input rate of 182 gpm for 100 minutes
  c. Input activity = output activity of waste collector tank, Table 12.2-58, with DF=1 for waste filter, and DF=2 for Cs and Rb, DF=100 for all other isotopes for waste demineralizer
  d. Density = 1.0 g/cc

(2) 5.71-03 = 5.71×10^{-4}

(3) Isotopes with an activity less than 1.00-04 Ci are not listed

(Historical Information)

TABLE 12.2-64

WASTE SAMPLE PUMP SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (Ci/g)(2)(3)
Br-83	8.39-11
Br-84	1.76-11
I-131	6.26-10
I-132	2.68-09
I-133	3.00-09
I-134	3.37-10
I-135	2.16-09
Sr-89	1.60-10
Sr-91	1.56-09
Sr-92	7.09-10
Mo-99	9.73-10
Tc-99m	4.87-09
Tc-101	1.91-11
Te-132	2.25-09
Cs-134	4.05-10
Cs-136	2.71-10
Cs-137	6.05-10
Cs-138	6.10-09
Ba-139	4.38-10
Ba-140	4.41-10
Ba-141	3.78-11
Ba-142	1.44-11
Np-239	1.03-08
Na-24	1.15-11
Mn-56	6.06-11
Co-58	5.03-11
W-187	2.10-11
La-142	7.07-11
La-141	1.35-10

(Historical Information)

TABLE 12.2-64 (Cont)

Isotope	Specific Activity (Ci/g)[(2)(3)]
La-140	8.86-11
Y-92	8.35-10
Y-91	1.30-11
Y-91m	9.44-10
Total	4.03-08

---

(1) Design basis:

a. Input activity is the output activity of the waste sample tank, Table 12.2-63

b. Density = 1.0 g/cc

(1) 8.39-11 = 8.39×10^{-12}

(3) Isotopes with specific activity less than 1.0-12 Ci/g are not listed

(Historical Information)

TABLE 12.2-65

CHEMICAL WASTE TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	6.12-03
Br-84	2.52-03
I-131	3.26-02
I-132	1.48-01
I-133	1.65-01
I-134	3.53-02
I-135	1.33-01
Sr-89	8.31-03
Sr-91	9.15-02
Sr-92	5.04-02
Mo-99	5.13-02
Tc-99m	2.94-01
Tc-101	5.54-03
Te-132	1.19-01
Cs-138	1.73-02
Ba-139	3.75-02
Ba-140	2.29-02
Ba-141	8.65-03
Ba-142	5.29-03
Np-239	5.48-01
Mn-56	4.36-03
Co-58	2.60-03
W-187	1.16-03
La-142	5.29-03
La-141	8.47-03
La-140	3.91-03

(Historical Information)

TABLE 12.2-65 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-92	4.84-02
Y-91m	5.26-02
Total	1.91+00

---

(1) Design basis:

a. Source volume = 4000 gallons

b. Input rate of 3 gpm for 0.926 day

c. Density = 1.0 g/cc

d. Input activity is primary coolant activity normalized to 0.1 μCi/cc (Tables 12.2-8, 12.2-9, and 12.2-11)

(2) 6.12-03 = 6.12×10^{-3}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-66

CHEMICAL WASTE PUMP SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	4.05-10
Br-84	1.66-10
I-131	2.16-09
I-132	9.80-09
I-133	1.09-08
I-134	2.33-09
I-135	8.76-09
Sr-89	5.49-10
Sr-91	6.04-09
Sr-92	3.33-09
Mo-99	3.39-09
Tc-99m	1.94-09
Tc-101	3.66-10
Te-132	7.82-09
Cs-138	1.14-09
Ba-139	2.48-09
Ba-140	1.52-09
Ba-141	5.72-10
Ba-142	3.49-10
Np-239	3.62-08
Mn-56	2.88-10
Co-58	1.72-10
La-142	3.49-10
La-141	5.60-10
La-140	2.58-10
Y-92	3.20-09
Y-91m	3.48-09
Total	1.09-07

(Historical Information)

TABLE 12.2-66 (Cont)

---

(1) Design basis:

a. Activity is the primary coolant activity normalized to 0.1 μCi/g, Tables 12.2-8, 12.2-9, and 12.2-11

b. Density = 1.0 g/cc

(2) 4.05-10 = 4.05 x 10^{-10}.

(3) Isotopes with specific activity less than 1.00-10 Ci/g are not listed

(Historical Information)

TABLE 12.2-67

DECONTAMINATION SOLUTION EVAPORATOR SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	1.47-08
Br-84	6.02-09
I-131	7.81-08
I-132	3.55-07
I-133	3.94-07
I-134	8.45-08
I-135	3.17-07
Sr-89	1.99-08
Sr-90	1.50-09
Sr-91	2.19-07
Sr-92	1.21-07
Mo-99	1.23-07
Tc-99m	7.04-07
Tc-101	1.33-08
Te-132	2.84-07
Cs-134	1.00-09
Cs-137	1.50-09
Cs-138	4.14-08
Ba-139	8.98-08
Ba-140	5.49-08
Ba-141	2.07-08
Ba-142	1.27-08
Np-239	1.31-06
Na-24	1.56-09
Mn-56	1.04-08
Co-58	6.23-09
W-187	2.77-09
La-142	1.27-08

(Historical Information)

TABLE 12.2-67 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)
La-141	2.03-08
La-140	9.36-09
Y-92	1.16-07
Y-91	1.42-09
Y-91m	1.26-07
Total	4.57-06

(1) Design basis:

a. Input activity of chemical waste tank content at 0.39 weight percent solids (Table 12.2-65)

b. Maximum concentration to 10 weight % solid with no decay, 110 gallons/batch

c. Density = 1.0 g/cc

d. Concentrate volume = 110 gallons

(2) 1.47-08 = 1.47×10^{-8}

(3) Isotopes with specific activity less than 1.00-09 Ci/g are not listed

(Historical Information)

TABLE 12.2-68

DECONTAMINATION CONCENTRATE WASTE TRANSFER/RECYCLE PUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	1.47-08
Br-84	6.02-09
I-131	7.81-08
I-132	3.55-07
I-133	3.94-07
I-134	8.45-08
I-135	3.17-07
Sr-89	1.99-08
Sr-90	1.50-09
Sr-91	2.19-07
Sr-92	1.21-07
Mo-99	1.23-07
Tc-99m	7.04-07
Tc-101	1.33-08
Te-132	2.84-07
Cs-134	1.00-09
Cs-137	1.50-09
Cs-138	4.14-08
Ba-139	8.98-08
Ba-140	5.49-08
Ba-141	2.07-08
Ba-142	1.27-08
Np-239	1.31-06
Na-24	1.56-09
Mn-56	1.04-08
Co-58	6.23-09
W-187	2.77-09
La-142	1.27-08

(Historical Information)

TABLE 12.2-68 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)
La-141	2.03-08
La-140	9.36-09
Y-92	1.16-07
Y-91	1.42-09
Y-91m	1.26-07
Total	4.57-06

---

(1) Design basis:

a. Activity is equal to the maximum concentrated waste in decontamination solution evaporator (Table 12.2-67)

b. Density of 1.0 g/cc.

(2) 1.47-08 = 1.47×10^{-8}

(3) Isotopes with specific activity less than 1.00-09 Ci/g are not listed

(Historical Information)

TABLE 12.2-69

DECONTAMINATION SOLUTION CONCENTRATE WASTE TANK

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	5.70-03
Br-84	2.34-03
I-131	1.60-01
I-132	4.81-01
I-133	2.86-01
I-134	3.28-02
I-135	1.37-01
Sr-89	4.84-02
Sr-90	3.78-03
Sr-91	1.07-01
Sr-92	4.70-02
Mo-99	1.80-01
Tc-99m	4.19-01
Tc-101	5.15-03
Te-132	4.43-01
Cs-134	2.51-03
Cs-136	1.49-03
Cs-137	3.78-03
Cs-138	1.61-02
Ba-139	3.49-02
Ba-140	1.21-01
Ba-141	8.04-03
Ba-142	4.92-03
Ce-141	1.52-03
Np-239	1.77+00
Cr-51	1.46-03
Mn-56	4.06-03
Co-58	1.53-02

(Historical Information)

TABLE 12.2-69 (Cont)

Isotope	Activity (Ci)(2)(3)
Co-60	1.57-03
W-187	2.23-03
La-142	4.92-03
La-141	8.04-03
La-140	7.90-02
Y-92	4.70-02
Y-91	6.57-03
Y-91m	6.30-02
Y-90	1.94-03
Total	4.56+00

---

(1) Design basis:

a. Source volume = 660 gallons

b. Process time of 6 days at 0.074 gpm

c. Decontamination solution evaporation volume reduction of 36.4 (Table 12.2-67)

d. Density = 1.0 g/cc

(2) 5.70-03 = 5.70×10^{-3}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-70

DECONTAMINATION CONCENTRATE WASTE PUMP

SHIELDING DESIGN SOURCE TERMS (1)

Isotope	Specific Activity (Ci/g) (2) (3)
Br-83	2.28-09
I-131	6.39-08
I-132	1.92-07
I-133	1.14-07
I-134	1.31-08
I-135	5.48-08
Sr-89	1.94-08
Sr-90	1.51-09
Sr-91	4.27-08
Sr-92	1.88-08
Mo-99	7.18-08
Tc-99m	1.68-07
Tc-101	2.06-09
Te-132	1.77-07
Cs-134	1.00-09
Cs-137	1.51-09
Cs-138	6.43-09
Ba-139	1.40-08
Ba-140	4.84-08
Ba-141	3.22-09
Ba-142	1.97-09
Np-239	7.08-07
Mn-56	1.63-09
Co-58	6.12-09
La-142	1.97-09
La-141	3.22-09
La-140	3.16-08
Y-92	1.88-08

(**Historical Information**)

TABLE 12.2-70 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)
Y-91	2.63-09
Y-91m	2.52-08
Total	1.82-06

---

(1) Design basis:

a. Activity equal to decontamination concentrate waste tank activity (Table 12.2-68)

b. Density = 1.0 g/cc

(2) 2.28-09 = 2.28×10^{-9}

(3) Isotopes with specific activity less than 1.00-09 Ci/g are not listed

(Historical Information)

TABLE 12.2-71

WASTE NEUTRALIZER TANK SHIELDING DESIGN SOURCE TERM[1]

Isotope	Activity (Ci)[2][3]
Br-83	1.88+00
I-131	4.03+02
I-132	4.44+01
I-133	2.58+02
I-134	1.39+00
I-135	9.40+01
Sr-89	1.89+01
Sr-90	1.97+00
Sr-91	6.11+00
Sr-92	1.30+00
Mo-99	1.15+01
Tc-99m	1.91+01
Te-132	3.07+01
Cs-134	1.27+00
Cs-137	1.96+00
Ba-140	2.43+01
Np-239	1.05+02
Co-58	6.19+00
La-140	2.44+01
Y-92	2.80+00
Y-91	4.10+00
Y-91m	3.93+00
Y-90	1.86+00
Total	1.07+03

(Historical Information)

TABLE 12.2-71 (Cont)

---

(1) Design basis:

a. Source volume = 12,730 gallons

b. Input activity of condensate demineralizer with 4 hours decay (Table 12.2-119)

c. Density = 1.0 g/cc

(2) 1.88+00 = 1.88×10^{0}

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-72

CONCENTRATE FEED/WASTE NEUTRALIZER PUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (Ci/g)(2)(3)
Br-83	3.90-08
I-131	8.39-06
I-132	9.23-07
I-133	5.37-06
I-134	2.89-08
I-135	1.96-06
Sr-89	3.94-07
Sr-90	4.10-08
Sr-91	1.27-07
Sr-92	2.71-08
Mo-99	2.40-07
Tc-99m	3.98-07
Te-132	6.39-07
Cs-134	2.65-08
Cs-137	4.07-08
Ba-139	1.51-08
Ba-140	5.05-07
Ce-141	1.76-08
Np-239	2.19-06
Co-58	1.29-07
Co-60	1.68-08
La-141	1.01-08
La-140	5.07-07
Y-92	5.83-08
Y-91	8.54-08

(Historical Information)

TABLE 12.2-72 (Cont)

Isotope	Specific Activity (Ci/g)[2][3]
Y-91m	8.18-08
Y-90	3.86-08
Total	2.23-05

---

(1) Design basis:

a. Activity is the output activity of waste neutralizer tank (Table 12.2-71)

b. Density = 1.0 g/cc

(2) 3.90-08 = 3.90×10^{-8}

(3) Isotopes with specific activity less than 1.00-08 Ci/g are not listed

(Historical Information)

TABLE 12.2-73

WASTE EVAPORATOR SHIELDING DESIGN SOURCE TERMS[1]

NOTE: Waste Evaporator has been abandoned in place.

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	3.08-07
I-131	2.06-04
I-132	1.78-05
I-133	1.17-04
I-135	3.22-05
Sr-89	9.80-06
Sr-90	1.02-06
Sr-91	2.38-06
Sr-92	2.43-07
Zr-95	1.30-07
Nb-95	1.63-07
Mo-99	5.74-06
Tc-99m	8.16-06
Te-132	1.54-05
Cs-134	6.60-07
Cs-136	1.44-07
Cs-137	1.02-06
Ba-140	1.25-05
Ce-141	4.38-07
Np-239	5.21-05
Cr-51	2.18-07
Co-58	3.21-06
Co-60	4.20-07
Pr-144	1.42-07
La-141	1.24-07
La-140	1.26-05
Y-92	8.79-07
Y-91	2.13-06

(Historical Information)

TABLE 12.2-73 (Cont)

Isotope	Specific Activity $(Ci/g)^{(2)(3)}$
Y-91m	1.54-06
Y-90	9.67-07
Total	5.05-04

(1) Design basis:

a. Input activity is the output activity of waste neutralizer tank with 4 hours decay (Table 12.2-71)

b. Volume reduction = 25

c. Concentrate batch volume = 509 gallons

d. Density = 1.0 g/cc

(2) 3.08-07 = 3.08×10^{-7}

(3) Isotopes with specific activity less than 1.00-07 Ci/g are not listed

(Historical Information)

TABLE 12.2-74

WASTE EVAPORATOR CONDENSER SHIELDING DESIGN SOURCE TERMS[1]

NOTE: Waste Evaporator has been abandoned in place.

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	3.90-11
I-131	8.39-09
I-132	9.23-10
I-133	5.37-09
I-134	2.89-11
I-135	1.96-09
Sr-89	3.94-11
Sr-91	1.27-11
Mo-99	2.40-11
Tc-99m	3.98-11
Te-132	6.39-11
Ba-140	5.05-11
Np-239	2.19-10
Co-58	1.29-11
La-140	5.07-11
Total	1.72-08

(1) Design basis:

a. Input activity to the waste evaporator is the output activity of the waste neutralizer tank (Table 12.2-71)

b. DF = 10^3 for anions and DF = 10^4 for other isotopes

c. Density = 1.0 g/cc

(2) 3.90-11 = 3.90×10^{-11}

(3) Isotopes with specific activity less than 1.00-11 Ci/g are not listed

(Historical Information)

TABLE 12.2-75

WASTE EVAPORATOR DISTILLATE TANK
SHIELDING DESIGN SOURCE TERMS[1]

NOTE: Waste Evaporator Distillate Tank has been abandoned in place.

Isotope	Activity (Ci)[2][3]
Br-83	1.62-04
I-131	3.49-02
I-132	3.84-03
I-133	2.23-02
I-134	1.20-04
I-135	8.15-03
Sr-89	1.64-04
Sr-90	1.71-05
Sr-91	5.28-05
Mo-99	1.00-04
Tc-99m	1.66-04
Te-132	2.66-04
Cs-134	1.10-05
Cs-137	1.69-05
Ba-140	2.10-04
Np-239	9.11-04
Co-58	5.37-05
La-140	2.11-04
Y-92	2.43-05
Y-91	3.55-05
Y-91m	3.40-05
Y-90	1.61-05
Total	7.18-02

(Historical Information)

TABLE 12.2-75 (Cont)

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(1) Design basis:

   a. Source volume = 1100 gallons

   b. Input activity is the waste evaporator condenser activity (Table 12.2-74)

   c. Density = 1.0 g/cc

(2) 1.62-04 = 1.62×10^{-4}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-76

WASTE EVAPORATOR DISTILLATE TRANSFER PUMP
SHIELDING DESIGN SOURCE TERMS[1]

NOTE: Waste Evaporator Distillate Transfer Pump has been abandoned in place.

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	3.90-11
I-131	8.39-09
I-132	9.23-10
I-133	5.37-09
I-134	2.89-11
I-135	1.96-09
Sr-89	3.94-11
Sr-91	1.27-11
Mo-99	2.40-11
Tc-99m	3.98-11
Te-132	6.39-11
Ba-140	5.05-11
Np-239	2.19-10
Co-58	1.29-11
La-140	5.07-11
Total	1.72-08

(1) Design basis:
   a. Activity is equal to the waste evaporator condenser activity (Table 12.2-74)
   b. Density = 1.0 g/cc

(2) 3.90-11 = 3.90×10^{-11}

(3) Isotopes with specific activity less than 1.00-11 Ci/g are not listed

(Historical Information)

TABLE 12.2-77

WASTE EVAPORATOR CONCENTRATE TRANSFER/RECYCLE PUMP SHIELDING DESIGN SOURCE TERMS[1]

NOTE: Waste Evaporator Concentrate Transfer/Recycle Pump has been abandoned in place.

Isotope	Specific Activity (Ci/g)[2][3]
Br-83	3.08-07
I-131	2.06-04
I-132	1.78-05
I-133	1.17-04
I-135	3.22-05
Sr-89	9.80-06
Sr-90	1.02-06
Sr-91	2.38-06
Sr-92	2.42-07
Zr-95	1.30-07
Nb-95	1.63-07
Mo-99	5.74-06
Tc-99m	8.16-06
Te-132	1.54-05
Cs-134	6.60-07
Cs-136	1.44-07
Cs-137	1.02-06
Ba-140	1.25-05
Ce-141	4.38-07
Np-239	5.21-05
Cr-51	2.18-07
Co-58	3.21-06
Co-60	4.20-07
Pr-144	1.42-07
La-141	1.24-07
La-140	1.26-05
Y-92	8.79-07
Y-91	2.13-06

(Historical Information)

TABLE 12.2-77 (Cont)

Isotope	Specific Activity (Ci/g)[2][3]
Y-91m	1.54-06
Y-90	9.67-07
Total	5.05-04

---

(1) Design basis:

a. Activity is the output activity of the waste evaporator (Table 12.2-73)

b. Volume reduction of 25 in waste evaporator

c. Density = 1.0 g/cc

(2) 3.08-07 = 3.08×10^{-7}

(3) Isotopes with specific activity less than 1.00-07 Ci/g are not listed

(Historical Information)

TABLE 12.2-78

CONCENTRATE WASTE TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
I-131	3.89+03
I-132	1.53+02
I-133	2.83+02
I-135	6.24+01
Sr-89	2.12+02
Sr-90	2.08+01
Sr-91	4.74+00
Zr-95	2.84+00
Nb-95	3.49+00
Mo-99	4.55+01
Tc-99m	4.86+01
Ru-103	1.10+00
Te-129m	2.17+00
Te-132	1.45+02
Cs-134	1.38+01
Cs-136	3.13+00
Cs-137	2.12+01
Ba-140	2.68+02
Ce-141	9.76+00
Ce-144	2.99+00
Pr-143	1.18+00
Np-239	3.36+02
Cr-51	4.87+00
Co-58	6.94+01
Co-60	8.77+00
Pr-144	2.99+00
La-140	3.02+02
Te-129	1.39+00
Y-92	1.69+00

(Historical Information)

TABLE 12.2-78 (Cont)

Isotope	Activity (Ci) (2) (3)
Y-91	4.69+01
Y-91m	3.05+00
Y-90	2.09+01
Rh-103m	1.07+00
Total	5.99+03

---

(1) Design basis:

a. Source volume = 9000 gallons

b. 18 batches of waste evaporator concentrate (Table 12.2-77)

c. 25 weight percent solid

d. Density = 1.0 g/cc

(2) $3.89+03 = 3.89 \times 10^3$

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-79

CONCENTRATE WASTE PUMP SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
I-131	1.14-04
I-132	4.50-06
I-133	8.32-06
I-135	1.84-06
Sr-89	6.23-06
Sr-90	6.12-07
Sr-91	1.39-07
Nb-95	1.03-07
Mo-99	1.34-06
Tc-99m	1.43-06
Te-132	4.26-06
Cs-134	4.06-07
Cs-137	6.23-07
Ba-140	7.88-06
Ce-141	2.87-07
Np-239	9.88-06
Cr-51	1.43-07
Co-58	2.04-06
Co-60	2.58-07
La-140	8.88-06
Y-91	1.38-06
Y-90	6.15-07
Total	1.75-04

(Historical Information)

TABLE 12.2-79 (Cont)

---

(1) Design basis:

a. Activity is the output activity of concentrate waste tank without any decay (Table 12.2-78)

b. 25 weight percent solid

c. Density = 1.0 g/cc

(2) 1.14-04 = 1.14×10^{-4}

(3) Isotopes with specific activity less than 1.00-07 Ci/g are not listed

(Historical Information)

TABLE 12.2-80

RADWASTE CATION REGENERATION TANK

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	2.77-01
Br-84	1.14-01
I-131	1.90-01
I-132	3.18+01
I-133	1.40+01
I-134	1.59+00
I-135	6.65+00
Sr-89	1.54+01
Sr-90	1.59+00
Sr-91	5.19+00
Sr-92	2.28+00
Zr-95	2.04-01
Nb-95	2.56-01
Mo-99	1.12+01
Tc-99m	2.28+01
Tc-101	2.50-01
Te-129m	1.64-01
Te-132	2.98+01
Cs-134	1.03+00
Cs-136	2.57-01
Cs-137	1.59+00
Cs-138	7.81-01
Ba-139	1.69+00
Ba-140	1.99+01
Ba-141	3.91-01
Ba-142	2.39-01
Ce-141	4.21-01
Ce-144	2.24-01

(Historical Information)

TABLE 12.2-80 (Cont)

Isotope	Activity (Ci)(2)(3)
Np-239	1.04+02
Cr-51	3.71-01
Mn-56	1.97-01
Co-58	5.20+00
Co-60	6.56-01
W-187	1.10-01
Pr-144	2.24-01
La-142	2.39-01
La-141	3.91-01
La-140	1.97+01
Te-129	1.05-01
Y-92	2.28+00
Y-91	2.36+00
Y-91m	3.06+00
Y-90	1.48+00
Total	3.30+02

(1) Design basis:

a. Input activity is the accumulated activity in the waste demineralizer (Table 12.2-62)

b. Backwash volume = 2181 gallons

c. Density = 1.0 g/cc

(2) 2.77-01 = 2.7×10^{-1}

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-81

RADWASTE ANION REGENERATION TANK

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2]
Br-83	2.77-01
Br-84	1.14-01
Br-85	6.51-03
I-131	1.90+01
I-132	2.08+00
I-133	1.40+01
I-134	1.59+00
I-135	6.65+00
Total	4.37+01

(1) Design basis:

a. Input activity is the accumulated anion activity in the waste demineralizer (Table 12.2-62)

b. Backwash volume = 2181 gallons

c. Density = 1.0 g/cc

(2) $2.77\text{-}01 = 2.77 \times 10^{-1}$

(Historical Information)

TABLE 12.2-82

SPENT RESIN TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	4.12+00
Br-84	1.38+00
I-131	5.12+02
I-132	6.32+01
I-133	2.31+02
I-134	2.14+01
I-135	1.01+02
Sr-89	2.58+01
Sr-90	2.70+00
Sr-91	5.19+00
Sr-92	2.39+00
Mo-99	1.20+01
Tc-99m	2.06+01
Te-132	3.26+01
Cs-134	1.87+00
Cs-137	2.82+00
Ba-139	2.48+00
Ba-140	3.12+01
Ce-141	1.16+00
Np-239	1.06+02
Co-58	8.86+00
Co-60	1.78+00
Fe-59	1.04+00
La-140	3.26+01
Y-92	2.50+00
Y-91	5.21+00

(Historical Information)

TABLE 12.2-82 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-91m	3.09+00
Y-90	2.58+00
Total	1.24+03

---

(1) Design basis:

a. Source volume = 4362 gallons

b. Two consecutive batches of spent resin from condensate demineralizer. One batch with no decay and the other has decayed for 3.67 days

c. Input activity is the condensate demineralizer activity (Table 12.2-119)

(2) 5.12+02 = 5.12×10^2

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-83

SPENT RESIN PUMP SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)[2][3]
Br-83	2.50-07
Br-84	8.36-08
I-131	3.10-05
I-132	3.83-06
I-133	1.40-05
I-134	1.30-06
I-135	6.12-06
Sr-89	1.56-06
Sr-90	1.64-07
Sr-91	3.15-07
Sr-92	1.45-07
Mo-99	7.27-07
Tc-99m	1.25-06
Te-132	1.98-06
Cs-134	1.13-07
Cs-136	2.23-08
Cs-137	1.71-07
Cs-138	5.22-08
Cs-139	4.20-08
Ba-139	1.50-07
Ba-140	1.89-06
Ba-141	2.81-08
Pr-144	2.30-08
Ce-141	7.03-08
Ce-144	2.30-08
Np-239	6.42-06
Zr-95	2.10-08
Rb-90	1.55-08
Rb-89	2.55-08

(Historical Information)

TABLE 12.2-83 (Cont)

Isotope	Specific Activity (Ci/cc)(2)(3)
Cr-51	3.58-08
Co-58	5.37-07
Co-60	1.08-07
Fe-59	6.30-08
La-142	1.68-08
La-141	3.95-08
La-140	1.98-06
Te-129	1.03-08
Nb-95	2.70-08
Y-93	1.01-08
Y-92	1.52-07
Y-91	3.16-07
Y-91m	1.87-07
Y-90	1.56-07
Total	7.54-05

(1) Design basis:

a. Input activity is the spent resin tank output activity (Table 12.2-82)

b. No decay in spent resin tank

c. Density = 1.0 g/cc

(2) 2.50-07 = 2.50 x 10^{-7}

(3) Isotopes with specific activity less than 1.00-08 Ci/cc are not listed

(Historical Information)

TABLE 12.2-84

RADWASTE TANK VENT FILTER SHIELDING DESIGN

SOURCE TERMS[1]

Isotope	Activity (Ci)[2]
Br-83	6.96-04
Br-84	4.75-05
Br-85	1.21-05
I-131	1.54-01
I-132	5.54-03
I-133	1.15-01
I-134	9.58-04
I-135	2.74-02
Total	3.04-01

(1) Design basis:

a. Based on operating experience

b. Source activity is normalized to yield a contact dose rate of 1.0 rem/h

(2) 6.96-04 = 6.96×10^{-4}

(Historical Information)

TABLE 12.2-85

WASTE SLUDGE PHASE SEPARATOR SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br--85	5.41+01
I--131	1.04+01
I--132	4.22+01
I--133	3.75+01
I--134	5.03+00
I--135	2.12+01
Sr--89	1.24+01
Sr--90	4.65-01
Sr--91	1.63+01
Sr--92	7.22+00
Zr-95	1.00+01
Nb-95	1.03+01
Mo-99	1.50+01
Tc-99m	5.26+01
Tc-101	7.90-01
Te-132	3.36+01
Cs-136	1.29-01
Cs-137	6.58-01
Cs-138	2.47+00
Ba-139	5.35+00
Ba-140	1.25+01
Ba-141	1.23+00
Ba-142	7.54-01
Ce-141	6.32+00
Ce-143	2.14-01
Pr-143	4.91+00
Np-239	1.56+02
Mn-56	6.23-01
Co-58	8.63-01
W-187	2.76-01

(Historical Information)

TABLE 12.2-85 (Cont)

Isotope	Activity (Ci)[2][3]
La-142	7.54-01
La-141	1.24+00
La-140	5.09+00
Y-92	7.28+00
Y-91	8.20+00
Y-91m	9.62+00
TOTAL	5.06+02

(1) Design basis:

a) Source volume = 409 gallons

b) Two batches of backwash from waste filter (Table 12.2-61) - first batch with 24 hours decay, second batch with no decay, one batch of FPCC filter demineralizer backwash (Table 12.2-30), and two batches of floor drain filter backwash (Table 12.2-48)

(2) $1.00 + 01 = 1.00 \times 10^{1}$

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-86

WASTE SLUDGE DISCHARGE MIXING PUMP SHIELDING

DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)[2][3]
Br--85	3.50-06
I--131	6.72-06
I--132	2.73-05
I--133	2.42-05
I--134	3.25-06
I--135	1.37-05
Sr-89	8.01-06
Sr-90	3.00-07
Sr--91	1.05-05
Sr--92	4.66-06
Zr--95	6.46-06
Nb--95	6.65-06
Mo--99	9.69-06
Tc-99m	3.40-05
Tc-101	5.10-07
Te-132	2.17-05
Cs-137	4.25-07
Cs-138	1.60-06
Ba-139	3.46-06
Ba-140	8.08-06
Ba-141	7.95-07
Ba-142	4.87-07
Ce-141	4.08-06
Ce-143	1.38-07
Pr-143	3.17-06
Np-239	1.01-04
Mn--56	4.02-07
Co-58	5.58-07

(Historical Information)

TABLE 12.2-86 (Cont)

Isotope	Specific Activity (Ci/cc)(2)(3)
W--187	1.78-07
La-142	4.87-07
La-141	8.01-07
La-140	3.29-06
Y-92	4.70-06
Y-91	5.30-06
Y-91m	6.21-06
Total	3.27-04

---

(1) Design basis:
   a) Source volume = 409 gallons
   b) Input activity of waste sludge activity in waste sludge phase separator without decay (Table 12.2-85)

(2) $3.50\text{-}06 = 3.50 \times 10^{-6}$

(3) Isotopes with specific activity less than 1.00-07 Ci/cc are not listed

(Historical Information)

TABLE 12.2-87

PIPING BETWEEN THE AFTERCONDENSER AND RECOMBINER SYSTEMS

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)[2][3]
N-16	1.08-05[4][5]
O-19	6.70-08
Kr-83m	1.14-08
Kr-85m	2.08-08
Kr-87	6.44-08
Kr-88	6.44-08
Kr-89	4.14-07
Kr-90	6.97-07
Kr-91	4.20-07
Kr-92	4.65-08
Rb-92	1.37-08
Xe-133	2.74-08
Xe-135	7.38-08
Xe-137	4.82-07
Xe-138	2.99-07
Xe-139	7.47-07
Xe-140	4.00-07
Cs-140	1.36-08
Total	1.56-05

TABLE 12.2-87 (Cont)

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(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 and 12.2-10

b. Effective off-gas density = 9.82-04 g/cc

c. The following transit time is assumed: 4.66 seconds from the aftercondenser to the drain pot in the recombiner system

d. It is assumed that 20 percent of the N-16 in steam in both the moisture separator and the main condensers is washed out

(2) 1.08-05 = 1.08×10^{-5}

(3) Isotopes with activities less than 1.00-08 are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-88

OFF-GAS RECOMBINER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
N-16	9.62+00[4][5]
O-19	7.77-01
Kr-83m	1.44-02
Kr-85m	2.63-02
Kr-87	8.45-02
Kr-88	8.46-02
Kr-89	5.16-01
Kr-90	8.17-01
Kr-91	4.12-01
Kr-92	2.51-02
Xe-133	3.47-02
Xe-135m	1.09-01
Xe-135	9.33-02
Xe-137	6.03-01
Xe-138	3.77-01
Xe-139	8.91-01
Xe-140	5.90-01
Total	1.51+01

(1) Design basis:

a. Source volume = 113 ft^3

b. Residence time = 0.85 s

c. Input activity is the outlet of SJAE (Table 12.2-87) with 5 s decay

d. 75 scfm air flow

(2) 7.77-01 = 7.77×10^{-1}

(3) Isotopes with a less than 1.00-02 Ci are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-89

OFF-GAS RECOMBINER COOLER CONDENSER SHIELDING

DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
N-16	5.53+00(4)(5)
O-19	4.54-01
Kr-85m	1.54-02
Kr-87	4.97-02
Kr-88	4.98-02
Kr-89	3.03-01
Kr-90	4.78-01
Kr-91	2.38-01
Kr-92	1.39-02
Xe-133	2.04-02
Xe-135m	6.39-02
Xe-135	5.49-02
Xe-137	3.54-01
Xe-138	2.22-01
Xe-139	5.22-01
Xe-140	3.43-01
Total	8.71+00

(1) Design basis:

a. Source free volume = 52.50 ft^3

b. Residence time = 0.5 s

c. Input activity is the outlet activity of recombiner (Table 12.2-88) with no decay

(2) 4.54-01 = 4.54×10^{-1}

(3) Isotopes with an activity less than 1.00-02 Ci are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-90

OFF-GAS HOLDUP DECAY PIPE SHIELDING DESIGN SOURCE TERMS[(1)]

Isotope	Specific Inlet Activity (Ci/cc)[(2)(3)]	Specific Outlet Activity (Ci/cc)
N-16	1.50-03[(4)(5)]	2.53-05[(4)(5)]
O-19	1.25-04	8.20-06
Kr-83m	2.35-06	2.21-06
Kr-85m	4.29-06	4.18-06
Kr-87	1.38-05	1.26-05
Kr-88	1.38-05	1.33-05
Kr-89	8.41-05	2.65-05
Kr-90	1.32-04	9.72-06
Kr-91	6.50-05	1.50-06
Xe-133	5.68-06	5.66-06
Xe-135m	1.78-05	1.23-05
Xe-135	1.53-05	1.53-05
Xe-137	9.83-05	3.54-05
Xe-138	6.15-05	4.41-05
Xe-139	1.44-04	1.35-05
Xe-140	9.43-05	3.50-06
Total	2.38-03	2.35-04

(1) Design basis:

a. Input activity is the outlet activity of the cooler condenser (Table 12.2-89) with no decay

b. Residence time = 10 minutes

c. Average flow = 0.2544 ft^3/s

(2) 1.50-03 = 1.50×10^{-3}

(3) Isotopes with specific activity less than 1.00-06 Ci/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-91

OFF-GAS CHARCOAL BED COOLER CONDENSER

SHIELDING DESIGN SOURCE TERM[1]

Isotope	Activity (Ci)[2][3]
Kr-83m	1.50-01
Kr-85m	2.84-01
Kr-87	8.57-01
Kr-88	8.99-01
Kr-89	1.79+00
Kr-90	6.54-01
Kr-91	1.00-01
Kr-92	1.67-03
Xe-133m	1.40-02
Xe-133	3.84-01
Xe-135m	8.31-01
Xe-135	1.03+00
Xe-137	2.39+00
Xe-138	2.99+00
Xe-139	9.14-01
Xe-140	2.35-01
N-13	4.78-01[4][5]
N-16	1.67+00[4][5]
O-19	5.49-01
F-18	6.34-02
Rb-91	1.15-02
Rb-90	5.57-02
Rb-89	6.51-02
Rb-88	2.91-02
Cs-140	2.54-02

(Historical Information)

TABLE 12.2-91 (Cont)

Isotope	Activity (Ci)(2)(3)
Cs-139	4.61-02
Cs-138	6.15-02
	1.62+01

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(1) Design basis:

a. Source volume = 165 ft^3

b. Residence time = 9.3 s

c. Input activity is the outlet activity of holdup delay pipe (Table 12.2-90) with no decay

(2) $1.50\text{-}01 = 1.50 \times 10^{-1}$

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-92

OFF-GAS GUARD BED SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Kr-83m	4.57+00
Kr-85m	8.72+00
Kr-87	2.60+01
Kr-88	2.76+01
Kr-89	3.43+01
Kr-90	3.29+00
Xe-133m	7.65+00
Xe-133	2.11+02
Xe-135m	1.17+02
Xe-135	5.45+02
Xe-137	8.60+01
Xe-138	4.66+02
Xe-139	6.00+00
Ba-140	1.70+00
La-140	1.70+00
Cs-139	3.63+00
Ba-139	6.08+00
Cs-138	1.64+01
Cs-137	9.63+00
Rb-90	1.34+00
Sr-90	2.88+00
Y-90	2.88+00
Rb-89	8.31+00
Rb-88	4.30+00
Sr-89	1.21+01
Total	1.62+03

(Historical Information)

TABLE 12.2-92 (Cont)

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(1) Design basis:

a. Source volume = 17 ft^3

b. Input activity is the outlet activity of cooler condenser (Table 12.2-89) with no decay

c. Transient times: krypton isotopes - 0.081 h, xenon isotopes - 1.44 h

d. 100 percent daughter product accumulation for 40 years.

(2) 2.60+01 = 2.60×10^1

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-93

OFF-GAS FIRST CHARCOAL BED SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Kr-83m	1.24+02
Kr-85m	3.57+02
Kr-87	5.31+02
Kr-89	1.82+01
Xe-131m	2.04+01
Xe-133m	2.71+02
Xe-133	9.88+03
Xe-135	4.74+03
Xe-138	1.57+01
Rb-89	2.24+01
Sr-89	2.42+01
Rb-88	9.43+02
Total	1.70+04

(1) Design basis:

a. Source volume = 1000 ft^3

b. Input activity is the outlet activity of guard bed (Table 12.2-92) with no decay

c. Transient times: krypton isotopes - 4.74 h, xenon isotopes - 84 h

d. 100 percent daughter product accumulation for 40 years

(2) 1.24+02 = 1.24×10^2

(3) Isotopes with an activity less than 1.00+01 Ci are not listed

(Historical Information)

TABLE 12.2-94

OFF-GAS SECOND CHARCOAL BED SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Kr-83m	8.46+00
Kr-85m	9.59+01
Kr-85	1.70+00
Kr-87	1.43+01
Kr-88	1.36+00
Xe-131m	1.37+01
Xe-133m	4.51+01
Xe-133	4.27+03
Xe-135	2.71+00
Cs-138	1.37+00
Sr-89	1.00+00
Rb-88	1.36+02
Total	4.59+03

(1) Design basis:

a. Source volume = 1000 ft^3. (Total for three beds in parallel = 3000 ft^3)

b. Input activity is 1/3 the outlet activity of the first charcoal bed (Table 12.2-93) with no decay

c. Transient times: krypton isotopes - 14.2 h, xenon isotopes - 252 h

d. 100 percent daughter product accumulation for 40 years

(2) $9.59+01 = 9.59 \times 10^1$

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-95

OFF-GAS THIRD CHARCOAL BED SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Kr-85m	1.02+01
Kr-85	1.70+00
Kr-88	4.06+00
Xe-131m	7.38+00
Xe-133m	1.80+00
Xe-133	1.08+03
Rb-88	5.24+00
Total	1.11+03

(1) Design basis:

a. Source volume = 1000 ft^3 (three beds in parallel)

b. Input activity is the outlet activity of the second charcoal bed (Table 12.2-94) with no decay

c. Transient times: krypton isotopes - 14.2 h, xenon isotopes - 252 h

d. 100 percent daughter product accumulation for 40 years

(2) 1.02+01 = 1.02×10^{-1}

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-96

OFF-GAS FOURTH CHARCOAL BED SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Kr-85m	1.09+00
Kr-85	1.70+00
Kr-88	1.21-01
Xe-131m	3.99+00
Xe-133	2.72+02
Ba-139	1.27-01
Cs-138	3.42-01
Cs-137	2.01-01
Rb-89	1.73-01
Sr-89	2.51-01
Rb-88	2.10-01
Total	2.81+02

---

(1) Design basis:

a. Source volume = 1000 ft^3 (three beds in parallel)

b. Input activity is the outlet activity of the third charcoal bed (Table 12.2-95) with no decay

c. Transient times: krypton isotopes - 14.2 h, xenon isotopes - 252 h

d. 100 percent daughter product accumulation for 40 years

(2) 1.21-01 = 1.21×10^{-1}

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-97

OFF-GAS HEPA FILTER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Cs-140	1.57-03
Ba-140	1.18-02
La-140	1.18-02
Cs-139	2.51-02
Ba-139	4.22-02
Cs-138	1.14-01
Cs-137	6.69-02
Sr-90	2.08-02
Sr-91	5.31-03
Y-91	5.36-03
Y-91m	3.16-03
Rb-90	9.23-03
Y-90	2.00-02
Rb-89	5.76-02
Sr-89	8.37-02
Rb-88	2.98-02
Total	5.10-01

(1) Design basis:

a. Source volume = 0.44 ft^3

b. Input activity: 0.0276 of particulates from outlet of delay holdup pipe and daughter products from outlet activity of the fourth charcoal beds that decayed for 1.0 s, accumulated for 40 years

c. Filter efficiency of 100 percent for all particulates

(2) 1.57-03 = 1.57×10^{-3}

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

TABLE 12.2-98

CENTRIFUGE FEED TANK SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
I-131	1.73+03
I-132	2.14+03
I-133	4.68+02
I-135	1.29+02
Sr-89	1.91+03
Sr-90	2.16+02
Sr-91	1.19+02
Sr-92	2.35+01
Zr-95	2.58+01
Nb-95	3.36+01
Mo-99	7.37+02
Tc-99m	9.23+02
Ru-103	1.01+01
Te-129m	1.96+01
Te-132	2.09+03
Cs-134	1.40+02
Cs-136	2.68+01
Cs-137	2.16+02
Ba-140	2.05+03
Ce-141	5.03+01
Ce-144	2.99+01
Pr-143	1.01+01
Np-239	6.25+03
Cr-51	4.32+01
Co-58	6.61+02
Co-60	8.91+01
Ag-110m	1.00+01
Pr-144	2.99+01
La-140	2.20+03

Isotope	Activity (Ci)(2)(3)
Te-129	1.25+01
Y-92	5.09+01
Y-91	3.00+02
Y-91m	7.63+01
Y-90	2.09+02
Total	2.30+04

(1) Design basis:

a. Maximum of 30 weight percent solids

b. Slurry volume = 435 gallons

c. RWCU phase separators, Table 12.2-56, source term with 4 hours decay

(2) 1.73+03 = 1.73 x 10^3

(3) Isotopes with an activity less than 1.00+01 Ci are not listed

(Historical Information)

TABLE 12.2-99

CENTRIFUGE FEED TANK RECIRCULATION PUMP

SHIELDING DESIGN SOURCE TERMS (1)

NOTE: Centrifuge Feed Tank Recirculation Pump has been abandoned in place.

Isotope	Specific Activity (Ci/g) (2) (3)
I-131	1.05-03
I-132	1.30-03
I-133	2.83-04
I-135	7.80-05
Sr-89	1.16-03
Sr-90	1.31-04
Sr-91	7.20-05
Sr-92	1.42-05
Zr-95	1.56-05
Nb-95	2.04-05
Mo-99	4.66-04
Tc-99m	5.60-04
Te-129m	1.18-05
Te-132	1.27-03
Ce-134	8.48-05
Cs-136	1.62-05
Cs-137	1.31-04
Ba-140	1.24-03
Ce-141	3.04-05
Ce-144	1.81-05
Np-239	3.78-03
Cr-51	2.62-05
Co-58	4.00-04
Co-60	5.40-05
Pr-144	1.81-05
La-140	1.33-03
Y-92	3.08-05
Y-91	1.82-04

(Historical Information)

TABLE 12.2-99 (Cont)

Isotope	Specific Activity (Ci/g)[2][3]
Y-91m	4.62-05
Y-90	1.27-04
Total	1.40-02

(1) Design basis:

a. Maximum of 30 weight percent solids for direct feed to extruder evaporator

b. Slurry volume = 435 gallons, Table 12.2-98

c. Density = 1.0 g/cc

(2) 1.05-03 = 1.05×10^{-3}

(3) Isotopes with specific activity less than 1.00-05 Ci/g are not listed

(Historical Information)

TABLE 12.2-100

SRW CENTRIFUGE SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
I-131	6.46+01
I-132	7.90+01
I-133	1.55+01
I-135	3.23+00
Sr-89	7.22+01
Sr-90	8.21+00
Sr-91	3.39+00
Sr-92	3.21+01
Zr-95	9.77+01
Nb-95	1.28+00
Mo-99	2.69+01
Tc-99m	3.09+01
Ru-103	3.82-01
Te-129m	7.41-01
Te-132	7.68+01
Cs-134	5.32+00
Cs-136	1.01+00
Cs-137	8.21+00
Ba-140	7.71+01
Ce-141	1.90+00
Ce-144	1.14+00
Pr-143	3.80-01
Nd-147	1.03-01
Np-239	2.26+02
Cr-51	1.63+00
Mn-54	2.54-01
Co-58	2.51+01
Co-60	3.39+00

(Historical Information)

TABLE 12.2-100 (Cont)

Isotope	Activity (Ci)(2)(3)
Fe-59	3.41-01
Ag-110m	3.75-01
W-187	1.28-01
Pr-144	1.14+00
La-141	1.11-01
La-140	8.32+01
Te-129	4.75-01
Rh-103m	3.74-01
Y-92	1.17+00
Y-91	1.14+01
Y-91m	2.19+00
Y-90	7.98+00
Total	9.73+02

(1) Design basis:

a. Volume = 15 gallons per batch

b. Maximum of 33 weight percent solid

c. Total of 8 hours decay (from output of RWCU phase separator, Table 12.2-56

(2) $6.46+01 = 6.46 \times 10^{1}$

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-101

SRW EXTRUDER EVAPORATOR SHIELDING DESIGN SOURCE TERMS[1]

NOTE: SRW Extruder Evaporator has been abandoned in place

Isotope	Specific Activity (Ci/g)[2][3]
I-131	3.44-03
I-132	4.21-03
I-133	8.26-04
I-135	1.72-04
Sr-89	3.85-03
Sr-90	4.37-04
Sr-91	1.80-04
Sr-92	1.71-05
Zr-95	5.20-05
Nb-95	6.78-05
Mo-99	1.43-03
Tc-99m	1.65-03
Ru-103	2.03-05
Te-129m	3.95-05
Te-132	4.09-03
Cs-134	2.84-04
Cs-136	5.39-05
Cs-137	4.37-04
Ba-140	4.11-03
Ce-141	1.01-04
Ce-144	6.06-05
Pr-143	2.03-05
Np-239	1.21-02
Cr-51	8.71-05
Mn-54	1.36-05
Co-58	1.34-03
Co-60	1.80-04
Fe-59	1.82-05
Ag-110m	2.00-05

(Historical Information)

TABLE 12.2-101 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)
Pr-144	6.06-05
La-140	4.44-03
Te-129	2.53-05
Rh-103m	1.99-05
Y-92	6.22-05
Y-91	6.06-04
Y-91m	1.16-04
Y-90	4.25-04
Total	4.51-02

(1) Design basis:

a. 100 percent dry RWCU resin

b. RWCU separator, Table 12.2-56, source with 8 hours decay

(2) 3.44-03 = 3.44×10^{-3}

(3) Isotopes with specific activity less than 1.00-05 Ci/g not listed

(Historical Information)

TABLE 12.2-102

VOLUME REDUCTION RECIRCULATION PUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
I-131	2.28-04
I-132	8.98-06
I-133	1.66-05
I-135	3.66-06
Sr-89	1.24-05
Sr-90	1.22-06
Sr-91	2.78-07
Zr-95	1.67-07
Nb-95	2.05-07
Mo-99	2.67-06
Tc-99m	2.85-06
Te-129m	1.27-07
Te-132	8.51-06
Cs-135	8.10-07
Cs-136	1.84-07
Cs-137	1.24-06
Ba-140	1.57-05
Ce-141	5.73-07
Ce-144	1.76-07
Np-239	1.97-05
Cr-51	2.86-07
Co-58	4.07-06
Co-60	5.15-07
Pr-144	1.76-07
La-140	1.77-05

(**Historical Information**)

TABLE 12.2-102 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)
Y-91	2.75-06
Y-91m	1.79-07
Y-90	1.23-06
Total	3.35-04

(1) Design basis:

a. 50 weight percent solids

b. Output of concentrate waste tank, Table 12.2-78, without decay

c. Density = 1.0 g/cc

(2) 2.28-04 = 2.28×10^{-4}

(3) Isotopes with specific activity less than 1.00-07 Ci/g are not listed

(Historical Information)

TABLE 12.2-103

CRYSTALLIZER BOTTOM TANK
SHIELDING DESIGN SOURCE TERMS[1]

NOTE: Crystallizer Bottom Tank has been abandoned in place

Isotope	Activity (Ci)[2][3]
I-131	2.60+03
I-132	1.02+02
I-133	1.89+02
I-135	4.16+01
Sr-89	1.41+02
Sr-90	1.39+01
Sr-91	3.16+00
Zr-95	1.89+00
Nb-95	2.33+00
Mo-99	3.03+01
Tc-99m	3.24+01
Te-129m	1.45+00
Te-132	9.67+01
Cs-134	9.20+00
Cs-136	2.09+00
Cs-137	1.41+01
Ba-140	1.79+02
Ce-141	6.51+00
Ce-144	1.99+00
Np-239	2.24+02
Cr-51	3.25+00
Co-58	4.63+01
Co-60	5.85+00
Pr-144	1.99+00
La-140	2.01+02
Y-92	1.13+00
Y-91	3.13+01

(Historical Information)

TABLE 12.2-103 (Cont)

Isotope	Activity (Ci)[(2)(3)]
Y-91m	2.03+00
Y-90	1.39+01
Total	4.00+03

---

(1) Design basis:

a. Tank volume = 3000 gallons

b. 50 weight percent solids

c. Output of concentrate waste tank, Table 12.2-78, without decay

(2) 2.60+03 = 2.60×10^3

(3) Isotopes with an activity less than 1.00+00 Ci are not listed

(Historical Information)

TABLE 12.2-104

VOLUME REDUCTION SYSTEM (VRS) ENTRAINMENT SEPARATOR SHIELDING DESIGN SOURCE TERMS[1]

NOTE: VRS Entrainment Separator has been abandoned in place

Isotope	Specific Activity (Ci/g)[2][3]
I-131	1.15-06
I-132	4.49-08
I-133	8.31-08
I-135	1.83-08
Sr-89	6.23-08
Sr-90	6.11-09
Sr-91	1.39-09
Nb-95	1.02-09
Mo-99	1.34-08
Tc-99m	1.43-08
Te-132	4.26-08
Cs-134	4.05-09
Cs-137	6.22-09
Ba-140	7.87-08
Ce-141	2.86-09
Nb-239	9.87-08
Cr-51	1.43-09
Co-58	2.04-08
Co-60	2.57-09
La-140	8.90-08
Y-91	1.37-08
Y-90	6.14-09
Total	1.76-06

(Historical Information)

TABLE 12.2-104 (Cont)

Isotope	Specific Activity (Ci/g)(2)(3)

---

(1) Design basis:

a. Carryover factor of 0.01 for all isotopes in the VRS vapor body

b. Output of concentrate waste tank, Table 12.2-78, at 25 weight percent solids

c. Density = 1.0 g/cc

(2) 1.15-06 = 1.15×10^{-6}

(3) Isotopes with specific activity less than 1.00-09 Ci/g are not listed

(Historical Information)

TABLE 12.2-105

VRS CONDENSER COOLER

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/g)[2][3]
I-131	1.15-07
I-132	4.49-09
I-133	8.31-09
I-135	1.83-09
Sr-89	6.23-09
Sr-90	6.11-10
Sr-91	1.39-10
Nb-95	1.02-10
Mo-99	1.34-09
Tc-99m	1.43-09
Te-132	4.26-09
Cs-134	4.05-10
Cs-137	6.22-10
Ba-140	7.87-09
Ce-141	2.86-10
Np-239	9.87-09
Cr-51	1.43-10
Co-58	2.04-09
Co-60	2.57-10
La-140	8.90-09
Y-91	1.37-09
Y-90	6.14-10
Total	1.76-07

(Historical Information)

TABLE 12.2-105 (Cont)

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(1) Design basis:

a. Carryover factor of 0.01 for all isotopes in the VRS vapor body

b. Carryover factor of 0.1 for all isotopes in the VRS entrainment separator

c. Output of concentrate waste tank Table 12.2-78, at 25 weight percent solid

d. Density = 1.0 g/cc

(2) 1.15-07 = 1.15×10^{-7}

(3) Isotopes with specific activity less than 1.00-10 Ci/g are not listed

(Historical Information)

TABLE 12.2-106

RADWASTE BUILDING EXHAUST SYSTEM FILTER

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
I-131	1.28-03
I-132	1.41-04
I-133	9.36-04
I-134	1.06-04
I-135	4.42-04
Sb-124	9.85-05
Cs-134	1.05-02
Cs-137	1.98-02
Cr-51	4.68-04
Mn-54	1.45-02
Co-58	3.25-04
Co-60	3.32-02
Fe-59	3.26-04
Zn-65	1.01-03
Zr-95	1.21-03
Nb-95	1.16-03
Nb-95m	2.41-05
Total	8.70-02

(1) Design basis:
- a. Based on release rates given by NUREG-0016
- b. Iodine concentrations are normalized to correspond to a 500,000 μCi/s noble gas release at 30 minutes decay
- c. Particulate release rates are increased by a factor of five
- d. 100% efficiency for particulates and particulate iodines and accumulated for one year

(2) 1.28-03 = 1.28×10^{-3}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-107

AUXILIARY BUILDING SHIELDING DESIGN RADIATION SOURCE DESCRIPTION
(RADWASTE AREA - ELEVATION 54'-0")

Room Number	Dominant Radiation Source	Identification	Source Geometry	Effective Source Density (g/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-107 (Cont)

Security Related Information
Table withheld Under 10 CFR 2.390

Room Number	Dominant Radiation Source	Identification	Source Geometry	Effective Source Density (g/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table
			height			
**3147	Decontamination solution concentration waste pump	OOP328	10' (2"-diam pipe)	1.0	0.154	12.2-70

** Abandoned in place

(Historical Information)

TABLE 12.2-107 (Cont)

Room Number	Dominant Radiation Source	Identification	Source Geometry	Effective Source Density (g/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

** Abandoned in place

(Historical Information)

TABLE 12.2-107 (Cont)

Room Number	Dominant Radiation Source	Identification	Source Geometry	Effective Source Density (g/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

7

(Historical Information)

TABLE 12.2-107 (Cont)

Room Number	Dominant Radiation Source	Identification	Source Geometry	Effective Source Density (g/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

** Abandoned in place

(Historical Information)

TABLE 12.2-108

MAIN STEAM AND FEEDWATER HEATER SYSTEM SHIELDING DESIGN SOURCE TERMS(1)(2)

Equipment	Transit Time (seconds)	N-16(3) (μCi/g)	Steam Density (g/cc)
Reactor pressure vessel nozzle	0.0	50.0	0.0345
26-inch main steam lines	0.0	50.0	0.0345
Main steam valves (at 28-inch MS lines)	3.46	35.7	0.0345
High pressure turbine inlet	4.15	33.4	0.0330
High pressure turbine outlet	4.20	33.3	0.00744
Moisture separator inlet	5.35	29.8	0.00744
Moisture separator outlet	7.06	25.2	0.00648
Low pressure turbine inlet	7.60	24.3	0.00648
Feedwater heaters			
E-101 (A, B or C)	7.6	24.3	0.000216
E-102 (A, B or C)	7.6	24.3	0.000805
E-103 (A, B or C)	8.3	22.7	0.00238
E-104 (A, B or C)	8.5	22.3	0.00408
E-105 (A, B or C)	5.6	26.5	0.00752
E-106 (A, B or C)	6.4	24.5	0.0129
RFPT 6-inch feedline	9.4	20.1	0.0345
RFPT 10-inch feedline	8.6	21.7	0.00623
RFPT 60-inch outlet line	8.6	21.7	0.000101
Steam sealing evaporator 8 inch feedline	3.5	35.6	0.00186

Design basis:

(1) Based on 50 μCi/g of N-16 at RPV steam outlet nozzle

(2) Main steam flow = 14.8×10^6 lb/h at 105 percent of rated power (valves wide open condition)

(3) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5. For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-109

FEEDWATER HEATER NO. 6 (A, B, OR C) DRAIN

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	1.27-03[4]
N-16	3.24+00[4]
Ba-143	1.74-03
Cs-142	1.22-02
Ba-142	1.25-03
Cs-141	3.54-02
Ba-141	1.06-03
Cs-140	6.79-02
Cs-139	8.88-03
Cs-138	1.88-03
Rb-94	4.37-03
Rb-93	3.56-02
Rb-92	1.93-01
Rb-91	6.20-02
Rb-90	2.79-02
Rb-89	2.72-03
Sr-89	1.55-03
Br-83	1.50-03
Br-84	2.67-03
Br-85	1.64-03
I-131	1.28-03
I-132	1.18-02
I-133	9.10-03
I-134	2.35-02
I-135	1.28-02

(Historical Information)

TABLE 12.2-109 (Cont)

Isotope	Specific Activity (μCi/g)[2][3]
Tc-99m	1.39-03
Np-239	1.18-03
Total	3.77+00

---

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.7 through 12.2-11

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water. Assume that 100 percent of halogens and particulates are washed out

(2) $1.27\text{-}3 = 1.27 \times 10^{-3}$

(3) Isotopes with specific activities less than 1.00-03 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5. For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-110

FEEDWATER HEATER NO. 5 (A, B, OR C) DRAIN

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (μCi/g)(2)(3)
N-13	1.18-03(4)(5)
N-16	7.63-01(4)(5)
Cs-142	3.53-03
Ba-142	1.19-03
Cs-141	1.03-02
Ba-141	1.53-03
Cs-140	3.20-02
Cs-139	6.80-03
Cs-138	1.67-03
Rb-94	1.20-03
Rb-93	8.53-03
Rb-92	4.44-02
Rb-91	3.11-02
Rb-90	1.82-02
Rb-89	2.12-03
Br-83	1.49-03
Br-84	2.60-03
Br-85	1.30-03
I-131	1.28-03
I-132	1.17-02
I-133	9.08-03
I-134	2.31-02
I-135	1.28-02
Tc-99m	1.39-03
Np-239	1.18-03
Total	9.99-01

(Historical Information)

TABLE 12.2-110 (Cont)

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(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11, and 12.2-109

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water, plus 100 percent of the halogens and particulates

(2) 1.18-3 = 1.18×10^{-3}

(3) Isotopes with specific activities less than 1.00-03 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-111

FEEDWATER HEATER NO. 4 (A, B, OR C) DRAIN

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	1.14-03[4][5]
N-16	3.62-01[4][5]
Ba-143	2.04-04
Cs-142	9.15-04
Ba-142	1.13-03
Cs-141	5.58-03
Ba-141	1.59-03
Cs-140	2.43-02
Cs-139	6.85-03
Ba-139	8.97-04
Cs-138	1.69-03
Rb-94	3.52-04
Sr-94	4.45-04
Rb-93	3.62-03
Sr-93	8.37-04
Rb-92	2.06-02
Sr-92	7.60-04
Rb-91	2.36-02
Sr-91	4.14-04
Rb-90	1.66-02
Rb-89	2.19-03
Rb-88	3.00-04
Br-83	1.48-03
Br-84	2.56-03
Br-85	1.10-03
I-131	1.28-03
I-132	1.16-02
I-133	9.08-03

(Historical Information)

TABLE 12.2-111 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
I-134	2.29-02
I-135	1.28-02
Mo-99	1.07-04
Tc-99m	1.39-03
Tc-101	6.29-04
Te-132	2.41-04
Np-239	1.18-03
Total	5.43-01

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11, and 12.2-110. (FWH No. 5 drains into FWH No. 4)

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water, plus 100 percent of the particulates and halogens

(2) $1.14\text{-}3 = 1.14 \times 10^{-3}$

(3) Isotopes with specific activities less than 1.00-04 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-112

FEEDWATER HEATER NO. 3 (A, B, OR C) DRAIN

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	1.10-03[4][5]
N-16	6.45-01[4][5]
Cs-144	1.46-04
Cs-143	1.70-04
Ba-143	2.97-04
Cs-142	2.55-03
Ba-142	1.09-03
Cs-141	6.73-03
Ba-141	1.52-03
Cs-140	2.14-02
Cs-139	6.64-03
Ba-139	9.11-04
Cs-138	1.66-03
Rb-94	8.74-04
Sr-94	3.66-04
Rb-93	6.52-03
Sr-93	7.62-04
Rb-92	3.47-02
Sr-92	7.44-04
Rb-91	2.08-02
Sr-91	4.16-04
Rb-90	1.51-02
Rb-89	2.15-03
Rb-88	2.96-04
Br-83	1.46-03
Br-84	2.50-03
Br-85	9.83-04
I-131	1.27-03

TABLE 12.2-112 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
I-132	1.15-02
I-133	8.96-03
I-134	2.25-02
I-135	1.26-02
Mo-99	1.06-04
Tc-99m	1.37-03
Tc-101	6.06-04
Te-132	2.38-04
Np-239	1.16-03
Total	8.38-01

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-8, 12.2-9, 12.2-10, and 12.2-111. (FWH No. 4 drains into FWH No. 3)

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water, plus 100 percent of the particulates and halogens

(2) $1.10\text{-}3 = 1.10 \times 10^{-3}$

(3) Isotopes with specific activities less than 1.00-04 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-113

FEEDWATER HEATER NO. 2 (A, B, OR C) DRAIN

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (μCi/g)(2)(3)
N-13	1.03-03(4)(5)
N-16	5.28-01(4)(5)
Cs-144	1.10-04
Cs-143	2.06-04
Ba-143	2.11-04
Cs-142	2.69-03
Ba-142	1.010-03
Cs-141	4.82-03
Ba-141	1.47-03
Cs-140	1.44-02
Cs-139	6.02-03
Ba-139	9.32-04
Cs-138	1.59-03
Rb-94	8.61-04
Sr-94	2.66-04
Rb-93	5.23-03
Sr-93	7.04-04
Rb-92	2.64-02
Sr-92	7.30-04
Rb-91	1.44-02
Sr-91	4.19-04
Rb-90	1.21-02
Rb-89	2.00-03
Rb-88	2.76-04
Br-83	1.43-03
Br-84	2.40-03
Br-85	8.16-04
I-131	1.24-03

(Historical Information)

TABLE 12.2-113 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
I-132	1.12-02
I-133	8.79-03
I-134	2.18-02
I-135	1.24-02
Mo-99	1.04-04
Tc-99m	1.34-03
Tc-101	5.70-04
Te-132	2.33-04
Np-239	1.14-03
Total	6.90-01

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11, and 12.2-112. (FWH No. 3 drains into FWH No. 2)

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water, plus 100 percent of the halogens and particulates

(2) $1.03\text{-}3 = 1.03 \times 10^{-3}$

(3) Isotopes with specific activities less than 1.00-04 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-114

FEEDWATER HEATER NO. 1 (A, B, OR C) DRAIN

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	1.42-03[4][5]
N-16	4.67+00[4][5]
Cs-143	1.87-03
Ba-143	1.82-03
Cs-142	2.42-02
Ba-142	1.20-03
Cs-141	3.71-02
Ba-141	1.00-03
Cs-140	5.56-02
Cs-139	6.94-03
Rb-94	7.72-03
Rb-93	4.62-02
Sr-93	4.91-04
Rb-92	2.32-01
Rb-91	5.22-02
Rb-90	2.20-02
Rb-89	2.10-03
Br-83	1.50-03
Br-84	2.67-03
Br-85	1.66-03
I-131	1.28-03
I-132	1.18-02
I-133	9.10-03
I-134	2.35-02
I-135	1.28-02

(Historical Information)

TABLE 12.2-114 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
Tc-99m	1.39-03
Np-239	1.18-03
Total	5.24+00

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(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11, 12.2-112 and 12.2-113

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water, plus 100 percent of the halogens and particulates

(2) $1.42\text{-}3 = 1.42 \times 10^{-3}$

(3) Isotopes with specific activities less than 1.00-04 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-115

DRAIN COOLER (A, B, OR C)

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	1.03-03[4][5]
N-16	2.98-01[4][5]
Ba-143	1.74-04
Cs-142	4.54-04
Ba-142	1.01-03
Cs-141	4.07-03
Ba-141	1.48-03
Cs-140	1.36-02
Cs-139	5.98-03
Ba-139	9.36-04
Cs-138	1.59-03
Rb-94	2.10-04
Sr-94	2.75-04
Rb-93	2.52-03
Sr-93	7.29-04
Rb-92	1.22-02
Sr-92	7.37-04
Rb-91	1.36-02
Sr-91	4.20-04
Rb-90	1.18-02
Rb-89	1.99-03
Rb-88	2.74-04
Br-83	1.43-03
Br-84	2.40-03
Br-85	7.97-04
I-131	1.24-03
I-132	1.12-02
I-133	8.79-03

(Historical Information)

TABLE 12.2-115 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
I-134	2.17-02
I-135	1.24-02
Mo-99	1.04-04
Tc-99m	1.34-03
Tc-101	5.67-04
Te-132	2.33-04
Np-239	1.14-03
Total	4.37-01

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11, and 12.2-113 (FWH No. 2 drains into the drain cooler)

b. Assume that 20 percent of the nitrogens entering the feedwater heater are washed out into the drain water, plus 100 percent of the halogens and particulates

(2) $1.03\text{-}3 = 1.03 \times 10^{-3}$

(3) Isotopes with specific activities less than 1.00-04 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-116

MAIN CONDENSER HOTWELL SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (Ci/cc)(2)(3)
N-13	7.27-10(4)(5)
N-16	4.54-09(4)(5)
La-143	1.81-11
Ba-142	5.47-10
La-142	4.38-11
Cs-141	2.85-10
Ba-141	9.36-10
La-141	2.88-11
Cs-140	5.16-09
Ba-140	3.48-11
Cs-139	2.70-09
Ba-139	7.76-10
Cs-138	9.76-10
Sr-94	6.30-11
Y-94	3.67-11
Rb-93	2.17-11
Sr-93	3.71-10
Rb-92	9.69-11
Sr-92	5.35-10
Y-92	1.73-11
Rb-91	2.52-09
Sr-91	3.19-10
Y-91m	2.31-11
Rb-90	3.40-09
Rb-89	1.00-09
Sr-89	1.13-11
Rb-88	1.42-10
Mn-56	3.45-11
Br-83	1.04-09

(Historical Information)

TABLE 12.2-116 (Cont)

Isotope	Specific Activity (Ci/cc)[(2)(3)]
Br-84	1.58-09
Br-85	2.70-10
I-131	9.30-10
I-132	8.12-09
I-133	6.56-09
I-134	1.50-08
I-135	9.15-09
Mo-99	7.74-11
Tc-99m	9.90-10
Tc-101	3.26-10
Te-132	1.74-10
Np-239	8.51-10
Total	7.33-08

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11

b. Source volume = 4050 ft^3, corresponds to a 3-minute minimum holdup time

c. Based on inflows to the hotwell from the feedwater heater system and the main condenser

d. Assumes a steam transit time of 7.64 seconds

e. Water density = 0.98 g/cc

(2) 7.267-10 = 7.267×10^{-10}

(3) Isotopes with a specific activity less than 1.00-11 Ci/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-117

PRIMARY CONDENSATE PUMP SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	7.22-04[4][5]
N-16	9.01-04[4][5]
La-143	1.81-05
Ba-142	5.45-04
La-142	4.55-05
Cs-141	1.78-04
Ba-141	9.41-04
La-141	2.99-05
Cs-140	4.38-03
Ba-140	3.53-05
Cs-139	2.68-03
Ba-139	7.91-04
Cs-138	9.83-04
Sr-94	5.53-05
Y-94	3.74-05
Sr-93	3.68-04
Rb-92	1.09-05
Sr-92	5.41-04
Y-92	1.80-05
Rb-91	2.18-03
Sr-91	3.24-04
Y-91m	2.40-05
Rb-90	3.22-03
Rb-89	1.00-03
Sr-89	1.15-05
Rb-88	1.42-04
Mn-56	3.49-05
Zn-69m	2.18-02

(Historical Information)

TABLE 12.2-117 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
Br-83	1.05-03
Br-84	1.59-03
Br-85	2.56-04
I-131	9.43-04
I-132	8.22-03
I-133	6.65-03
I-134	1.52-02
I-135	9.26-03
Mo-99	7.84-05
Tc-99m	1.00-03
Tc-101	3.25-04
Te-132	1.76-04
Np-239	8.63-04
Total	8.77-02

---

(1) Design basis:

a. Based on shielding design source terms given in Table 12.2-116

b. Decay time for transit from the main condenser outlet to the pump inlet = 16.8 seconds

(2) 7.22-4 = 7.22×10^{-4}

(3) Isotopes with specific activities less than 1.00-05 μCi/g are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-118

FEEDWATER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (μCi/g)[2][3]
N-13	6.69-04
Ba-142	5.45-05
Cs-141	8.90-05
Ba-141	9.41-05
Cs-140	2.19-03
Cs-139	1.34-03
Ba-139	7.91-05
Cs-138	4.92-04
Sr-93	3.68-05
Sr-92	5.41-05
Rb-91	1.09-03
Sr-91	3.24-05
Rb-90	1.61-03
Rb-89	5.02-04
Rb-88	7.12-05
Br-83	1.05-04
Br-84	1.59-04
Br-85	2.56-05
I-131	9.43-05
I-132	8.22-04
I-133	6.65-04
I-134	1.52-03
I-135	9.26-04
Tc-99m	1.00-04
Tc-101	3.25-05
Te-132	1.77-05
Np-239	8.63-05
Xe-135	4.74-03
Xe-135m	7.14-02

(Historical Information)

TABLE 12.2-118 (Cont)

Isotope	Specific Activity (μCi/g)(2)(3)
Xe-133m	5.97-05
Xe-133	1.04-03
Kr-83m	1.36-03
Total	9.16-02

---

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7 through 12.2-11 and 12.2-117

b. No decay is assumed for the transit between the primary condensate pump and the condensate demineralizer. A 66-second transit time through the condensate demineralizer is taken into account for the nitrogens

c. DF = 2 for Cs and Rb; DF = 10 for all others, except nitrogens where the DF = 0

d. Noble gases are emitted by particulate parents that have accumulated on the filter demineralizer resins prior to regeneration

(2) 6.69-04 = 6.69×10^{-4}

(3) Isotopes with specific activities less than 1.00-05 μCi/g are not listed

(Historical Information)

TABLE 12.2-119

CONDENSATE DEMINERALIZER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Ce-144	1.91-01
Pr-144	1.91-01
Ba-142	1.63-01
La-142	2.77-01
Ba-141	4.63-01
La-141	6.52-01
Ce-141	6.03-01
Cs-140	1.31-01
Ba-140	1.71+01
La-140	1.70+01
Cs-139	6.93-01
Ba-139	2.48+00
Cs-138	8.61-01
Cs-137	1.41+00
Zr-95	1.76-01
Nb-95	2.25-01
Y-93	1.66-01
Sr-92	2.39+00
Y-92	2.50+00
Sr-91	5.19+00
Y-91	2.64+00
Y-91m	3.09+00
Rb-90	2.55-01
Sr-90	1.35+00
Y-90	1.27+00
Rb-89	4.20-01
Sr-89	1.32+01
Rb-88	6.90-02
Cr-51	3.09-01

TABLE 12.2-119 (Cont)

Isotope	Activity (Ci)[2][3]
Mn-56	1.47-01
Co-58	4.51+00
Co-60	5.89-01
Fe-59	5.32-01
Br-83	4.12+00
Br-84	1.38+00
I-131	2.96+02
I-132	5.27+01
I-133	2.20+02
I-134	2.14+01
I-135	1.01+02
Mo-99	8.53+00
Tc-99m	1.73+01
Tc-101	1.23-01
Te-129m	1.36-01
Te-132	2.24+01
Cs-134	9.34-01
Cs-136	2.01-01
Np-239	7.93+01
Total	9.07+02

(1) Design basis:

a. Based on shielding design source terms given in Table 12.2-116

b. Flow through one condensate demineralizer = 5000 gpm, assume water density = 1 g/cc

(Historical Information)

TABLE 12.2-119 (Cont)

c. Volume = 1045 ft^3

d. Accumulation time = 60 days; 100 percent efficiency for all particulates, except nitrogens for which DF=0

e. No decay is assumed for transit between the primary condensate pump and the filter demineralizer

(2) 1.91-01 = 1.91×10^{-1}

(3) Isotopes with an activity less than 1.00-02 Ci are not listed

(Historical Information)

TABLE 12.2-120

RESIN SEPARATION AND CATION REGENERATION VESSEL

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Ce-144	1.91-01
Pr-144	1.91-01
Ba-142	1.63-01
La-142	2.77-01
Ba-141	4.63-01
La-141	6.52-01
Ce-141	6.03-01
Cs-140	1.31-01
Ba-140	1.71+01
La-140	1.70+01
Cs-139	6.93-01
Ba-139	2.48+00
Cs-138	8.61-01
Cs-137	1.41+00
Zr-95	1.76-01
Nb-95	2.25-01
Y-93	1.66-01
Sr-92	2.39+00
Y-92	2.50+00
Sr-91	5.19+00
Y-91	2.64-00
Y-91m	3.09+00
Rb-90	2.55-01
Sr-90	1.35+00
Y-90	1.27+00
Rb-89	4.20-01
Sr-89	1.32+01
Cr-51	3.09-01

TABLE 12.2-120 (Cont)

Isotope	Activity (Ci)(2)(3)
Mn-56	1.47-01
Co-58	4.51+00
Co-60	5.89-01
Fe-59	5.32-01
Br-83	4.12+00
Br-84	1.38+00
I-131	2.96+02
I-132	5.26+01
I-133	2.20+02
I-134	2.41+01
I-135	1.01+02
Mo-99	8.54-00
Tc-99m	1.73+01
Tc-101	1.24-01
Te-129m	1.37-01
Te-132	2.24+01
Cs-134	9.34-01
Cs-136	2.01-01
Np-239	7.93+01
Total	9.07+02

(1) Design basis:

a. Based on shielding design source terms given in Table 12.2-119
b. One backwash from a condensate demineralizer is assumed
c. Source volume = 2181 gal

(2) 1.90781-01 = 1.90781×10^{-1}

(3) Isotopes with an activity less than 1.00-01 Ci are not listed

(Historical Information)

TABLE 12.2-121

ANION REGENERATION VESSEL SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2]
Br-83	4.12+00
Br-84	1.38+00
Br-85	2.09-02
I-131	2.96+02
I-132	5.27+01
I-133	2.20+02
I-134	2.14+01
I-135	1.01+02
Total	6.96+02

(1) Design basis:

a. Based on shielding design source terms given in Table 12.2-119.

b. Source volume = 2181 gal

(2) $4.12+00 = 4.12 \times 10^0$

(Historical Information)

TABLE 12.2-122

ULTRASONIC RESIN CLEANER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Ce-144	1.91-01
Pr-144	1.91-01
Ba-142	1.63-01
La-142	2.77-01
Ba-141	4.63-01
La-141	6.52-01
Ce-141	6.03-01
Cs-140	1.31-01
Ba-140	1.71+01
La-140	1.70+01
Cs-139	6.93-01
Ba-139	2.48+00
Cs-138	8.61-01
Cs-137	1.41+00
Zr-95	1.76-01
Nb-95	2.25-01
Y-93	1.66-01
Sr-92	2.39+00
Y-92	2.50+00
Sr-91	5.19+00
Y-91	2.64+00
Y-91m	3.09+00
Rb-90	2.55-01
Sr-90	1.35+00
Y-90	1.27+00
Rb-89	4.20-01
Sr-89	1.32+01
Cr-51	3.09-01
Mn-56	1.47-01

TABLE 12.2-122 (Cont)

Isotope	Activity (Ci)(2)(3)
Co-58	4.51+00
Co-60	5.89-01
Fe-59	5.32-01
Br-83	4.12+00
Br-84	1.38+00
Br-85	2.09-02
I-131	2.96+02
I-132	5.27+01
I-133	2.20+02
I-134	2.14+01
I-135	1.01+02
Mo-99	8.54+00
Tc-99m	1.73+01
Tc-101	1.24-01
Te-129m	1.37-01
Te-132	2.24+01
Cs-134	9.34-01
Cs-136	2.01-01
Np-239	7.93+01
Total	9.07+02

(1) Design basis:
  a. Based on shielding design source terms given in Table 12.2-119
  b. Assume one backwash from the condensate demineralizers
  c. Source volume = 2181 gal

(2) 1.91-01 = 1.91×10^{-1}

(3) Isotopes with an activity less than 1.00-02 Ci are not listed

(Historical Information)

TABLE 12.2-123

TURBINE SEALING STEAM SHIELDING DESIGN SOURCE TERMS (1)

Isotope	Specific Activity ($\mu Ci/g$) (2) (3)
N-13	6.69-04
Ba-142	5.45-05
Cs-141	8.88-05
Ba-141	9.41-05
Cs-140	2.19-03
Cs-139	1.34-03
Ba-139	7.91-05
Cs-138	4.92-04
Sr-93	3.68-05
Sr-92	5.41-05
Rb-91	1.09-03
Sr-91	3.24-05
Rb-90	1.61-03
Rb-89	5.02-04
Rb-88	7.12-05
Br-83	1.05-04
Br-84	1.59-04
Br-85	2.56-05
I-131	9.43-05
I-132	8.22-04
I-133	6.65-04
I-134	1.52-03
I-135	9.27-04
Tc-99m	1.00-04
Tc-101	3.25-05
Te-132	1.77-05
Np-239	8.63-05
Xe-135	4.74-03
Xe-135m	7.14-02

(Historical Information)

TABLE 12.2-123 (Cont)

Isotope	Specific Activity (μCi/g)[2][3]
Xe-133m	5.97-05
Xe-133	1.04-03
Kr-83m	1.36-03
Total	9.16-02

---

(1) Design basis:

a. Based on shielding design source terms given in Table 12.2-118.

b. No decay is assumed between the condensate demineralizer and entrance into the Turbine Steam Sealing System

(2) 6.69-04 = 6.69×10^{-4}

(3) Isotopes with specific activities less than 1.00-05 are not listed

(Historical Information)

TABLE 12.2-124

PIPING BETWEEN THE MAIN CONDENSER AND SJAE SYSTEM

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Specific Activity (Ci/cc)[2][3]
N-16	2.89-06[4][5]
O-19	1.26-07
Kr-83m	1.91-09
Kr-85m	3.50-09
Kr-87	1.13-08
Kr-88	1.13-08
Kr-89	7.09-08
Kr-90	1.30-07
Kr-91	9.96-09
Kr-92	2.31-08
Kr-93	2.41-09
Rb-92	2.11-09
Xe-133	4.63-09
Xe-135m	1.45-08
Xe-135	1.24-08
Xe-137	8.20-08
Xe-138	5.05-08
Xe-139	1.36-07
Xe-140	1.13-07
Xe-141	3.36-09
Total	3.80-06

(Historical Information)

TABLE 12.2-124 (Cont)

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(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7, 12.2-8, and 12.2-10

b. The following transit times are assumed: 7.64 seconds from the RPV nozzle to the main condenser inlet; 0.35 seconds through the main condenser to the offgas piping inlet; 1.41 seconds from the offgas inlet to the SJAE intercondenser

c. It is assumed that 20% of the N-16 in steam in both the moisture separator and the main condenser is washed out

d. Effective density of offgas = 5.408-05 g/cc.

(2) 2.89-06 = 2.89×10^{-6}

(3) Isotopes with specific activities less than 1.00-09 Ci/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-125

SJAE INTERCONDENSER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
N-16	1.50+01[4][5]
O-19	7.06-01
Kr-83m	1.10-02
Kr-85m	2.00-02
Kr-87	6.45-02
Kr-88	6.46-02
Kr-89	4.04-01
Kr-90	7.28-01
Kr-91	5.29-01
Kr-92	5.29-01
Xe-133	2.65-02
Xe-135m	8.33-02
Xe-135	7.11-02
Xe-137	4.70-01
Xe-138	2.89-01
Xe-139	7.65-01
Xe-140	6.19-01
Xe-141	1.25-02
Total	2.04-01

(1) Design basis:

a. Based on shielding design source terms given in Table 12.2-7, 12.2-8, 12.2-10, and 12.2-124

b. SJAE intercondenser steam volume = 2.74+06 cc

c. Effective off-gas density = 1.621-04 g/cc

d. Transit time of 0.6485 seconds through the intercondenser

e. It is assumed that 20 percent of the N-16 in steam in both the moisture separator and the main condensers is washed out

(Historical Information)

TABLE 12.2-125 (Cont)

(2) 5.64-03 = 5.64×10^{-3}

(3) Isotopes with activities less than 1.00-02 Ci are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-126

SJAE AFTERCONDENSER SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
N-13	1.03-02[4][5]
N-16	2.51+01[4][5]
O-19	1.27+00
Kr-83m	2.01-02
Kr-85m	3.67-02
Kr-87	1.18-01
Kr-88	1.18-01
Kr-89	7.39-01
Kr-90	1.31+00
Kr-91	9.10-01
Kr-92	1.54-01
Kr-93	1.28-02
Xe-133	4.96-02
Xe-135m	1.53-01
Xe-135	1.30-01
Xe-137	8.60-01
Xe-138	5.30-01
Xe-139	1.38+00
Xe-140	1.09+00
Xe-141	1.59-02
Total	3.41+01

(1) Design basis:

a. Based on shielding design source terms given in Tables 12.2-7, 12.2-8, and 12.2-10

b. SJAE aftercondenser steam volume = 9.87+05 cc

(Historical Information)

TABLE 12.2-126 (Cont)

c. Effective off-gas density = 4.499-4 g/cc

d. The following transit times are assumed: 0.00 seconds between the SJAE condensers; 1.19 seconds through the aftercondenser

e. It is assumed that 20 percent of the N-16 in steam in both the moisture separator and the main condensers is washed out

(2) 1.03-02 = 1.03×10^{-2}

(3) Isotopes with activities less than 1.00-02 Ci are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-127

SJAE INTERCONDENSER DRAIN

SHIELDING DESIGN SOURCE TERMS[(1)]

Isotope	Specific Activity (μCi/cc)[(2)(3)]
N-16	6.05+00[(4)(5)]
Cs-144	3.88-01
Ba-144	2.81-02
Cs-143	2.51-02
Cs-142	9.41-01
Cs-141	6.52-01
Cs-140	8.58+00
Cs-139	1.22+00
Cs-138	1.35-01
Rb-94	1.96-01
Rb-93	1.78+00
Rb-92	1.82+01
Rb-91	6.87+00
Rb-90	3.78+00
Rb-89	3.93-01
Rb-88	5.42-02
I-132	1.19-02
I-134	2.38-02
I-135	1.30-02
Total	4.94+01

(1) Design basis:

a. Source terms are a result of two inputs, daughter particulates from the off-gas flow, and 20 percent N-16 and 100 percent particulates from the motive steam

(Historical Information)

TABLE 12.2-127 (Cont)

b. Motive steam data are based on shielding design source terms given in Tables 122-8 with 4.31 seconds decay

c. Off-gas data are based on shielding design source terms/assumptions in Table 12.2-125

d. Based on a drain flow of 39 gpm

e. Density = 1 g/cc.

(2) 6.058 + 00 = 6.05×10^0

(3) Isotopes with specific activities less than 1.00-02 μCi/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-128

SJAE AFTERCONDENSER DRAIN

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (μCi/cc)(2)(3)
N-16	4.60+0(4)(5)
Cs-144	3.94-01
Ba-144	1.54-02
Cs-143	1.05-02
Cs-142	4.72-01
Cs-141	3.17-01
Cs-140	7.31+00
Cs-139	1.08+00
Cs-138	1.22-01
Rb-94	9.22-02
Rb-93	9.84-01
Rb-92	1.21+01
Rb-91	5.40+00
Rb-90	3.32+00
Rb-89	3.55-01
Rb-88	4.92-02
I-134	1.90-02
I-135	1.04-02
Total	3.67+01

(1) Design basis:

a. Source terms are a result of two inputs: daughter particulates from the off-gas flow, and 20 percent N-16 and 100 percent particulates from the motive steam

(Historical Information)

TABLE 12.2-128 (Cont)

b. Off-gas data are based on shielding design source terms/assumptions in Table 12.2-126.

c. Motive steam data are based on shielding design source terms given in Tables 12.2-8 with 4.85 seconds decay

d. Based on a 24.8 gpm drain flow from the aftercondenser

e. Density = 1 g/cc

(2) 1.03-03 = 1.03×10^{-3}

(3) Isotopes with specific activities less than 1.00-02 μCi/cc are not listed

(4) For HWC operation at a nominal 22.5 scfm of H_2 multiply steam concentration source term by a factor of 2.5.

(5) For HWC operation at a nominal 35 scfm of H_2 multiply steam concentration source term by a factor of 4.3.

(Historical Information)

TABLE 12.2-129

MECHANICAL VACUUM PUMP SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Specific Activity (Ci/cc)(2)(3)
Kr-83m	3.29-11
Kr-85m	6.89-09
Kr-85	2.16-09
Kr-88	1.67-09
Xe-131m	1.52-09
Xe-133m	2.31-08
Xe-133	7.73-07
Xe-135m	2.31-09
Xe-135	3.09-07
I-131	2.85-08
I-132	4.75-09
I-133	8.44-08
I-135	1.43-08
Total	1.25-06

(1) Design basis:

a) Based on shielding design source terms given in Tables 12.2-7 to 12.2-8

b) Source terms are based on main steam normalized to xenon release data for the mechanical vacuum pump given in NUREG-0016, Section 2.2.7, and increased to corresponding to 500,000 μCi/s release rate

c) Density = 1.043-04 g/cc

(2) 3.29-11 = 3.29×10^{-11}

(3) Isotopes with a specific activity less than 1.00-11 Ci/cc are not listed

(Historical Information)

TABLE 12.2-130

TURBINE BUILDING EQUIPMENT DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	1.50-03
Br-84	9.63-04
I-131	2.06-03
I-132	1.48-02
I-133	1.35-02
I-134	1.28-02
I-135	1.74-02
Sr-89	5.08-04
Sr-91	9.85-03
Sr-92	1.16-02
Mo-99	3.44-03
Tc-99m	3.72-02
Tc-101	2.14-03
Te-132	7.84-03
Cs-138	6.61-03
Ba-139	1.20-02
Ba-140	1.43-03
Ba-141	3.34-03
Ba-142	2.04-03
Np-239	3.74-02
Mn-56	1.03-03
Co-58	1.59-04
La-142	1.56-03
La-141	1.41-03

TABLE 12.2-130 (Cont)

Isotope	Activity (Ci)[2][3]
Y-92	3.62-03
Y-91m	4.02-03
Total	2.14-01

(1) Design basis:

a. Sump volume = 840 gallons

b. Input rate of 4 gpm for 210 minutes

c. Input activity = 0.001 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11

(2) 1.50-03 = 1.50×10^{-3}

(3) Isotopes with an activity less than 1.00-04 Ci are not listed

(Historical Information)

TABLE 12.2-131

TURBINE BUILDING HIGH CONDUCTIVITY SUMP

SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
Br-83	8.60-04
Br-84	3.65-04
I-131	2.60-03
I-132	1.37-02
I-133	1.51-02
I-134	5.11-03
I-135	1.52-02
Sr-89	6.52-04
Sr-91	9.60-03
Sr-92	6.99-03
Mo-99	4.23-03
Tc-99m	3.28-02
Tc-101	8.03-04
Te-132	9.69-03
Cs-138	2.51-03
Ba-139	5.42-03
Ba-140	1.82-03
Ba-141	1.25-03
Ba-142	7.66-04
Np-239	4.56-02
Mn-56	6.08-04
Co-58	2.04-04
W-187	1.04-04
La-142	7.62-04
La-141	1.10-03
La-140	1.76-04

(Historical Information)

TABLE 12.2-131 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-92	5.46-03
Y-91m	5.27-03
TOTAL	1.89-01

(1) Design basis:

a. Sump volume = 540 gallons

b. Input rate of 0.75 gpm for 720 minutes

c. Input activity 0.02 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11.

(2) 8.60-04 = 8.60×10^{-4}

(3) Isotopes with an activity less than 1.00-04 Ci are not listed

(Historical Information)

TABLE 12.2-132

TURBINE BUILDING FLOOR DRAIN SUMP

SHIELDING DESIGN SOURCE TERMS(1)

Isotope	Activity (Ci)(2)(3)
Br-83	7.78-05
Br-84	3.38-05
I-131	2.03-04
I-132	1.12-03
I-133	1.21-03
I-134	4.74-04
I-135	1.28-03
Sr-89	1.28-03
Sr-91	7.94-04
Sr-92	6.28-04
Mo-99	3.33-04
Tc-99m	2.76-03
Tc-101	7.44-05
Te-132	7.62-04
Cs-138	2.32-04
Ba-139	5.00-04
Ba-140	1.42-04
Ba-141	1.16-04
Ba-142	7.10-05
Np-239	3.59-03
Mn-56	5.48-05
Co-58	1.59-05
La-142	7.02-05
La-141	9.55-05
La-140	1.17-05

(Historical Information)

TABLE 12.2-132 (Cont)

Isotope	Activity (Ci)(2)(3)
Y-92	4.46-04
Y-91m	4.26-04
Total	1.56-02

(1) Design basis:

a. Sump volume = 840 gallons

b. Input rate of 1.39 gpm for 605 minutes

c. Input activity 0.001 x primary coolant activity, Tables 12.2-8, 12.2-9, and 12.2-11.

(2) 7.78-05 = 7.78×10^{-5}

(3) Isotopes with an activity less than 1.00-05 Ci are not listed

(Historical Information)

TABLE 12.2-133

TURBINE BUILDING SHIELDING DESIGN RADIATION SOURCE DESCRIPTION

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-133 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-133 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-133 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-133 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

Security Related Information
Table withheld Under 10 CFR 2.390

(Historical Information)

TABLE 12.2-133 (Cont)

Room Number	Dominant Radiation Source	Source Identification	Source Geometry	Effective Source Density (gm/cc)	Equipment Self-Shielding (inches of steel)	FSAR Source Table

(Historical Information)

TABLE 12.2-134

POST-ACCIDENT SOURCE TERMS FOR DRYWELL AND GAS CONTAINING SYSTEMS[1]

Isotope	Curies
I-131	2.17+07
I-132	3.29+07
I-133	4.86+07
I-134	5.69+07
I-135	4.41+07
Kr-83m	1.44+07
Kr-85m	4.49+07
Kr-85	1.42+06
Kr-87	8.07+07
Kr-88	1.11+08
Kr-89	1.38+08
Xe-131m	8.97+05
Xe-133m	4.79+06
Xe-133	1.94+08
Xe-135m	5.38+07
Xe-135	1.86+08
Xe-137	1.77+08
Xe-138	1.65+08
Total	1.37+09

(1) Source terms are based on 100 percent noble gases and 25 percent halogens of the core inventory, with no credit taken for decay. See Table 15.6-13 for detailed description

(Historical Information)

TABLE 12.2-135

POST-ACCIDENT SOURCE TERMS FOR DEPRESSURIZED

LIQUID CONTAINING SYSTEMS(1)

Isotope	Curies
I-131	4.32+07
I-132	6.44+07
I-133	9.96+07
I-134	1.16+08
I-135	9.07+07
I-136	4.61+07
Br-83	7.58+06
Br-84	1.36+07
Br-85	1.91+07
Rb-88	1.06+06
Rb-89	1.41+06
Rb-90	1.71+06
Rb-91	1.61+06
Rb-92	1.30+06
Sr-89	1.41+06
Sr-91	1.72+06
Sr-92	1.57+06
Sr-93	1.66+06
Sr-94	1.28+06
Y-90	1.71+06
Y-91m	1.01+06
Y-91	1.69+06
Y-92	1.75+06
Y-93	1.81+06
Y-94	1.60+06
Y-95	1.84+06
Zr-95	1.84+06
Zr-97	2.85+06
Nb-95	1.90+06

(Historical Information)

TABLE 12.2-135 (Cont)

Isotope	Curies
Nb-97m	1.77+06
Nb-97	1.84+06
Mo-99	1.84+06
Mo-101	1.48+06
Mo-102	1.19+06
Tc-99m	1.63+06
Tc-101	1.49+06
Tc-102	1.23+06
Ru-103	8.92+05
Rh-103m	8.92+05
Sn-130	5.95+05
Sb-130	5.95+05
Sb-131	8.03+05
Sb-132	9.96+05
Sb-133	1.01+06
Te-131	7.73+05
Te-132	1.28+06
Te-133m	1.39+06
Te-133	8.92+05
Te-134	2.05+06
Cs-138	1.90+06
Cs-139	1.92+06
Cs-140	1.75+06
Cs-142	9.22+05
Ba-137m	1.75+06
Ba-139	1.86+06
Ba-140	1.87+06
Ba-141	1.87+06
Ba-142	1.70+06
La-140	1.87+06
La-141	1.90+06
La-143	1.74+06

(Historical Information)

TABLE 12.2-135 (Cont)

Isotope	Curies
La-142	1.73+06
Ce-141	1.90+06
Ce-143	1.74+06
Ce-144	1.45+06
Ce-145	1.15+06
Ce-146	8.80+05
Pr-143	1.74+06
Pr-144	1.48+06
Pr-145	1.15+06
Pr-146	9.13+05
Nd-147	6.69+05
Total	5.99+08

(1) Source terms are based on 50 percent halogens and 1 percent solid fission products of the core inventory, with no credit taken for decay. See Table 12.3-2 for detailed description. Isotopes with an activity less than 5.00+05 Ci are not listed

(Historical Information)

TABLE 12.2-136

POST-ACCIDENT SOURCE TERMS FOR PRESSURIZED

LIQUID CONTAINING SYSTEMS[1]

Isotope	Curies
I-131	4.32+07
I-132	6.44+07
I-133	9.96+07
I-134	1.16+08
I-135	9.07+07
I-136	4.61+07
Br-83	7.58+06
Br-84	1.36+07
Br-85	1.91+07
Kr-83m	1.54+07
Kr-85m	3.86+07
Kr-85	1.31+06
Kr-87	7.43+07
Kr-88	1.03+08
Kr-89	1.36+08
Xe-133m	5.05+06
Xe-133	1.98+08
Xe-135m	5.35+07
Xe-135	1.87+08
Xe-137	1.78+08
Xe-138	1.75+08
Rb-88	1.06+06
Rb-89	1.41+06
Rb-90	1.71+06
Rb-91	1.61+06
Rb-92	1.30+06
Sr-89	1.41+06
Sr-91	1.72+06

TABLE 12.2-136 (Cont)

Isotope	Curies
Sr-92	1.57+06
Sr-93	1.66+06
Sr-94	1.28+06
Y-90	1.71+06
Y-91m	1.01+06
Y-91	1.69+06
Y-92	1.75+06
Y-93	1.81+06
Y-94	1.60+06
Y-95	1.84+06
Zr-95	1.84+06
Zr-97	2.85+06
Nb-95	1.90+06
Nb-97m	1.77+06
Nb-97	1.84+06
Mo-99	1.84+06
Mo-101	1.48+06
Mo-102	1.19+06
Tc-99m	1.63+06
Tc-101	1.49+06
Tc-102	1.23+06
Sb-133	1.01+06
Te-132	1.28+06
Te-133m	1.39+06
Te-134	2.05+06
Cs-138	1.90+06
Cs-139	1.92+06
Cs-140	1.75+06
Ba-137m	1.75+06
Ba-139	1.86+06
Ba-140	1.87+06
Ba-141	1.87+06

(Historical Information)

TABLE 12.2-136 (Cont)

Isotope	Curies
Ba-142	1.70+06
La-140	1.87+06
La-141	1.90+06
La-143	1.74+06
La-142	1.73+06
Ce-141	1.90+06
Ce-143	1.74+06
Ce-144	1.45+06
Ce-145	1.15+06
Pr-143	1.74+06
Pr-144	1.48+06
Pr-145	1.15+06
Total	1.77+09

(1) Source terms are based on 100 percent noble gases, 50 percent halogens, and 1 percent solid fission products of the core inventory, with no credit taken for decay. See Table 12.3-2 for detailed description. Isotopes with an activity of less than 1.0 x 10^6 Ci are not listed

(Historical Information)

TABLE 12.2-137

CONDENSATE STORAGE TANK SHIELDING DESIGN SOURCE TERMS[1]

Isotope	Activity (Ci)[2][3]
I-131	7.40-03
Sr-89	7.86-01
Sr-90	2.95-02
Zr-95	1.14+00
Zr-97	1.06-03
Nb-95	1.15+00
Te-129m	1.80-03
Te-132	2.12-01
Cs-136	1.78-02
Cs-137	2.24-01
Ba-140	5.38-01
Ce-141	7.02-01
Ce-143	1.07-02
Pr-143	5.25-01
La-140	1.01-01
Nb-95m	6.25-03
Y-91	9.00-01
Total	6.35+00

(1) Design basis:
   a) Source volume = 500,000 gallons
   b) Based on operating experiences
   c) Source activity is normalized to yield a dose rate of 2 mrem/h at 1 foot from tank

(2) $7.40\text{-}03 = 7.40 \times 10^{-3}$

(3) Isotopes with an activity less than 1.00-03 Ci are not listed

(Historical Information)

TABLE 12.2-138

ESTIMATED INPLANT AIRBORNE RADIOACTIVE RELEASES (CURIES/YEAR)(1)

Nuclide	Drywell	Turbine Building	Reactor Building	Radwaste Building(2)
H-3	.0	2.6+01	2.6+01	.0
I-131	1.1-02	1.2-01	2.1-02	1.1-02
I-133	1.5-01	1.6+00	3.0-01	1.5-01
Ar-41	1.5+01	.0	.0	.0
Kr-83m	.0	.0	.0	.0
Kr-85m	1.0+00	2.5+01	3.0+00	.0
Kr-85	.0	.0	.0	.0
Kr-87	.0	6.1+01	2.0+00	.0
Kr-88	1.0+00	9.1+01	3.0+00	.0
Kr-89	.0	5.8+02	2.0+00	2.9+01
Xe-131m	.0	.0	.0	.0
Xe-133m	.0	.0	.0	.0
Xe-133	2.7+01	1.5+02	8.3+01	2.2+02
Xe-135m	1.5+01	4.0+02	4.5+01	5.3+02
Xe-135	3.3+01	3.3+02	9.4+01	2.8+02
Xe-137	4.5+01	1.0+03	1.3+02	8.3+01
Xe-138	2.0+00	1.0+03	6.0+00	2.0+00
Cr-51	2.0-04	9.0-04	9.0-04	7.0-04
Mn-54	4.0-04	6.0-04	1.0-03	4.0-03
Co-58	1.0-04	1.0-03	2.0-04	2.0-04
Fe-59	9.0-05	1.0-04	3.0-04	3.0-04
Co-60	1.0-03	1.0-03	4.0-03	7.0-03
Zn-65	1.0-03	6.0-03	4.0-03	3.0-04
Sr-89	3.0-05	6.0-03	2.0-05	.0
Sr-90	3.0-06	2.0-05	7.0-06	.0
Nb-95	1.0-03	6.0-06	9.0-03	4.0-06
Zr-95	3.0-04	4.0-05	7.0-04	8.0-04
Mo-99	6.0-03	2.0-03	6.0-02	3.0-06
Ru-103	2.0-04	5.0-05	4.0-03	1.0-06

(Historical Information)

TABLE 12.2-138 (Cont)

Nuclide	Drywell	Turbine Building	Reactor Building	Radwaste Building[2]
Ag-110m	4.0-07	.0	2.0-06	.0
Sb-124	2.0-05	1.0-04	3.0-05	7.0-05
Cs-134	7.0-04	2.0-04	4.0-03	2.4-03
Cs-136	1.0-04	1.0-04	4.0-04	.0
Cs-137	1.0-03	1.0-03	5.0-03	4.0-03
Ba-140	2.0-03	1.0-02	2.0-02	4.0-06
Ce-141	2.0-04	1.0-02	7.0-04	7.0-06

(1) BWR-GALE Computer Code (Reference 12.2-1)

(2) Includes off-gas system releases

(Historical Information)

TABLE 12.2-139

ESTIMATED INPLANT DISTRIBUTION OF AIRBORNE

RADIOACTIVITY RELEASES[1]

Turbine Building	Airborne Release Fraction (percent)
Condenser areas	60.30
Steam jet air ejector areas	7.00
Mechanical vacuum pump areas	3.50
Turbine hall areas	9.20
Other equipment areas[2]	20.00
	100.00
Reactor Building	
RWCU pump areas	37.84
RWCU filter/demin areas	15.68
Refueling area[3]	14.64
Emergency Core Cooling System (ECCS) areas	11.84
Other equipment areas[2]	20.00
	100.00
Drywell	
Drywell area	100.00
Radwaste Building[4]	
Liquid waste handling areas	71.60
Solid waste handling areas	8.40
Other equipment areas[2]	20.00
	100.00

(Historical Information)

TABLE 12.2-139 (Cont)

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(1) Based on EPRI-NP495, in Reference 12.2-2

(2) Assumes 80 percent of building release by major areas listed and 20 percent by all other equipment areas

(3) All of the reactor building tritium is assigned to the refueling deck

(4) Includes off-gas equipment areas

(Historical Information)

TABLE 12.2-140

AIRBORNE SOURCE DESCRIPTIONS

Area	Rooms(1)	Total Flow (cfm)	Total Annual Exhaust (cc/yr)(2)
Turbine Building:			
Condenser	1113,1114,1115,1116, 1117,1220,1310,1311, 1312,1313,1405	38,500	5.7×10^{14}
Steam jet air ejector (SJAE)	1218,1219	2600	3.9×10^{13}
Mechanical vacuum pump	1206	3100	4.6×10^{13}
Turbine hall	1512,1513,1514	17,100	2.5×10^{14}
Others	1102,1103,1104,1106, 1107,1108,1110,1111, 1112,1128,1207,1208, 1209,1210,1211,1212, 1213,1214,1215,1216, 1230,1305,1308,1309, 1504,1505,1506,1508, 1509,1510,1511	38,050	5.7×10^{14}
Reactor Building:			
Reactor water cleanup pumps (RWCU pumps)	4319,4321,4402,4403, 4405,4505,4506,4509	5600	8.3×10^{13}

TABLE 12.2-140 (Cont)

Area	Rooms(1)	Total Flow (cfm)	Total Annual Exhaust Flow (cc/yr)(2)
Reactor water cleanup filter demineralizers (RWCU F/D)	4406,4407,4502,4503, 4620,4621	2800	4.2×10^{13}
Refueling deck	4706	23,020(3)	3.4×10^{14}(4)
Emergency Core Cooling Systems (ECCS)	4102,4104,4105,4107, 4109,4110,4111,4113, 4114,4116,4118,4128, 4208,4214,4215,4326, 4327	10,200	1.5×10^{14}
Others	4106,4115,4316,4318, 4320,4321,4326,4329, 4603	19,500	2.9×10^{14}
Drywell:	4220,4221,4341	9000	1.1×10^{13}(5)
Radwaste Building:			
Liquid	3103,3104,3105,3107, 3108,3109,3112,3113, 3114,3115,3116,3121, 3122,3123,3124,3136, 3137,3138,3145,3146, 3149,3150,3151,3171, 3172,3173,3174,3175, 3176,3191,3192,3193, 3194,3195,3196,3316, 3320,3321,3325	25,970	3.9×10^{14}

TABLE 12.2-140 (Cont)

Area	Rooms(1)	Total Flow (cfm)	Total Annual Exhaust Flow (cc/yr)(2)
Solid	3118,3119,3120,3128, 3129,3130,3132,3133, 3134,3135,3147,3148, 3309,3311,3322,3323, 3326, solidification area	25,210	3.8×10^{14}
Others	3096,3101,3102,3112, 3117,3141,3142,3143, 3155,3156,3158,3159, 3160,3161,3165,3166, 3167,3168,3170,3180, 3181,3182,3183,3187, 3189,3198,3199,3317, 3318,3319,3324,3424	34,240	5.1×10^{14}

Notes:

(1) See Figures 12.3-1 through 12.3-21

(2) Based on continuous release, 365 days per year, except as noted

(3) Average value based on 80 percent operation in normal ventilation mode and 20 percent operation in refueling ventilation mode

(4) Tritium releases assumed to occur during operation in refueling ventilation mode with a total exhaust flow of 1.7×10^{14} cc/yr

(5) Drywell releases are assumed to occur over a 30-day period

(Historical Information)

TABLE 12.2-141

TURBINE BUILDING CONCENTRATIONS OF MPC FRACTIONS

Nuclide	MPC µCi/cc	Condenser Area Concen µCi/cc	Condenser Area Fract of MPC	Steam Jet Air Ejector Area Concen µCi/cc	Steam Jet Air Ejector Area Fract of MPC	Mechanical Vacuum Pump Area Concen µCi/cc	Mechanical Vacuum Pump Area Fract of MPC	Turbine Hall Area Concen µCi/cc	Turbine Hall Area Fract of MPC	Other Areas Concen µCi/cc	Other Areas Fract of MPC
H-3	5.0-06	2.7-08	5.5-03	4.7-08	9.4-03	2.0-08	3.9-03	9.4-09	1.9-03	9.2-09	1.8-03
I-131	9.0-09	1.3-10	1.4-02	2.2-10	2.4-02	9.1-11	1.0-02	4.3-11	4.8-03	4.2-11	4.7-03
I-133	3.0-08	1.7-09	5.6-02	2.9-09	9.6-02	1.2-09	4.0-02	5.8-10	1.9-02	5.7-10	1.9-02
Ar-41	2.0-06	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Kr-83m	1.0-06	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Kr-85m	6.0-06	2.6-08	4.4-03	4.5-08	7.5-03	1.9-08	3.2-03	9.0-09	1.5-03	8.8-09	1.5-03
Kr-85	1.0-05	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Kr-87	1.0-06	6.4-08	6.4-02	1.1-07	1.1-01	4.6-08	4.6-02	2.2-08	2.2-02	2.2-08	2.2-02
Kr-88	1.0-06	9.6-08	9.6-02	1.6-07	1.6-01	6.9-08	6.9-02	3.3-08	3.3-02	3.2-08	3.2-02
Kr-89	1.0-06	6.1-07	6.1-01	1.0-06	1.0+00	4.4-07	4.4-01	2.1-07	2.1-01	2.0-07	2.0-01
Xe-131m	2.0-05	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Xe-133m	1.0-05	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Xe-133	1.0-05	1.6-07	1.6-02	2.7-07	2.7-02	1.1-07	1.1-02	5.4-08	5.4-03	5.3-08	5.3-03
Xe-135m	1.0-06	4.2-07	4.2-01	7.2-07	7.2-01	3.0-07	3.0-01	1.4-07	1.4-01	1.4-07	1.4-01
Xe-135	4.0-06	3.5-07	8.7-02	6.0-07	1.5-01	2.5-07	6.3-02	1.2-07	3.0-02	1.2-07	2.9-02
Xe-137	1.0-06	1.1-06	1.1+00	1.8-06	1.8+00	7.6-07	7.6-01	3.6-07	3.6-01	3.5-07	3.5-01
Xe-138	1.0-06	1.1-06	1.1+00	1.8-06	1.8+00	7.6-07	7.6-01	3.6-07	3.6-01	3.5-07	3.5-01
Cr-51	2.0-06	9.5-13	4.7-07	1.6-12	8.1-07	6.8-13	3.4-07	3.3-13	1.6-07	3.2-13	1.6-07
Mn-54	4.0-08	6.3-13	1.6-05	1.1-12	2.7-05	4.6-13	1.1-05	2.2-13	5.4-06	2.1-13	5.3-06
Co-58	5.0-08	1.1-12	2.1-05	1.8-12	3.6-05	7.6-13	1.5-05	3.6-13	7.2-06	3.5-13	7.1-06
Fe-59	5.0-08	1.1-13	2.1-06	1.8-13	3.6-06	7.6-14	1.5-06	3.6-14	7.2-07	3.5-14	7.1-07
Co-60	9.0-09	1.1-12	1.2-04	1.8-12	2.0-04	7.6-13	8.4-05	3.6-13	4.0-05	3.5-13	3.9-05
Zn-65	6.0-08	6.3-12	1.1-04	1.1-11	1.8-04	4.6-12	7.6-05	2.2-12	3.6-05	2.1-12	3.5-05
Sr-89	3.0-08	6.3-12	2.1-04	1.1-11	3.6-04	4.6-12	1.5-04	2.2-12	7.2-05	2.1-12	7.1-05
Sr-90	1.0-09	2.1-14	2.1-05	3.6-14	3.6-05	1.5-14	1.5-05	7.2-15	7.2-06	7.1-15	7.1-06
Nb-95	1.0-07	6.3-15	6.3-08	1.1-14	1.1-07	4.6-15	4.6-08	2.2-15	2.2-08	2.1-15	2.1-08
Zr-95	3.0-08	4.2-14	1.4-06	7.2-14	2.4-06	3.0-14	1.0-06	1.4-14	4.8-07	1.4-14	4.7-07
Mo-99	2.0-07	2.1-12	1.1-05	3.6-12	1.8-05	1.5-12	7.6-06	7.2-13	3.6-06	7.1-13	3.5-06
Ru-103	8.0-08	5.3-14	6.6-07	9.0-14	1.10-6	3.8-14	4.7-07	1.8-14	2.3-07	1.8-14	2.2-07
Ag-110m	1.0-08	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Sb-124	2.0-08	1.1-13	5.3-06	1.8-13	9.0-06	7.6-14	3.8-06	3.6-14	1.8-06	3.5-14	1.8-06
Cs-134	1.0-08	1.2-13	2.1-05	3.6-13	3.6-05	1.5-13	1.5-05	7.2-14	7.2-06	7.1-14	7.1-06
Cs-136	2.0-07	1.1-13	5.3-07	1.8-13	9.0-07	7.6-14	3.8-07	3.6-14	1.8-07	3.5-14	1.8-07
Cs-137	1.0-08	1.1-12	1.1-04	1.8-12	1.8-04	7.6-13	7.6-05	3.6-13	3.6-05	3.5-13	3.5-05
Ba-140	4.0-08	1.1-11	2.6-04	1.8-11	4.5-04	7.6-12	1.9-04	3.6-12	9.0-05	3.5-12	8.8-05
Ce-141	2.0-07	1.1-11	5.3-05	1.8-11	9.0-05	7.6-12	3.8-05	3.6-12	1.8-05	3.5-12	1.8-05

Note: MPC's for noble gases are calculated on the basis of a free field dose equivalent rate of 2.5 mrem/hr in an infinite cloud geometry. These concentrations will result in much lower dose equivalent rates in the finite geometry of power plant compartments. In addition, the dose rates from the airborne noble gases will be much lower than the dose rates from the contained sources in the area.

(Historical Information)

TABLE 12.2-142

REACTOR BUILDING CONCENTRATIONS OF MPC FRACTIONS

Nuclide	MPC μCi/cc	Reactor Water Cleanup Pump Area Concen μCi/cc	Reactor Water Cleanup Pump Area Fract of MPC	Reactor Water Cleanup Filter Area Concen μCi/cc	Reactor Water Cleanup Filter Area Fract of MPC	Refueling Area Concen μCi/cc	Refueling Area Fract of MPC	ECCS Area Concen μCi/cc	ECCS Area Fract of MPC	Other Areas Concen μCi/cc	Other Areas Fract of MPC
H-3	5.0-06	.0	.0	.0	.0	1.6-07	3.1-02	.0	.0	.0	.0
I-131	9.0-09	9.5-11	1.1-02	7.9-11	8.8-03	9.0-12	1.0-03	1.6-11	1.8-03	1.4-11	1.6-03
I-133	3.0-08	1.4-09	4.5-02	1.1-09	3.8-02	1.3-10	4.3-03	2.3-10	7.8-03	2.1-10	6.9-03
Ar-41	2.0-06	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Kr-83m	1.0-06	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Kr-85m	6.0-06	1.4-08	2.3-03	1.1-08	1.9-03	1.3-09	2.1-04	2.3-09	3.9-04	2.1-09	3.4-04
Kr-85	1.0-05	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Kr-87	1.0-06	9.1-09	9.1-03	7.5-09	7.5-03	8.5-10	8.5-04	1.6-09	1.6-03	1.4-09	1.4-03
Kr-88	1.0-06	1.4-08	1.4-03	1.1-08	1.1-02	1.3-09	1.3-03	2.3-09	2.3-03	2.1-09	2.1-03
Kr-89	1.0-06	9.1-09	9.1-03	7.5-09	7.5-03	8.5-10	8.5-04	1.6-09	1.6-03	1.4-09	1.4-03
Xe-131m	2.0-05	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Xe-133m	1.0-05	.0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Xe-133	1.0-05	3.8-07	3.8-02	3.1-07	3.1-02	3.5-08	3.5-03	6.5-08	6.5-03	5.7-08	5.7-03
Xe-135m	1.0-06	2.0-07	2.0-01	1.7-07	1.7-01	1.9-08	1.9-02	3.5-08	3.5-02	3.1-08	3.1-02
Xe-135	4.0-06	4.3-07	1.1-01	3.5-07	8.8-02	4.0-08	1.0-02	7.3-08	1.8-02	6.5-08	1.6-02
Xe-137	1.0-06	5.9-07	5.9-01	4.9-07	4.9-01	5.6-08	5.6-02	1.0-07	1.0-01	9.0-08	9.0-02
Xe-138	1.0-06	2.7-08	2.7-02	2.3-08	2.3-02	2.6-09	2.6-03	4.7-09	4.7-03	4.1-09	4.1-03
Cr-51	2.0-06	4.1-12	2.0-06	3.4-12	1.7-06	3.8-13	1.9-07	7.0-13	3.5-07	6.2-13	3.1-07
Mn-54	4.0-08	4.5-12	1.1-04	3.8-12	9.4-05	4.3-13	1.1-05	7.8-13	1.9-05	6.9-13	1.7-05
Co-58	5.0-08	9.1-13	1.8-05	7.5-13	1.5-05	8.5-14	1.7-06	1.6-13	3.1-06	1.4-13	2.8-06
Fe-59	5.0-08	1.4-12	2.7-05	1.1-12	2.3-05	1.3-13	2.6-07	2.3-13	4.7-06	2.1-13	4.1-06
Co-60	9.0-09	1.8-11	2.0-03	1.5-11	1.7-03	1.7-12	1.9-04	3.1-12	3.5-04	2.8-12	3.1-04
Zn-65	6.0-08	1.8-11	3.0-04	1.5-11	2.5-04	1.7-12	2.8-05	3.1-12	5.2-05	2.8-12	4.6-05
Sr-89	3.0-08	9.1-14	3.0-06	7.5-14	2.5-06	8.5-15	2.8-07	1.6-14	5.2-07	1.1-14	4.6-07
Sr-90	1.0-09	3.2-14	3.2-05	2.6-14	2.6-05	3.0-15	3.0-06	5.5-15	5.5-06	4.8-15	4.8-06
Nb-95	1.0-07	4.1-11	4.1-04	3.4-11	3.4-04	3.8-12	3.8-05	7.0-12	7.0-05	6.2-12	6.2-05
Zr-95	3.0-08	3.2-12	1.1-04	2.6-12	8.8-05	3.0-13	1.0-05	5.5-13	1.8-05	4.8-13	1.6-05
Mo-99	2.0-07	2.7-10	1.4-03	2.3-10	1.1-03	2.6-11	1.3-04	4.7-11	2.3-04	4.1-11	2.1-04
Ru-103	8.0-08	1.8-11	2.3-04	1.5-11	1.9-04	1.7-12	2.1-05	3.1-12	3.9-05	2.8-12	3.4-05
Ag-110m	1.0-08	9.1-15	9.1-07	7.5-15	7.5-07	8.5-16	8.5-08	1.6-15	1.6-07	1.4-15	1.4-07
Sb-124	2.0-08	1.4-13	6.8-06	1.1-13	5.6-06	1.3-14	6.4-07	2.3-14	1.2-06	2.1-14	1.0-06
Cs-134	1.0-08	1.8-11	1.8-03	1.5-11	1.5-03	1.7-12	1.7-04	3.1-12	3.1-04	2.8-12	2.8-04
Cs-136	2.0-07	1.8-12	9.1-06	1.5-12	7.5-06	1.7-13	8.5-07	3.1-13	1.6-06	2.8-13	1.4-06
Cs-137	1.0-08	2.3-11	2.3-03	1.9-11	1.9-03	2.1-12	2.1-04	3.9-12	3.9-04	3.4-12	3.4-04
Ba-140	4.0-08	9.1-11	2.3-03	7.5-11	1.9-03	8.5-12	2.1-04	1.6-11	3.9-04	1.4-11	3.4-04
Ce-141	2.0-07	3.2-12	1.6-05	2.6-12	1.3-05	3.0-13	1.5-06	5.5-13	2.7-06	4.8-13	2.4-06

NOTE: MPC's for noble gases are calculated on the basis of a free field dose equivalent rate of 2.5 mrem/hr in an infinite cloud geometry. These concentrations will result in much lower dose equivalent rates in the finite geometry of power plant compartments. In addition, the dose rates from the airborne noble gases will be much lower than the dose rates from the contained sources in the area.

(Historical Information)

TABLE 12.2-143

DRYWELL CONCENTRATIONS AND MPC FRACTIONS(1)

Nuclide	MPC µCi/cc	Drywell Area Concen µCi/cc	Drywell Area Fract of MPC
H-3	5.0-06	.0	.0
I-131	9.0-09	1.0-09	1.1-01
I-133	3.0-08	1.4-08	4.5-01
Ar-41	2.0-06	1.4-06	6.8-01
Kr-83m	1.0-06	.0	.0
Kr-85m	6.0-06	9.1-08	1.5-02
Kr-85	1.0-05	.0	.0
Kr-87	1.0-06	.0	.0
Kr-88	1.0-06	9.1-08	9.1-02
Kr-89	1.0-06	.0	.0
Xe-131m	2.0-05	.0	.0
Xe-133m	1.0-05	.0	.0
Xe-133	1.0-05	2.5-06	2.5-01
Xe-135m	1.0-06	1.4-06	1.4+00
Xe-135	4.0-06	3.0-06	7.5-01
Xe-137	1.0-06	4.1-06	4.1+00
Xe-138	1.0-06	1.8-07	1.8-01
Cr-51	2.0-06	1.8-11	9.1-06
Mn-54	4.0-08	3.6-11	9.1-04
Co-58	5.0-08	9.1-12	1.8-04
Fe-59	5.0-08	8.2-12	1.6-04
Co-60	9.0-09	9.1-11	1.0-02
Zn-65	6.0-08	9.1-11	1.5-03
Sr-89	3.0-08	2.7-12	9.1-05
Sr-90	1.0-09	2.7-13	2.7-04
Nb-95	1.0-07	9.1-11	9.1-04
Zr-95	3.0-08	2.7-11	9.1-04
Mo-99	2.0-07	5.4-10	2.7-03

(Historical Information)

TABLE 12.2-143 (Cont)

		Drywell Area	
	MPC	Concen	Fract
Nuclide	μCi/cc	μCi/cc	of MPC
Ru-103	8.0-08	1.8-11	2.3-04
Ag-110m	1.0-08	3.6-14	3.6-06
Sb-124	2.0-08	1.8-12	9.1-05
Cs-134	1.0-08	6.4-11	6.4-03
Cs-136	2.0-07	9.1-12	4.5-05
Cs-137	1.0-08	9.1-11	9.1-03
Ba-140	4.0-08	1.8-10	4.5-03
Ce-141	2.0-07	1.8-11	9.1-05

(1) MPCs for noble gases are calculated on the basis of a free field dose equivalent rate of 2.5 mrem/h in an infinite cloud geometry. These concentrations will result in much lower dose equivalent rates in the finite geometry of power plant compartments. In addition, the dose rates from the airborne noble gases will be much lower than the dose rates from the contained sources in the area.

(Historical Information)

TABLE 12.2-144

RADWASTE BUILDING CONCENTRATIONS AND MPC FRACTIONS[(1)]

		Liquid Radwaste Handling Area		Solid Radwaste Handling Area		Others Area	
Nuclide	MPC µCi/cc	Concen µCi/cc	Fract of MPC	Concen µCi/cc	Fract of MPC	Concen µCi/cc	Fract of MPC
H-3	5.0-06	.0	.0	.0	.0	.0	.0
I-131	9.0-09	2.0-11	2.3-03	2.5-12	2.7-04	4.3-12	4.8-04
I-133	3.0-08	2.8-10	9.3-03	3.4-11	1.1-03	5.9-11	2.0-03
Ar-41	2.0-06	.0	.0	.0	.0	.0	.0
Kr-83m	1.0-06	.0	.0	.0	.0	.0	.0
Kr-85m	6.0-06	.0	.0	.0	.0	.0	.0
Kr-85	1.0-05	.0	.0	.0	.0	.0	.0
Kr-87	1.0-06	.0	.0	.0	.0	.0	.0
Kr-88	1.0-06	.0	.0	.0	.0	.0	.0
Kr-89	1.0-06	5.4-08	5.4-02	6.5-09	6.5-03	1.1-08	1.1-02
Xe-131m	2.0-05	.0	.0	.0	.0	.0	.0
Xe-133m	1.0-05	.0	.0	.0	.0	.0	.0
Xe-133	1.0-05	4.1-07	4.1-02	4.9-08	4.9-03	8.6-08	8.6-03
Xe-135m	1.0-06	9.8-07	9.8-01	1.2-07	1.2-01	2.1-07	2.1-01
Xe-135	4.0-06	5.2-07	1.3-01	6.3-08	1.6-02	1.1-07	2.7-02
Xe-137	1.0-06	1.5-07	1.5-01	1.9-08	1.9-02	3.3-08	3.3-02
Xe-138	1.0-06	3.7-09	3.7-03	4.5-10	4.5-04	7.8-10	7.8-04
Cr-51	2.0-06	1.3-12	6.5-07	1.6-13	7.8-08	2.7-13	1.4-07
Mn-54	4.0-08	7.4-12	1.9-04	9.0-13	2.2-05	1.6-12	3.9-05
Co-58	5.0-08	3.7-13	7.4-06	4.5-14	9.0-07	7.8-14	1.6-06
Fe-59	5.0-08	5.6-13	1.1-05	6.7-14	1.3-06	1.2-13	2.4-06
Co-60	9.0-09	1.3-11	1.4-03	1.6-12	1.7-04	2.7-12	3.1-04
Zn-65	6.0-08	5.6-13	9.3-06	6.7-14	1.1-06	1.2-13	2.0-06
Sr-89	3.0-08	.0	.0	.0	.0	.0	.0
Sr-90	1.0-09	.0	.0	.0	.0	.0	.0
Nb-95	1.0-07	7.4-15	7.4-08	9.0-16	9.0-09	1.6-15	1.6-08
Zr-95	3.0-08	1.5-12	4.9-05	1.8-13	6.0-06	3.1-13	1.0-05

(Historical Information)

TABLE 12.2-144 (Cont)

		Liquid Radwaste Handling Area		Solid Radwaste Handling Area		Others Area	
	MPC	Concen	Fract	Concen	Fract	Concen	Fract
Nuclide	Ci/cc	μCi/cc	of MPC	μCi/cc	of MPC	μCi/cc	of MPC
Mo-99	2.0-07	5.6-15	2.8-08	6.7-16	3.4-09	1.2-15	5.9-09
Ru-103	8.0-08	1.9-15	2.3-08	2.2-16	2.8-09	3.9-16	4.9-09
Ag-110m	1.0-08	.0	.0	.0	.0	.0	.0
Sb-124	2.0-08	1.3-13	6.5-06	1.6-14	7.8-07	2.7-14	1.4-06
Cs-134	1.0-08	4.4-12	4.4-04	5.4-13	5.4-05	9.4-13	9.4-05
Cs-136	2.0-07	.0	.0	.0	.0	.0	.0
Cs-137	1.0-08	7.4-12	7.4-04	9.0-13	9.0-05	1.6-12	1.6-04
Ba-140	4.0-08	7.4-15	1.9-07	9.0-16	2.2-08	1.6-15	3.9-08
Ce-141	2.0-07	1.3-14	6.5-08	1.6-15	7.8-09	2.7-15	1.4-08

(1) MPC's for noble gases are calculated on the basis of a free field dose equivalent rate of 2.5 mrem/h in an infinite cloud geometry. These concentrations will result in much lower dose equivalent rates in the finite geometry of power plant compartments. In addition, the dose rates from the airborne noble gases will be much lower than the dose rates from the contained sources in the area.

TABLE 12.2-145

LOW LEVEL RADWASTE STORAGE FACILITY SHIELDING DESIGN SOURCE TERMS

Isotope	Total* Drum Activity (Ci)	Total* DAW Box Activity (Ci)	Total* HIC Activity (Ci)
Cr-51	7,052.30	3.74	--
Mn-54	2,079.39	2.01	1,085.56
Fe-55	18,253.02	39.11	4,713.95
Co-58	129.64	11.45	3,878.91
Co-60	1,234.15	7.52	5,575.93
Fe-59	357.80	0.33	--
Ni-63	103.71	4.58	5,306.56
Sb-124	36.30	0.02	--
Sb-125	--	0.49	--
Zn-65	2,229.77	1.18	--
Ag-110m	108.90	0.09	--
H-3	124.45	0.10	--
C-14	--	0.01	21.55
Sr-90	--	0.03	5.39
Zr-95	--	0.13	--
Nb-95	--	0.38	--
Cs-134	--	0.81	2,882.24
Cs-137	15.56	1.10	3,286.30
Ba-137m	15.56	1.10	3,286.30
Ce-144	51.86	0.18	169.70

* Activity at the end of the fifth year.

(Historical Information)

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

Specific design features for maintaining personnel exposures as low as is reasonably achievable (ALARA) are discussed in this section. These features are used in addition to and in conjunction with the more generalized design features described in Section 12.1.2.

12.3.1.1 Common Equipment and Component Layout and Designs for ALARA

This section describes the design features used for several general classes of equipment and components. Since these classes of equipment are common to many of the plant systems, the features employed in each system to minimize exposures are similar and can be discussed generically by equipment type.

1. Filters - Filters that accumulate a significant amount of radioactivity are supplied with the means either to backflush and recharge the filter remotely or to perform cartridge replacement with semi-remote tools, i.e., long handled tools. For cartridge filters, adequate space is provided to allow removing, loading, and transporting the cartridges to the solid radwaste storage area in a shielded cask or drum. To make the removal of cartridges easier, quick disconnect features are provided to minimize the time needed for personnel to open/close the filter housings.

   Filters that are expected to accumulate very high amounts of radioactivity are provided with shielded remote reading instrument plug openings in their concrete ceilings. Remote reading instruments allow semi-remote readings of filter surface dose rates for radiation surveys without the opening and entering of filter cubicles and aid plant personnel in planning and preparation of radiation protection features.

(Historical Information)

2. Demineralizers - Demineralizers in radioactive waste and water treatment systems are designed so that spent resin can be remotely transferred to spent resin or regeneration tanks, and fresh resin can be remotely loaded into the demineralizers. The demineralizers and piping can be flushed with condensate or demineralized water to remove any accumulation of resin. The equipment and floor drain filter/demineralizers are typical examples, shown on Plant Drawings M-62-0 and M-63-0, respectively.

   The cubicles for the condensate demineralizers expected to accumulate very large amounts of radioactivity are provided with a shielded remote reading instrument plug opening in their concrete ceilings. Remote reading instruments allow semiremote readings of demineralizer surface dose rates for radiation surveys without opening and entering demineralizer cubicles. They also aid in planning and preparation of shielding features to provide for cubicle entry, if necessary.

3. Evaporators - Evaporators are provided with chemical addition and condensate or demineralized water flushing connections that also allow the use of chemicals for decontamination operations shown on Plant Drawing M-64-0. Adequate space is provided to facilitate removal of evaporator tube bundles. High activity concentrate carrying and low activity distillate carrying components are separated from each other wherever possible.

4. Pumps - Wherever practicable, pumps in radioactive areas are provided with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space and access for maintenance.

Small pumps are installed in a manner that allows easy removal if necessary. All pumps in radioactive waste and water treatment systems are provided with flanged connections for ease of removal. Pump casings and pump associated pipe spools have connections for draining the pump and pipe sections prior to maintenance. The use of base plates with drains connected to the floor drain system minimizes the spread of contamination resulting from pump leakage. Where possible, pumps in radioactive areas are placed in individual cubicles to enhance radiation protection and minimize the spread of contamination.

5. Tanks - Whenever practicable, tanks that contain radioactive material are provided with sloped bottoms and bottom outlet connections. Overflow lines are lower than vent lines and are directed to the waste collection system to prevent an overflow from spreading contamination within plant structures. For tanks containing radioactive material, each tank can be rinsed, and associated piping can be flushed to reduce radiation levels if required for entry into the tank cells. Adequate access and headroom are provided for removal, maintenance, or inspection of tank motor agitators and also for cleaning operations involving tank internals. The Solid Radwaste Collection System shown on Plant Drawings M-66-0, M-67-0 and M-68-0 provides examples of these features. All radioactive tanks are placed in individual, shielded cubicles. Active components such as valves are placed outside the cubicles to the extent possible. Those tanks holding radioactive liquids are surrounded by walls or dikes providing containment in case of a tank rupture. The diked-in areas are provided with floor drains for normal low flow leakage.

6. Heat exchangers - Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials with tube-to-tube sheet joints, welded to minimize leakage. Impingement baffles are provided, and tube side and shell side velocities are limited to minimize erosive effects. Space is reserved for tube bundle removal, and provisions are made for drainage and flushing.

7. Turbine condenser - Most of the condenser instrument readout devices are located in a shielded and low radiation condenser instrument compartment. The condenser is provided with corrosion resistant tubes of titanium-alloy steel to minimize corrosion effects and, in turn, reduce the need for tube plugging or other related maintenance and repair activities. Radioactive pipes have been routed in such a way as to minimize radiation exposures to personnel working at or near the condenser. Permanent ladders and platforms are provided to further enhance plant personnel maintenance operations and reduce radiation exposures.

8. Instruments - Instrument transmitting and readout devices are located in low radiation zones and away from radiation sources whenever practicable to reduce personnel exposure during maintenance. Sensing instrument devices, which for functional reasons are located in high radiation zones, are designed for easy removal to a lower radiation zone for maintenance and calibration. Some instruments located in high radiation zones, such as thermocouples, are provided in duplicate to preclude the need for immediate entry in case of instrument failure and to allow maintenance to be performed at a later time when radiation levels may be lower. Instrument and sensing lines are provided with flushing capability and are routed to minimize radioactive crud buildup. Backflushing capability exists for reactor vessel sensing lines. Tanks containing two phase fluids are fitted with probe type instruments.

   Instrument and sensing line connections are typically located at or above the piping midplane to avoid corrosion product buildup.

(Historical Information)

9. Valves - To minimize personnel exposures from valve operations and maintenance, motor operated, air operated, or other remotely actuated valves are used to the maximum extent practicable. Savings in personnel exposures will result from the large reductions of operational exposures compared to those from the maintenance of remote-operator components.

   Whenever practicable, valves are located in valve galleries and are shielded separately from the major components that accumulate radioactivity. Long runs of exposed radioactive piping are minimized in valve galleries. In areas where manual valves are used on frequently operated radioactive process lines, radiation shielding is provided to minimize personnel exposure. No valves or other active components are placed in radioactive pipe chases.

   For equipment located in high radiation zones, remote actuators are provided for frequently operated valves associated with system operation. All other valve operations are either infrequent or are performed when the equipment is not operating.

   Provisions are made in many radiation areas to drain and flush adjacent radioactive components when maintenance is required on valves.

(Historical Information)

Wherever practicable, valves for clean, non-radioactive systems are separated from radioactive sources and are located in readily accessible areas. Vent, drain, and instrument root isolation valves on radioactive systems are located close to the process piping or equipment with which they are associated. This minimizes the lengths of piping carrying process fluids when these valves are closed. These valves are generally manually operated since they are used infrequently.

Where possible, valves required for normal operation and shutdown are not located in filter and demineralizer compartments.

For large valves (2-1/2 inches and larger) in lines carrying radioactive fluids, a double set of packing with a lantern ring is usually provided. Full ported valves are used in systems containing radioactive slurries.

For the Radioactive Off-gas System, zero stem leakage valves are used to minimize the spread of airborne radioactive contamination.

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

10. Floor drains - Floor drains and sloped floors are provided for each room or cubicle that has serviceable components containing radioactive liquids. Whenever practicable, drain lines are embedded in concrete floors to provide shielding. If a radioactive drain line must pass through a radiation zone lower than that at which it will terminate, proper shielding is provided. Local gas traps or porous seals are not used for controlled area floor drains. Gas traps are provided at the common sump or tank. Some floor drains are designed to prevent

(Historical Information)

potential backflooding. Whenever practicable, provisions exist to remove plugging in the drain piping. Separate trains of floor drains are laid out for cubicles housing radioactive components from the general personnel corridors and work areas. This minimizes the potential for spreading surface and airborne contamination. The floor drain system layout is discussed in Section 9.3.3.

11. Equipment drains - Drains from equipment carrying radioactive fluids are piped in a hard and closed fashion to the Clean Radwaste (CRW) Drain System. The equipment drainage system is discussed in detail in Section 9.3.3.

12. Radioactive tank vents - Vents are provided for all tanks carrying radioactive waste that are also expected to house radioactive gases. Vents are piped in a closed system to a vent header for collection, filtering, and monitored release of gaseous radioactivity. The radioactive tank vent system is described in Section 11.3.

13. Lighting - Multiple electric lights are provided for each cell or room containing highly radioactive components so that the burnout of a single lamp does not require entry and immediate replacement of the defective lamp. Sufficient illumination remains available. Lighting in a radioactive area is actuated from outside the area, and long-life bulbs are used. Section 9.5.3 describes the lighting system.

14. HVAC - The Heating, Ventilating, and Air Conditioning (HVAC) System is designed to minimize buildup of radioactive contamination and provides easy access and fast replacement of the filter elements. Filter banks and components are separated from adjacent banks and components. Section 12.3.3 provides additional description of the HVAC radiation protection design features.

15. Sample stations - Sample stations for routine sampling of process fluids are located in accessible areas. The locations of these sample stations are such that piping and tubing runs are minimized. Shielding is provided at the local sample stations as required to maintain radiation zoning in proximate areas and minimize personnel exposure during sampling. For the unlikely event of a design basis accident (DBA), a post-accident sample station is provided in compliance with NRC regulations set down in NUREG 0737. The station is sufficiently shielded to allow post-accident sampling, sample transport to the onsite facility, and analysis without receiving radiation exposures above 5 rem for the whole body and 75 rem for any extremity. This complies with NRC regulations delineated in NUREG-0737. For a detailed discussion on process and post-accident sampling, see Section 9.3.2. The general layout of the sample stations is shown on Plant Drawings M-23-0 and M-38-0.

16. Clean services - Wherever practicable, clean service components such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipeways. In addition, active components of these clean services are located outside radiation areas wherever possible, to minimize any radiation exposure associated with maintenance of clean system components.

17. Electrical and electronic equipment - Equipment serving electrical or electronic needs is placed in low radiation areas whenever possible. This equipment is non-radioactive, and full access is provided with a minimum of radiation exposures for its surveillance, inspection, maintenance, and repair.

18. Counting room and laboratories - The counting room and radiochemical laboratory facilities are described and discussed in Section 12.5.2.

(Historical Information)

12.3.1.2 Common Facility and Layout Designs for ALARA

This section describes the design features used for standard plant process equipment and layout situations for radioactive systems and for potentially radioactive systems. These features are used in conjunction with the general equipment designs described in Section 12.3.1.1.

1. Valve galleries - Valve galleries are provided with shielded entrances for personnel protection. In many cases, the valve galleries are divided by shielding or distance into subcompartments that service only a few components so that personnel are exposed only to the valves and piping associated with a minimum of components at any given location. Process isolation valves are located close to wall penetrations. Floor drains are provided to collect radioactive leakage. The local areas within the vicinity of the valve gallery areas are provided with decontamination surface coatings. As a minimum, these areas are provided with decontamination surface base (one foot from the floor) and/or decontamination Wainscoat (six feet from the floor). This feature makes it easy to decontaminate surfaces exposed to possible leakage from valves and other components.

2. Piping - Each piping run is analyzed to determine the potential radioactivity level and maximum expected surface dose rate. Radioactive pipes are routed separately from non-radioactive pipes where possible to minimize personnel exposure. Pipes carrying radioactive materials are routed through controlled access areas zoned for a corresponding radiation dose level. Where radioactive piping must be routed through corridors or other low radiation areas, pipes are shielded using special lead or steel details or shielded pipeways.

(Historical Information)

Valves, instruments, or active components are not placed in radioactive pipeways. Wherever practicable, each equipment compartment is used only for those pipes associated with equipment in the compartment. This minimizes exposure due to the operation of one system while maintenance is being performed on another system that is shut down.

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Where possible, thermal expansion loops are raised rather than dropped to prevent crud traps. In radioactive systems, the use of non-removable backing rings in the piping joints is prohibited to eliminate a potential crud trap for radioactive materials. Wherever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe. Line size changes are typically made by eccentric reducers. Orifices are placed in vertical lines wherever possible. At strategic locations within some selected piping systems, connections are provided for the use of hydrolasers. High pressure hydrolasers will be used in removing pipe surface contaminations and therefore reducing radiation exposures to personnel during maintenance and other operations.

Piping carrying resin slurries or evaporator bottoms is run using large radius bends wherever possible, instead of elbows, and horizontal runs are minimized. To prevent possible crud buildup, flow control valves and orifices are used only if they are required for system operation. Large diameter piping is typically used with a minimum number of pipe fittings to reduce crud accumulation.

(Historical Information)

3. Field run piping - Most routing of radioactive piping, large and small, is done by the Bechtel engineering office to ensure that the radiation zone routing is proper and that the aforesaid principles are employed. Where the routing of radioactive piping is delegated to the Bechtel field office, ALARA principles are enforced by way of the ALARA Field Design Guide that was originated by the engineering office.

4. Penetrations - Wherever possible, penetrations are located with an offset between the source and accessible areas to minimize radiation streaming. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. If neither of these two methods is used, then alternative means are employed, such as using baffle shield walls or radiation shielding in the area around the penetration.

5. Contamination control - Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. Equipment vents and drains from radioactive systems are normally piped directly to the radwaste collection system instead of allowing any contaminated fluid to flow across the floor to the floor drain. All welded piping is used for radioactive systems to the maximum extent practicable to reduce system leakage and crud buildup at joints. The valves in some radioactive Nuclear Steam Supply Systems (NSSS) are provided with leakoff connections piped directly to the Radwaste Collection System. They are the main stop, control, bypass, and combined intermediate valves of the main turbine, and the stop and control valves for the reactor feed pump turbines.

(Historical Information)

Decontamination of areas expected to become contaminated within the plant is facilitated by the application of suitable smooth-surfaced coatings to the concrete floors and walls of these areas.

Floor drains and sloping floors are provided in all potentially radioactive areas of the plant. In addition, radioactive and potentially radioactive drains are separated from non-radioactive drains.

Systems that become highly radioactive, such as the radwaste slurry transport system, are provided with flush and drain connections, for normal or chemical cleaning prior to maintenance.

6. Equipment layout - In systems where process equipment is a major radiation source, e.g., fuel pool and reactor water cleanup, radwaste, condensate demineralizer, etc, the pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of items in reduced radiation areas. Control panels are located in the lowest dose radiation zones. Redundant equipment is separated from each other, and shielding is provided between the equipment to allow maintenance concurrent with system operation.

   Major components, e.g., tanks, demineralizers, and filters, in radioactive systems are located in individual shielded compartments. For highly radioactive components, e.g., filters and demineralizers, completely enclosed shielded compartments with hatch openings are provided. Provision is made for some major plant components for removal to lower dose radiation zones for maintenance. Large HVAC filter plenums with multiple filter cartridges are provided with external shielding.

(Historical Information)

Labyrinth entryway shields or shielding doors are provided for each compartment from which radiation could shine to access areas and exceed the radiation dose limits for those areas. Adequate space for removal of components is provided.

Plant Drawings N-1031, N-1033, N-1046 and N-1012 provide typical layout arrangements for demineralizers, liquid and gaseous radwaste filters, waste sludge tanks, radwaste evaporators, off-gas recombiners, sample stations, charcoal beds, and their associated valve compartments or galleries.

7. Special provisions for surveillance of radioactive system areas - To minimize plant personnel radiation exposure during routine area surveillances of radioactive systems, several special design features are provided. A number of radiation shielding viewing windows are installed at strategic locations in the shield walls surrounding radiation areas. Such areas as the pump and valve galleries of the radwaste collection and processing systems, the pump and heat exchanger rooms of the Reactor Water Cleanup (RWCU) and Fuel Pool Cooling and Cleanup (FPCC) Systems, and the resin regeneration station are provided with viewing windows. These windows will allow surveillance of active system components from the personnel corridors without the need to enter the radiation areas. For special viewing and remote control purposes, television cameras are provided in areas such as the radwaste drum filling turntable stations, drum conveyor aisle, and the drum storage hayloft, where high radiation fields can be expected, surveillance of the area is necessary, and the presence of individuals is prohibited.

(Historical Information)

Radioactive and non-radioactive systems are normally separated to limit radiation exposure from routine inspection of non-radioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is normally provided. In high radiation areas where routine surveillance is required, remote viewing devices are provided as needed. Typically, equipment and valves in high radiation areas are made easily accessible by providing permanent access platforms, easily removable insulation, etc. Equipment manways are readily accessible. Equipment laydown area requirements are considered in the layout, and adequate space is provided where necessary.

8. Facilities for handling unsealed radioactive material - As discussed in Section 12.2.1.7, special materials used in the radiochemistry laboratory require the design of special storage and handling equipment. For the handling of radioactive unsealed materials, the following are provided:

    a. Exhaust hoods that exhaust to the ventilation system are located in areas such as sample stations and the radiochemistry laboratory.

    b. Sample sinks for the collection and control of radioactive liquids at sample stations and the radiochemistry laboratory.

    c. Decontamination facilities, the radiochemistry laboratory, restricted machine shop, and instrument calibration and repair shop are situated at various locations in the plant and are described in Section 12.5.

(Historical Information)

d. An area for the repair and maintenance of removed control rod drives (CRDs) is provided in the reactor enclosure near the CRD removal hatch.

e. Area and airborne exhaust duct radiation monitor systems alert personnel to any abnormal condition due to the handling of unsealed sources.

12.3.1.3 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard is regulated and controlled. Each radiation area, as defined in 10CFR20, is provided with a personnel alert sign, each high radiation area (10CFR20) with an expected dose rate between 100 mrem/h and 1 rem/h will be conspicuously posted "CAUTION, HIGH RADIATION AREA" OR "DANGER, HIGH RADITION AREA" and controlled with an RWP while high radiation areas with an expected dose rate greater than 1 rem/h is provided with a locked personnel barrier, as shown on Plant Drawings N-1031 through N-1038, N-1041 through N-1047 and N-1011 through N-1016 and described in Section 12.3.2.2.11. Very high radiation areas, as defined in 10CFR20, are conspicuously posted "GRAVE DANGER, VERY HIGH RADIATION AREA" and provided with a locked personnel barrier. Section 12.5 describes both the control of ingress and egress of plant operating personnel to radiologically controlled access areas, and the procedures employed to ensure that personnel exposure is within the limits prescribed by 10CFR20.

All plant areas are categorized into radiation zones according to expected radiation levels. Each radiation zone defines either the highest component dose rate in the area or the radiation level from all contributing sources inside and outside of the area, whichever is higher. Each room, corridor, and pipeway of every plant structure is evaluated for potential radiation sources during normal operation, including anticipated operational occurrences and shutdown. The radiation zone categories used, and their descriptions, are given in Table 12.3-1, and the specific zoning for each plant area during normal operation and plant cold shutdown is shown on Plant Drawings N-1031 through N-1038, N-1041 through N-1047 and N-1011 through N-1016 and Figures 12.3-22 through 12.3-29, and described in Section 12.3.2.2.11. All frequently used areas, such as corridors are shielded for Zone I or Zone II access.

The locations of airborne radioactivity and area radiation monitors are described in Section 12.3.4. As a special radiation protection design feature, plant personnel are able to establish temporary control points. At the entrances to radioactive areas housing equipment expected to need routine maintenance and repair, sufficient space is allocated for placing an approximately 100-ft^2 contamination control area and a step-off pad. In addition to several 55-gallon drums to hold contaminated clothing, a table and chair can be moved in for radiation protection personnel controlling the ingress/egress of personnel working in the radioactive room. These temporary control points control and minimize any spread of contamination.

Section 12.3.2 provides a discussion of post-accident radiation zoning and shielding for vital plant areas, and post-accident emergency activities.

(Historical Information)

12.3.1.4 Control of Activated Corrosion Products

To minimize the radiation exposure associated with the deposit of activated corrosion products in reactor coolant and auxiliary systems, the following steps have been taken:

1. The Reactor Coolant System (RCS) consists mainly of austenitic stainless steel, carbon steel, and low alloy steel components. The nickel metal content of these materials is low, in accordance with applicable ASME material specifications, because nickel metal contains trace amounts of cobalt, and cobalt 60 is one of the predominant radioactive corrosion products.

(Historical Information)

2. Powdex filter demineralizer filtration is employed in the use of the RWCU system. This system is in continuous operation for the cleanup of the reactor water during full power operation, as well as during plant shutdowns. The system is described in Section 5.4.8. This filtration removes activated corrosion products and crud, and helps to keep down the radioactivity level of the reactor coolant.

3. Valve packing materials are selected primarily for their properties for use in the particular nuclear environments.

4. Various flushing, hydrolasing, draining, testing, and chemical addition connections have been incorporated into the design of piping and equipment that handle radioactive materials. These connections are used if corrosion product removal is to be performed.

5. The plant is designed with a full flow, deep bed condensate demineralizer system for the feedwater. This filtration system helps to minimize the buildup of corrosion products and other impurities in the Reactor Coolant System. This system is described in Section 10.4.6.

6. The Condensate Pre-filter system has been added to the plant upstream of the deep bed condensate demineralizers. This system operates at 100% condensate flow to remove insoluble impurities, primarily iron, in order to improve the performance of the deep bed demineralizers and to reduce corrosion products in the Reactor Coolant system. This system is described in Section 10.4.6.6.

12.3.2 Shielding

In this section, the bases for the nuclear radiation shielding design and the shielding layout are discussed. Also discussed are the mathematical methods and geometric models used in the calculations for the radiation shielding design.

(Historical Information)

12.3.2.1 Design Objectives

The basic objective of the plant nuclear radiation shielding is to reduce personnel exposures. In conjunction with a program of controlled personnel access to and occupancy of radiation areas, radiation levels are held within the allowable limits of NRC Regulations 10CFR20 and 10CFR50, and are as low as is reasonably achievable (ALARA), in accordance with Regulatory Guide 8.8. Radiation shielding, as well as the layout and the design of equipment, components, and piping, are considered in ensuring that exposures are kept ALARA during anticipated personnel activities in areas of the plant expected to contain radioactive materials.

Basic plant conditions considered in the radiation shielding design are normal operation at full power, plant shutdowns, and anticipated operational occurrences.

The shielding design objectives for the basic plant conditions are:

1. To ensure that radiation exposures to plant operating personnel, contractors, administrators, visitors, and proximate site boundary occupants are ALARA and within the limits of 10CFR20.

2. To ensure sufficient personnel access and occupancy time for normal anticipated surveillance, inspection, maintenance, repair, and safety-related test operations as required for each plant equipment, valve, instrumentation, and piping area.

3. To reduce potential equipment neutron activation and minimize the possibility of radiation damage to materials and components from neutron, gamma, and beta exposures.

For the unlikely event of a design basis accident (DBA), including a loss-of-coolant accident (LOCA), radiation shielding is provided for vital areas, such as the plant control room, to keep exposures within the guideline values of 10CFR50.67, and also to protect the general public and ensure that their accident exposures remain within the guideline values of limits of 10CFR50.67. For further discussion of the protection of plant vital areas, see Section 12.3.2.2.6.

12.3.2.2 General Radiation Shielding Design

For the protection of individuals, radiation shielding structures are provided to attenuate direct and scattered radiation to less than the upper limits of the radiation zones for each plant area, as shown on the radiation shielding and zoning drawings. For the definitions of radiation zones and personnel radiological controls, see Plant Drawings N-1031 through N-1038, N-1041 through N-1047 and N-1011 through N-1016, Figures 12.3-22 through 12.3-28 and Section 12.3.2.2.11. The minimum shielding requirements for all plant areas are shown on those shielding and radiation zoning drawings. General arrangement of the plant areas, structures, and equipment discussed in this section are also shown on the radiation shielding and zoning drawings.

The material used for most of the plant shielding is ordinary concrete with a minimum bulk density of 147 lb/ft^3. Wherever poured in place concrete is replaced by concrete blocks or other material, the design ensures protection on an equivalent shielding basis, as determined by the radiation attenuation characteristics of the concrete block or the other material selected. High density concrete having a minimum density of 200 lb/ft^3 is used where space limitations do not allow use of ordinary concrete.

In a few cases, special shielding design details use steel and/or lead materials where space limitations or layout complexities make the use of concrete impossible or uneconomical. For a number of entrances to highly radioactive areas, heavy solid steel shield doors are provided. This type of shielding is used only when other means of protection, such as labyrinths, cannot be employed.

Water is also used as a primary shield material for areas surrounding the spent fuel transfer and storage areas.

(Historical Information)

Special features incorporated in the structural layout and design of the plant shielding are introduced to maintain radiation exposures ALARA in routinely occupied areas, such as valve operating stations and pump rooms. These are described in more detail under Radiation Protection Design Features in Section 12.3.1. As discussed in Section 1.8, Regulatory Guide 1.69 is used for Hope Creek Generating Station (HCGS) in the design of nuclear radiation shielding. Regulatory Guide 8.8 is also used. Concrete shielding properties and standards are also discussed in Section 3.8.

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall or floor slab. Therefore, the actual anticipated radiation levels in the greater region of each plant area are less than this maximum dose and are also less than the radiation zone upper limit. The labyrinths are constructed such that the scattered dose rate, plus the directly transmitted dose rate through the shield wall from all contributing sources, is below the upper limit of the radiation zone specified for the particular plant area exposed.

12.3.2.2.1 Drywell and Reactor Building Shielding Design

During reactor operation and shutdown, the reinforced concrete drywell shield wall and the Reactor Building walls protect both personnel occupying adjacent plant structures, and yard areas, from radiation originating in the reactor vessel and associated equipment.

Where personnel or equipment hatches or system penetrations pass through the drywell shield wall, additional shielding is designed to attenuate the radiation to below the required level, as defined by the radiation zones outside the drywell wall during normal full power operation and shutdown. The drywell and Reactor Building shielding also functions as protection during accidents, including LOCAs, and ensures that exposure levels in the vital plant areas remain within limits as defined by 10CFR5067.

(Historical Information)

12.3.2.2.2 Drywell Interior Shielding Design

During plant operation, areas within the drywell are designed for Zone VI and higher levels, and they are inaccessible (See Plant Drawing N-1043). The biological shield, which surrounds the reactor pressure vessel (RPV), provides shielding for access in the drywell during shutdown. It also reduces the activation of, and radiation damage to, drywell equipment and materials during full power operations. Special shield doors are provided for a number of pipe penetrations through the biological shield. When inservice inspections are required during plant outages, these doors allow access to specific locations at the reactor pressure vessel and its pipe nozzles.

12.3.2.2.3 Reactor Building Interior Shielding Design

The drywell shield wall concrete shell structure is designed to reduce radiation levels in normally occupied areas of the Reactor Building, from sources such as the RPV and the main steam lines within the drywell, to less than the maximum level for a Zone II. As defined on Plant Drawings N-1031 through N-1038, N-1041 through N-1047 and N-1011 through N-1016, and Figures 12.3-22 through 12.3-28, this zone level allows fulltime access and occupancy of personnel corridor and general work areas located outside the drywell and inside the reactor building.

Penetrations and hatch openings in the drywell concrete shield wall are shielded, as necessary, to meet adjacent area radiation zoning levels. Shielding requirements for the personnel, equipment, and control rod drive (CRD) removal hatch openings are shown on Plant Drawing N-1043 for the areas numbered 4330, 4322, and 4326, respectively. Drywell piping and electrical penetrations are shielded by providing either local shields within the penetration assembly or a shielded pipe chase. Shielded pipe chase locations and bulk shielding requirements are shown on Plant Drawings N-1043 through N-1046.

(Historical Information)

These pipe chases, with room numbers 4316, 4319, 4321, 4327, 4329, 4402, 4505, and 4509 are designated as radiation areas having Zone III radiation levels and higher during reactor power operation and are provided with personnel access control barriers. Adjacent to certain vent lines below Elevation 100 feet-0 inches, the shield wall grout fill has been excavated to create tunnels without significant shielding. Room 4102, which is adjacent to the shield wall in these areas, is designated as having Zone VII radiation levels during reactor power operations and is provided with personnel access control barriers.

The components of the Reactor Water Cleanup (RWCU) System, most of which are in the Reactor Building and are described in Section 5.4, are located in shielded compartments. The RWCU compartments are designed as areas having Zone V radiation levels and higher and are restricted access areas. Shielding is provided for each piece of equipment in the RWCU system consistent with its calculated maximum activity, as described in Section 12.2, and with the access and zoning requirements of the adjacent areas. This equipment includes:

1. Regenerative heat exchanger
2. Nonregenerative heat exchanger
3. RWCU recirculation pumps and piping
4. RWCU filter demineralizers and precoat and holding pumps
5. RWCU backwash receiving tanks, transfer pumps, and piping.

The Residual Heat Removal (RHR) System pumps and heat exchangers are in operation after reactor shutdown to remove reactor core decay heat. The radiation levels in the vicinity of this equipment may temporarily reach as high as Zone VI levels due to activation and fission products in the reactor water. Shielding is designed to attenuate radiation from RHR equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas.

(Historical Information)

The Reactor Building also houses the Reactor Core Isolation Cooling (RCIC) System and the High Pressure Coolant Injection (HPCI) System. These systems require periodic functional tests. They use the radioactive main steam for the pump turbine drive. Shielding protects the surrounding areas according to applicable access requirements.

Since HCGS does not have a cattle chute design shield, actions will be taken such as administrative controls, remote monitoring, and physical barriers to preclude access to the drywell during fuel movement. However, personnel may be permitted limited access to the lower levels for necessary work during refueling operations.

One of the primary radiation sources inside the reactor building is the spent fuel assemblies. Spent fuel transfer and storage is performed underwater in the fuel transfer canal and in the spent fuel storage pool. Water and concrete shielding are provided for areas surrounding the fuel transfer canal and the storage pool, to ensure that radiation levels remain below zone levels specified for adjacent areas.

A portable, shielded fuel transfer chute also is installed in the reactor cavity during refueling operations to provide additional shielding to upper drywell areas.

Water is also used as shielding material for the steam dryer and separator storage pool. Concrete walls surrounding the pool and water in the pool are designed to provide Zone II dose rates in adjacent accessible areas during storage of the dryer and separator.

The Fuel Pool Cooling and Cleanup (FPCC) System shielding is based on the maximum activity discussed in Section 12.2 and the access and zoning requirements for adjacent areas. Equipment in the FPCC system to be shielded includes the FPCC heat exchangers, fuel pool filter/demineralizers, pumps, and piping.

The spent fuel shipping cask loading pit is designed and shielded for the safe transfer of spent fuel assemblies from the storage pool into the cask. The assemblies are transferred underwater, in order to protect surrounding floor areas to Zone II levels.

The traversing in-core gamma probe (TIP) system drive and housing components are located inside a shielded compartment to protect personnel from the neutron activated portions of the TIP cables.

Concrete shielding is provided for the section of the main steam tunnel that is located within the Reactor Building. The high energy gamma radiation emanating from the main steam is reduced by shielding to allow full personnel access in adjacent areas.

For the CRD removal, repair and decontamination area shielding is provided to protect the adjacent personnel work and corridor areas. Special lead shielding caps are provided on the CRD spuds.

12.3.2.2.4 Radwaste Areas Shielding Design

In the radwaste areas where gaseous, liquid, and solid radioactive wastes are collected, stored, and processed, shielding is provided to ensure that the radiation zoning and access requirements are met for areas occupied by plant personnel.

The following major equipment, which contains and handles liquid radwaste, is placed in individual shielded cubicles, as shown on Plant Drawings N-1031, N-1032, N-1037, N-1033 and N-1034:

1. Decontamination solution concentrated waste tank

2. Radwaste regeneration vessel

3. Radwaste resin holding tank

4. Concentrated waste tanks

5. Waste neutralizer tanks

6. Waste surge tank

(Historical Information)

7. Floor drain sample tanks

8. Waste sample tanks

9. Waste collector tanks

10. Cleanup phase separators

11. Spent resin tank

12. Waste sludge phase separator

13. Chemical waste tank

14. Decontamination solution concentrator

15. Floor drain collector tanks

16. Detergent drain tank

17. Waste evaporators

18. Fuel pool filter/demineralizers

19. Waste and floor drain filters

20. Waste and floor drain demineralizers.

The pumps, valve manifolds and instrumentation, and general piping associated with the equipment listed above are also shielded in cubicles, galleries, and pipeways, respectively.

The following major equipment, which contains and handles gaseous radwaste, is placed in individual shielded cubicles, as shown on Plant Drawings N-1031 and N-1032:

(Historical Information)

1. Feed gas cooler condenser

2. Preheater

3. Recombiner

4. Offgas holdup piping

5. Passive glycol cooler condenser

6. Charcoal guard bed

7. Charcoal adsorber vessels

8. High efficiency particulate air/absolute (HEPA) filters.

Valves, instruments, and general piping associated with the aforesaid equipment are also shielded in cubicles, galleries, and pipeways.

For the solidification, drumming, storing, and shipping of liquid and solid radioactive wastes, the following major equipment is placed in shielded individual cubicles, as shown on Plant Drawings N-1033, N-1034 and N-1035:

1. Centrifuge feed tank

2. Waste centrifuge

3. Vapor body

4. Extruder evaporators

5. Entrainment separator

6. Concentrated bottoms tank

(Historical Information)

7. Solid waste compactor

8. Drum capping, swiping, and conveyor equipment.

The storage and shipping areas for the solidified waste drums are heavily shielded. The pumps, valves and instruments, and general piping associated with the major equipment listed above, are also shielded in cubicles, galleries, and pipe chases, respectively.

12.3.2.2.5 Turbine Building Shielding Design

Radiation shielding is placed around the following major equipment inside the Turbine Building in order to comply with zone access requirements, shown on Plant Drawings N-1011 through N-1016, and exposure restrictions for adjacent areas:

1. Turbine condensers

2. Turbine hotwells

3. Primary condensate pumps

4. Steam jet air ejectors

5. Off-gas feed piping

6. Steam packing exhauster condenser

7. Seal water coolers

8. Mechanical vacuum pump

9. Condensate demineralizers

10. Cation and anion regeneration vessels

(**Historical Information)**

11. Resin mix and hold tank

12. Ultrasonic cleaning equipment

13. Feedwater heaters

14. Reactor feed pump turbines

15. Steam seal evaporators

16. Moisture separators

17. High and low pressure turbines

18. Main steam piping.

19. Backwash receiving tank, tipped HIC and transfer pumps of the condensate pre-filter system

Shielding is also provided at the turbine building operating floor because of the relatively close location of Zone I areas. These areas are offices, technical document and training facilities, quality engineering (QE) and conference rooms, lobbies, and toilet facilities that are designed for unlimited personnel access.

12.3.2.2.6 Post Accident Shielding Design and Access Review

A post accident shielding and access review was performed to ensure the accessibility of vital areas in which personnel will or may be present to perform mitigation or monitoring functions during post accident operations.

1. Source terms – Following a postulated accident where substantial core damage has occurred, radioactive materials will be released from the fuel due to fuel rod cladding failure.

2. Not Used

3. Airborne sources in the Auxiliary and Turbine Buildings – The transport pathway of the airborne sources in the Auxiliary Building (control and diesel generator areas, and service areas), and Turbine Building consists of leakage:

   a. from the drywell to the reactor building, and discharge to the environment through the Filtration, Recirculation, and Ventilation System (FRVS)

   b. from engineered safety feature components outside the primary containment, and discharge to the environment through the FRVS

   c. from the main steam isolation valves (MSIVs), and discharge to the environment from the Turbine Building

   The airborne activity then re-enters the buildings through the ventilation intake systems after dilution in the atmosphere. The fission product release to the environment is described in Section 15.6.5.5. The atmospheric dispersion factors for the airborne transport pathways to the control room emergency intake are given in Table 6.4-2. The atmospheric dispersion factors used in evaluating the total integrated doses at all other locations within the building wake cavity are given in Table 12.3-10.

4. Not Used

5. Vital areas - Vital areas are defined in NUREG-0737, item II.B.2, as those "which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident." The post accident vital areas for Hope Creek are listed in Table 12.3-3.

6. Personnel radiation exposure guidelines - The general design basis for personnel radiation exposure guidelines is 10CFR50.67. The maximum allowable radiation dose to personnel shall not exceed 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

   The dose received by personnel in vital areas of continuous occupancy should be <15 mrem/h TEDE (average over 30 days). The dose rate for these areas is determined using the control room occupancy factors contained in Regulatory Guide 1.183, i.e., 1.0 for 0-1 day; 0.6 for 1-4 days; and 0.4 for over 4 days.

   The dose received by personnel in an infrequent occupancy of vital areas is determined by taking into account the frequency and duration of the activities anticipated for that area, and is consistent with 10CFR50.67 limits. Average area dose rates are used to determine personnel exposure, although local hot spots may exist.

7. Not Used.

(Historical Information)

12.3.2.2.7 Diesel Generator Areas Shielding Design

The diesel generator areas are part of the control and diesel generator complex of the Auxiliary Building. There are no radiation sources in the diesel generator areas. Therefore, no shielding is required for these areas from internal sources. For post accident access, these areas have sufficient shielding to be protected from drywell and Reactor Building radiation shine.

12.3.2.2.8 Deleted

(Historical Information)

12.3.2.2.9 Counting Room Shielding Design

The plant counting room, which is part of the radiochemistry laboratory facility in the Auxiliary Building, has received special attention with regard to radiation shielding. Because the counting room contains instruments very sensitive to radioactivity, it is imperative that the background radiation levels be minimized. To accomplish this, no flash is used in the concrete mix for the walls and slabs surrounding the counting room. Flash normally contains a relatively large amount of slow decaying natural radioactive isotopes. In addition, the shield walls and slabs are sized to maintain a background radiation level of less than 130 mrem/year for anticipated operational occurrences and 45 mrem/year for normal operation.

12.3.2.2.10 General Plant Yard Areas Shielding Design

All general plant yard areas are at Zone I radiation levels, and are, therefore, fully accessible during all phases of normal full power operation. Only the condensate storage tank (CST) area is an exception, having a Zone II designation. This area is surrounded by a wall shielding the adjacent Zone I plant yard areas.

12.3.2.2.11 Low Level Radwaste Storage Facility Shielding Design

Shielding is provided to ensure that the radiation zoning and access requirements are met for areas occupied by plant personnel. Shielding is designed to provide Zone I dose rates outside the facility when DAW boxes are being moved and/or when no crane operations are taking place. When a HIC or Pallet is being moved with the crane, dose rates outside the facility will fall within Zones III, IV or V depending on the contact dose rate on the containers. The DAW storage area is designated Zone IV and the vault storage area is designated Zone VIII. The control room is designated Zone II. The truck bay is normally Zone II or III but will be higher when waste is being moved.

During periods when offsite disposal or storage is available and radwaste is not stored in the facility, zone designations will be assigned as appropriate for any alternate use of the facility.

(Historical Information)

12.3.2.3 Shielding Calculational Methods and Geometry Models

The shielding design of the structures provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure in accordance with ALARA is based on maximum equipment activities under the plant operating conditions described in Section 12.2. The thickness of each shield wall surrounding radioactive equipment is determined by approximating the actual geometry and physical conditions of the source or sources as closely as possible. The isotopic concentrations are converted to energy group sources using data from standard

references that are accepted and widely used by the nuclear industry. This is from material derived from References 12.3-3 through 12.3-5.

The geometric model assumed for the shielding evaluation of tanks, heat exchangers, filters, demineralizers, and evaporators is a finite cylindrical volume source. For the shielding evaluation of piping, the geometric model is a finite or infinite shielded cylinder. In cases where radioactive materials are deposited on cylindrical component surfaces, e.g., pipes, this is treated as an annular cylindrical surface source. Slab source, truncated cone, or flat surface source geometries, either finite or infinite, are employed in shielding analyses for extended flat face radiation sources, such as are encountered with the box type storage arrangement of spent fuel assemblies. Point source geometry is employed for highly concentrated sources having relatively small dimensions. Examples are component crud hot spots or instrument calibration sources.

Sources having large dimensions are encountered in the analyses of post accident fission product gas clouds. The analyses in this case use infinite cylindrical or spherical source geometries. Typical computer codes that are used for shielding analysis are listed in Table 12.3-4. Shielding attenuation data used in these codes include gamma attenuation coefficients, found in Reference 12.3-16; gamma buildup factors, found in Reference 12.3-17; neutron-gamma multigroup cross sections, found in Reference 12.3-18; and albedos, mentioned in Reference 12.3-19.

Where shielded entryways to compartments containing high radiation sources are necessary, labyrinths or mazes are designed using a general purpose gamma-ray scattering Code G-33, mentioned in Reference 12.3-15.

(Historical Information)

12.3.3 Ventilation

The plant ventilation systems are designed to provide a suitable environment for personnel and equipment during normal plant operation and anticipated abnormal occurrences. Detailed Heating, Ventilating, and Air Conditioning (HVAC) System descriptions, including compliance with and exceptions to Regulatory Guides 1.52 and 1.97, are provided in Sections 1.8 and 9.4. Control room habitability is discussed in Section 6.4. Instrumentation and controls for Engineered Safety Feature (ESF) HVAC Systems are discussed in Section 7.3. Section 7.4 discusses the safe shutdown HVAC system.

12.3.3.1 Design Objectives

The ventilation systems are designed to maintain inplant airborne activity levels within the limits of 10CFR20 in personnel access areas and also operate to prevent the spread of airborne radioactivity during normal plant operation and anticipated abnormal occurrences.

During post accident conditions, the ventilation systems for the main control room (MCR) and technical support center (TSC) help to provide a suitable environment for personnel and equipment to ensure continuous occupancy in these areas, except for a fire/smoke condition, when personnel are to be evacuated. If smoke is detected in the outside air intake duct to the MCR or TSC, the affected system can be manually switched over to the recirculation mode. The plant ventilation systems are designed to comply with the airborne radioactivity release limits for offsite areas during normal and abnormal plant operations.

12.3.3.2 Design Criteria

Design criteria for the plant ventilation systems include the following:

(Historical Information)

1. During normal operation and anticipated abnormal occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in restricted areas of the plant are as low as is reasonably achievable (ALARA) and within the limits specified in 10CFR20.

   The average and maximum airborne radioactivity levels in unrestricted areas of the plant during normal operation and anticipated abnormal occurrences are ALARA and within the limits of 10CFR20.

2. During normal operation and anticipated abnormal operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary are ALARA and within the limits specified in 10CFR20 and 10CFR50.

3. The plant site dose guidelines of 10CFR100 are satisfied following those hypothetical accidents, described in Section 15, which involve a release of radioactivity from the plant.

4. The dose to MCR and TSC personnel must not exceed the limits specified in GDC 19 of Appendix A to 10CFR50, following those hypothetical accidents, described in Section 15, which involve a release of radioactivity from the plant.

12.3.3.3 Design Guidelines

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

To accomplish the design objectives, the following guidelines are used in plant design to minimize airborne radioactivity:

(Historical Information)

1. Access control and traffic patterns are considered in the basic plant layout to minimize the exposure to contamination.

2. Radioactive vents and drains from equipment and piping are piped directly to the collection system, instead of allowing contaminated fluid to flow openly across the floor to the floor drain.

3. All welded piping is used on contaminated systems to reduce system leakage. See Section 11.1.5 for additional discussion.

4. Metal diaphragm or bellows seal valves are used in those radioactive piping systems where no leakage can be tolerated.

5. Coatings are applied to the concrete floors and walls of potentially contaminated areas to facilitate any decontamination.

6. Potentially contaminated equipment has design features that minimize the potential for airborne radioactive contamination during maintenance operations. These features include flush connections on pump casings, for draining and flushing the pumps prior to maintenance, and flush connections on piping systems that could be highly radioactive.

7. Exhaust ventilation is used in plant areas to direct any airborne radioactive contaminants away from the personnel breathing areas and into the ventilation filtering systems. High efficiency air filters are used in all supply air systems to minimize particulate inventory within the plant.

8. Concrete barrier walls are provided for shielding of radioactive process equipment and piping.

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

To accomplish the design objectives, the following guidelines are used in ventilation design to control airborne radioactivity:

1. Airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination.

2. For building compartments that have a potential for contamination, either a greater volumetric flow is exhausted from the compartment than is supplied, or the entire makeup for the exhaust air from the compartment is induced from the surrounding less contaminated areas.

3. The radwaste area tanks that contain radioactive materials are vented to a negative pressure collection header and the vapors are processed by the tank vent filter system.

4. In areas where airborne radioactivity is present, exhaust air is processed through prefilters and high efficiency particulate air (HEPA) filters, or a combination of prefilters, HEPA, and charcoal filters, where necessary, before release to the atmosphere to reduce onsite and offsite airborne concentrations. Once-through ventilation systems are used in the radwaste area and reactor building with all exhaust air passing through prefilters and HEPA filters before being discharged to a vent stack. Further description of filtration is given in Tables 12.3-5 through 12.3-8.

5. Redundant, Seismic Category I systems or components are provided for ventilation systems required for safe shutdown of the reactor and for mitigation of the consequences of design basis accidents (DBAs).

(Historical Information)

During refueling operations in the reactor building, at least two of three supply and exhaust fans (three 50 percent capacity supply and three 50 percent capacity exhaust) are operated, and manual duct dampers are repositioned to provide increased ventilation capacity.

6. In the radwaste area systems, the supply and exhaust fans are interlocked to ensure exhaust fan operation prior to supply fan operation.

7. Individual areas can be isolated upon indication of contamination to prevent the discharge of contaminants to the environment.

8. Redundant containment isolation dampers are installed at ventilation duct penetrations in accordance with 10CFR50, Appendix A, GDC 54 and 56, including valve controls, to ensure that the containment integrity is maintained. See Section 6.2.4 for additional discussion.

9. Tornado dampers are provided in outside air intake and exhaust ducts in safety-related HVAC systems, where it is necessary to protect HVAC equipment, ductwork, and process equipment in the event of a tornado.

10. The Reactor Building Ventilation System (RBVS) supply air quantity is adjusted so that a negative building pressure is maintained during normal operation.

11. Before the drywell and suppression chamber are purged by the RBVS, the containment atmosphere is recirculated through the Containment Prepurge Cleanup System (CPCS) to reduce atmospheric radioactivity as required.

(Historical Information)

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from Ventilation Equipment

To accomplish the design objectives, the following guidelines are used to minimize personnel exposure from potentially contaminated ventilation equipment:

1. Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel exposure time. The ventilation systems are designed to allow rapid replacement of deteriorated components, inplace testing operations, and use of material-handling facilities.

2. Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts.

3. Access to and service of ventilation systems in potentially radioactive areas are facilitated by equipment location as follows:

    a. The outside air supply units and building exhaust equipment are enclosed in ventilation equipment rooms. These equipment rooms are located in radiation Zone II and are accessible to the operators. The location of charcoal filter adsorber units complies with the requirements of Regulatory Guides 1.52 and 1.140, to the extent discussed in Sections 6.5.1 and 9.4, respectively.

    b. Local cooling equipment is located in low radiation zones areas whenever possible.

4. Radioactivity buildup in any filter train is minimized by continuous monitoring of the filters. Further internal decontamination of the filter housing is facilitated by using local electrical outlets for operation of decontamination equipment.

(Historical Information)

5. HEPA filters are designed and tested to be free of bypass leakage when clamped in place against compression seals. Each filter housing is designed and tested to be airtight with bulkhead doors that are closed against compression seals.

6. Potentially contaminated filter equipment is shielded to reduce exposure to personnel and equipment in surrounding areas.

7. Particulate filters are changed when required by the indicated pressure drop across the filter bank. Charcoal adsorbers are changed when required by the residual adsorption capacity of the bed. Removable test canisters are located in the carbon bed. The testing of the carbon adsorbers and all other components is described in Sections 6.5.1 and 9.4.

12.3.3.4 Design Description

The ventilation systems serving the Reactor Building, Auxiliary Building radwaste area, and Turbine Building are assumed to be potentially radioactive, and are discussed in detail in Section 9.4.

Although the MCR and TSC are considered to be non-radioactive areas, radiation protection is provided to ensure habitability, as described in Section 6.4.

Ventilation system design parameters are given in Tables 12.3-5 through 12.3-8.

Typical shielding arrangement for the potentially radioactive charcoal and HEPA filter units is shown on Figure 12.3-58.

12.3.4 Area Radiation and Airborne Radioactive Materials Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Design

12.3.4.1.1 Design Bases

The Area Radiation Monitoring System (ARMS) is provided to supplement the personnel and area radiation survey provisions of the plant radiation protection program, described in Section 12.5, to ensure compliance with the personnel radiation protection guidelines of 10CFR20, 10CFR50, 10CFR70, and Regulatory Guides 8.2, 8.8, and 8.10.

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

1. Immediately alert plant personnel entering or working in background or low radiation areas of increasing or abnormally high radiation fluxes that could result in inadvertent exposures if unnoticed.

2. Inform the control room operator of the occurrence and approximate location of an abnormal radiation flux increase in background or low radiation areas, and provide a continuous record for operating and health radiation protection.

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

4. Alarm by means of three area monitors, which are located near the new and spent fuel storage areas.

5. Assist, in general, in maintaining personnel exposures as low as reasonably achievable (ALARA).

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

Training and qualification of radiation protection, and other personnel assigned to use and maintain the ARMS, follow ANSI/ANS-3.1-1981 guidance.

12.3.4.1.2 Criteria for Area Monitor Selection

The following design criteria are applicable to the Area Radiation Monitoring System:

1. Rangeability - There are five decades of range with the high alarm setpoint preferably not lower than the second decade, and not higher than the fourth decade, and set at the maximum allowable exposure rate for the area being monitored. The lower range limit is either natural background or one decade below the normal operating level of each particular area.

2. Overrange Response - The system continues to read off-range upscale if exposed to radiation levels above maximum range.

3. Sensitivity - The system is sensitive to gamma energies of 80 keV and above.

4. Response - In any range, the readout indicates at least 90 percent of its endpoint reading within 5 seconds after a step change in radiation flux at the detector.

5. Energy Dependence - The indicated exposure rate mrem/h at the local indicating unit (LIU) is within 20 percent of

the actual dose rate in each detected area for gamma energies between 80 keV and 2.5 MeV.

6. Environmental Dependence - The system meets the above requirements for all variations of temperature, pressure, and relative humidity within each area delineated in Section 3.11.

7. Exposure Life - Each detector maintains its characteristics up to an integrated exposure of 10^5 rads.

8. Electromagnetic Interference (EMI) - EMI from any source in the operating environment will not disturb the meter indication at the (LIU).

9. Accuracy - The value of radiation flux indicated in the main control room will be ~ 2 percent of the value indicated at the LIU.

10. Drift - The drift of any detector indication during one year will be no more than 2 percent of the actual value in the midranges.

12.3.4.1.3 Criteria for Location of Area Monitors

Area radiation monitors are provided in areas to which personnel normally have access, and in which there is a potential for personnel to receive radiation doses in excess of 10CFR20 limits in a short period of time because of system failure or improper personnel action. The detectors are wall mounted and are located so that the flux measurement is as representative as possible of the area. The detectors are designed and manufactured to be suitable for their locations. Any plant area that meets one or more of the following criteria is monitored:

1. Zone II areas where personnel could otherwise unknowingly receive high levels of radiation exposure because of process system failure or personnel error

2. New fuel storage: two detectors are installed to provide area radiation monitoring

3. At HCGS, area monitors are provided in accordance with GDC 63 of 10CFR50 Appendix A.

Acceptance Criteria II.B.17 of standard review plan 12.3 - 12.4 provides criteria for the establishment of locations for fixed continuous area gamma radiation monitors. The specific document referenced is ANSI/ANS-HPSSC-6.8.1-1981. The locations and numbers of monitors used at HCGS are not in full compliance with this standard. The location of these monitors is in the vicinity of personnel access areas only. These locations are based on the dose assessment and operating experiences from other nuclear power plants. In addition, these locations were finalized prior to the issuance of this standard and provide an acceptable method of monitoring area radiation levels.

Acceptance Criterion II.4.b.3 requires ventilation monitors to be placed upstream of the high efficiency particulate air (HEPA) filters. Discussion on the ventilation monitors is provided in Section 12.3.4.2.2.

Acceptance Criterion II.4.a.3, of the standard review plan 12.3 - 12.4, provides criteria for on-scale readings of dose rates for normal and anticipated operational occurrences and accidents. The on-scale reading ranges designed for the monitors at HCGS do not in all cases comply with this standard. The general area and airborne ventilation monitors do not have post accident functions. Only a few selected monitors, located in vital areas such as the plant control room and the technical support center, have post accident functions and are designed in compliance. (Also see Section 11.5.)

Acceptance Criteria II.4.a.8 and II.4.b.7 of the standard review plan 12.3 - 12.4 provide criteria for emergency power. The HCGS

design is not in compliance with the standards. None of these monitors in question have any safety-related functions and are not on emergency power.

Acceptance Criteria II.4.e of Standard Review Plan 12.3-12.4 identifies the requirements to provide instrumentation capable of monitoring accidental criticality in accordance with the requirements of 10CFR70.24(a)(1), Regulatory Guide 8.12 and ANSI Standard N16.2. Based upon NRC evaluation of the information presented in the Hope Creek Special Nuclear Material (SNM) License Application dated May 23, 1985, PSE&G has been granted exemption to 10CFR70.24 as documented in the Hope Creek SNM License No. 1953 dated August 21, 1985. When the Special Nuclear Material License expired, the exemption conditions were incorporated into the Operating License in SSER 5. These conditions are specific to GE fuel only. Alternately, 10CFR50.68 can be used to demonstrate compliance with 10CFR70.24. both of these approaches eliminate the need for the instrumentation since criticality is not credible.

12.3.4.1.4 System Description (Area Radiation Monitoring)

The Area Radiation Monitoring System detects, measures, and indicates ambient gamma radiation fluxes at various locations in the plant. It also provides audible and visual alarms in areas monitored, and in the main control room if the gamma radiation exceeds a specified limit. Local indicator units (LIUs) indicate the gamma flux in the area monitored. An indication is provided by a main control room annunciator if there is an alarm or a malfunction in any area monitor.

Each area radiation monitoring channel consists of a detector and LIU that provide radiation flux indication and alarm at or near the detector location. Radiation indication and trip status are indicated on the CRT of the display/keyboard/ printer (DKP) and indicated on the CRIDs and SPDS CRTs located in the main control room. The

area radiation monitor provided in the main control room has no local alarm unit because it is included in the Radiation Monitoring System (RMS) control room annunciator. The detector data also are included in the RMS computer data base. An ARMS functional block diagram is shown in Figure 12.3-59.

The ARMS is a part of the plant RMS and the detector data are analyzed, processed, displayed, and recorded, as described in Section 11.5. Each detector channel has three alarms: high, alert, and low (or failed). The failure alarm is adjustable and is activated if the detector, high voltage, signal, or power source fail. The high and alert setpoints are adjustable, and the local audible and visual alarms (at or near the detector) are actuated when the radiation flux exceeds the setpoint. The alarm setpoints can be changed only from the DKP or local radiation processor (LRP) and only after password/keylock entry. The LIUs have meter readouts with a five decade logarithmic scale that indicates the flux at the detector in mR/h. All channels have a five-decade range as appropriate for their detector locations, and as specified in Table 12.3-9.

With the exception of the Geiger-Muller detector tubes, all electronics are solid state, and the system is designed for high reliability. All ARMS detectors are independent, and failure of one detector has no effect on any other. However, some LRPs may have more than one ARMS detector connected, and failure of the LRP will result in the loss of data from all of the detectors connected to it.

The location of each area radiation detector is indicated on instrument location drawings and is listed in Table 12.3-9. The detectors can be removed from the wall and can be used as portable probes within the limit of the 25-ft detector cable to locate radiation sources. Consistent with the criteria above, the following general areas are monitored:

1. Main control room

2. Radwaste building corridors and processing areas

3. Restricted machine shop

4. Fuel storage and handling area

5. Containment hatchways and airlocks

6. Sampling stations

The ARMS data processing, display, and control functions are in the CRP and DKP, which are on the plant instrument bus that is powered from a battery through an inverter. The field located detectors, LIUs, and LRPs are powered from the plant instrument bus that is powered from a battery through an inverter. No part of the ARMS is on emergency power.

12.3.4.1.5 Safety Evaluation

The ARMS is not essential for safe shutdown of the plant, and it serves no active emergency function during operation. The ARMS does serve to warn plant personnel of high radiation levels in various plant areas. All ARMS detectors and LIUs are independent, and failure of one detector/LIU has no effect on any other.

The ARMS is designed to operate unattended for extended periods of time, detecting and measuring ambient gamma radiation. Ambient radiation exposure rate at the detector is indicated remotely in the main control room on the DKP and also at or near the detector. These monitors cause an audible and visual alarm at the detector, and in the main control room, if the radiation levels exceed preset limits.

The ARMS has no post accident monitoring function. The Drywell Atmosphere Post-Accident Radiation Monitoring System (DAPA RMS)

detectors are listed on Table 12.3-9 for information only and are discussed in Section 11.5.2.1.5.

The ARMS is powered from battery backed 120 V ac uninterruptible instrument bus power and is not affected by a loss of offsite power.

12.3.4.1.6 Calibration and Testing

Each of the ARMS monitors is calibrated by the instrument manufacturer prior to shipment using sources certified by or traceable to the National Bureau of Standards (NBS). Inplant calibration uses a standard radioactive point source traceable to NBS.

The proper functioning of each ARMS monitor is verified periodically by checking instrument response to the remotely operated radioactive check source provided with each detector. The check source is located inside the detector and can be actuated from the DKP or the LRP. The trip point settings are tested by manually entering simulated data and observing the channel operation. All trips latch and must be reset manually.

12.3.4.2 Airborne Radioactive Materials Monitoring

12.3.4.2.1 Samples Taken by Hand for Laboratory Analysis

Concentrations of airborne radioactive materials are determined by routine laboratory analysis of grab samples taken throughout the plant. Filter papers for particulates and charcoal cartridges for halogens are used in low volume samplers. Filter papers for particulates are used in high volume samplers. Both ventilation exhaust ducts and work areas are sampled.

12.3.4.2.2 Sampling Systems with Monitors

Refer to Section 11.5.2.2 for additional information

The ventilation systems that exhaust to the environment, north plant vent, south plant vent, and Filtration, Recirculation, and Ventilation System (FRVS) are monitored by taking a representative sample with an isokinetic probe at rates that follow the exhaust flow. The sample transport tubing is heat traced to prevent condensation. The north and south plant vent particulate filters and noble gases are monitored continuously with beta radiation detectors. The north and south plant vent halogen charcoal cartridges are monitored continuously by gamma detectors.

During operation of the FRVS, only noble gases are monitored continuously with beta radiation detectors. The particulate filter and halogen charcoal cartridge are removed periodically for laboratory analysis onsite. Some of the ducts that are tributary to the normal effluent release stacks are monitored by beta radiation detectors that are inserted into the vertical wall of the duct at appropriate places. These tributary monitors will assist in locating sources of airborne radioactive materials. Sampling taps are located in the ducts next to the detectors so that grab samples can be taken.

Additional mobile samplers with monitoring detectors are provided for use if needed.

The above described airborne radioactive material monitoring equipment and procedures are used to meet the applicable parts of Regulatory Guides 1.21, 1.97, 8.2, 8.8, and ANSI N13.1-1969.

Acceptance Criteria II.B.17 of standard review plan 12.3-12.4 provides criteria for the establishment of locations for fixed continuous area gamma radiation monitors. The specific document referenced is ANSI/ANS-HPSSC-6.8.1-1981. The locations and numbers of monitors used at HCGS are not in full compliance with

this standard. The locations of these monitors are in the vicinity of personnel access areas only. These locations are based on the dose assessment and operating experiences from other nuclear power plants. In addition, these locations were finalized prior to the issuance of this standard and provide an acceptable method of monitoring area radiation values.

Acceptance Criterion II.4.b.3 requires ventilation monitors to be placed upstream of the HEPA filters. The HCGS design places scintillation detectors in ducts that are tributary to the release vent in order to provide warning of increased releases within the plant. These instruments detect increases in the gross noble gas concentrations of the effluent. Hence, placement of the detectors relative to HEPA and/or charcoal filters does not significantly affect their response. Since releases of iodines and particulates will be accompanied by much larger releases of noble gases, the changes in ventilation monitor readings provide indication of a change in airborne activity concentration in one or more of the plant's areas. If an increase is detected, its source and magnitude will be determined using portable samplers.

Normally occupied non-radiation areas in the plant do not have potential for significant airborne concentrations of particulates and iodine during plant operation because:

1. The ventilation systems are designed to prevent the spread of airborne radioactivity into normally occupied areas.

2. Highly radioactive piping/components are not located in normally occupied areas.

Certain activities, such as refueling, solid waste handling, or turbine teardown, may increase the possibility of encountering significant airborne activities in some normally occupied areas. Continuous local airborne monitoring will be provided during these activities as needed.

Exposure of personnel to high concentrations of airborne activity in radiation areas will be prevented through in-plant surveys and portable particulate and iodine sampling monitors prior to personnel entrance. Continuous monitoring will be provided as required by area conditions and the nature of the entry. The locations of portable monitors (see 1 through 12 below), capable of detecting 10 DAC-hours of particulates and iodines, which are positioned within the station during normal operations to provide supplemental inplant monitoring of particulates and iodine levels.

1. Reactor Building 201′
2. Reactor Building 145′
3. Reactor Building 132′
4. Reactor Building 102′ (2 provided)
5. Reactor Building 77′
6. Service Rad Waste Building 124′
7. Service Rad Waste Building 102′
8. Service Rad Waste Building 54′ (2 provided)
9. Turbine Building 137′ (2 provided)
10. Turbine Building Hallway 102′
11. Turbine Building Hallway 77′
12. Turbine Building 54′

Fifteen Continuous Air Monitors (CAM) are situated throughout the facility to measure, indicate and record the levels of airborne radioactivity at locations where significant airborne radioactivity is likely. Each CAM has the ability to activate a local alarm when a predetermined level is exceeded. If a CAM is broken or otherwise out of service, a Low Volume air sample can take its place until another CAM can be placed in service.

During outages or special evolutions, these monitors may be augmented or shifted, as needed. The quantity required during normal operations and monitor type are identified in Table 12.5-1. Administrative control will prevent inadvertent entry of personnel into normally unoccupied areas (Zone III and above). The provisions discussed above ensure that personnel will not be inadvertently exposed to significant concentrations of airborne activity.

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TABLE 12.3-1

PLANT AREA RADIATION ZONES FOR NORMAL FULL POWER OPERATION

Designation	Max Design Dose Rate	Description
I	< 0.5 mrem/h	Very low or no radiation sources; no radiological control required
II	< 2.5 mrem/h	Low radiation sources; radiological control required
III	< 15.0 mrem/h	Low to moderate radiation sources; radiological control required
IV	< 100 mrem/h	Moderate radiation sources; radiological control required
V	< 1 rem/h	High radiation sources; radiological control required
VI	< 10 rem/h	Extremely high source; radiological control required
VII	< 100 rem/h	Extremely high source; radiological control required
VIII	> 100 rem/h	Extremely high source; radiological control required

TABLE 12.3-2

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Table 12.3-3

LIST OF POST-ACCIDENT VITAL AND USEFUL AREAS

Vital Areas Requiring Continuous Occupancy

Control Room
Technical Support Center (TSC)
Operations Support Center (OSC)
Guardhouse

Vital Areas Requiring Infrequent Access

FRVS RMS Post-Accident Skid
FRVS Sample Transport Path
Diesel Generator and Accessories

Useful Areas

HP/Access Control Areas
Emergency Assembly Points
OSC
TSC

Remote Shutdown Panel

TABLE 12.3-3a

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(Historical Information)

TABLE 12.3-4

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

GRACE 1	Multigroup, multiregion, gamma ray attenuation code used to compute gamma heating and gamma dose rates in slab geometry (Reference 12.3-6).
GRACE 2	Multigroup, multiregion, gamma ray attenuation code used to compute the dose rate or heat generation rate for a spherical or a cylindrical source with slab or concentric shields (Reference 12.3-7).
ANISN	Multigroup, multiregion code solving the Boltzmann transport equation for neutrons and gamma rays in one dimensional slab, cylindrical, or spherical geometry (Reference 12.3-8).
SDC	Multigroup, multiregion, gamma ray attenuation code that calculates dose rates for 13 geometry options (Reference 12.3-9).
QAD	Multigroup, multiregion, three dimensional, point kernel code that calculates fast neutron and gamma ray dose and heat generation rates (Reference 12.3-10).
NAP	Determines neutron activation and gamma emission source strengths as a function of neutron exposure and decay time (Reference 12.3-11).
MORSE-CG	Three dimensional Monte Carlo neutron and gamma ray general transport code (Reference 12.3-12).
DOT 3	Two dimensional neutron, gamma ray, discrete ordinate, transport code (Reference 12.3-13).

(Historical Information)

TABLE 12.3-4 (Cont)

ORIGEN	Isotopic generation and depletion code that solves equations of radioactive growth and decay for isotopes of arbitrary coupling (Reference 12.3-14).
G-33	A general purpose gamma ray scattering code (Reference 12.3-15).

(**Historical Information**)

TABLE 12.3-5

REACTOR BUILDING VENTILATION SYSTEM DESIGN FEATURES

Location	Radiological Safety Features	Supply/Exhaust Air Flow Rate, cfm
All areas	RBVS: Three 50 percent supply fans, and three 50 percent exhaust fans. Building isolation upon LOCA/high activity signal. Automatic/manual switch to FRVS. Supply air passes through high efficiency filters and exhaust air passes through HEPA filters.	Total with 2 fans operating: 90,500/ 98,500 Recirculation rate during normal operation is zero. Refueling mode: 107,000/110,400
Reactor Building	FRVS: (emergency condition) 120,000 cfm filtered by charcoal and HEPA filters.	Total of 4 recirculating fans operating: 120,000 One ventilation fan operating: 250 to 9000
Reactor Building	CPCS: (drywell and torus cleanup) 3000 cfm filtered by charcoal and HEPA filters.	Total of 1 fan operating: 3000 recirculation

(Historical Information)

TABLE 12.3-6

TURBINE BUILDING VENTILATION SYSTEM DESIGN FEATURES

Area	Radiological Safety Features	Supply/Exhaust Air Flow Rate, cfm
General personnel access areas	Three 50 percent supply fans, and three 50 percent exhaust fans. All supply air passes through high efficiency filters	Total with 2 fans operating: 180,000/ 127,500(1) Recirculation rate: 0 in summer 122,500 in winter
High radiation equipment areas	Two 100 percent exhaust fans	0(2)/40,200
Lube oil storage areas	Two 100 percent exhaust fans	0(2)/17,800

(1) Total normal exhaust from Turbine Building, which includes 5400 cfm of infiltration air, is 185,400 cfm.

(2) All supply by infiltration.

TABLE 12.3-7

AUXILIARY BUILDING VENTILATION SYSTEM DESIGN FEATURES

Area	Radiological Safety Features	Supply/Exhaust Air Flow Rate, cfm
General personnel access areas (radwaste area)	Two 50 percent supply and three 33-1/3 percent exhaust fans. All supply passes through high efficiency filters. All exhausts from all areas of the radwaste area are passed through HEPA filters	118,500/103,300 Recirculation rate: 0
Unrestricted shop (radwaste area)	One 100 percent exhaust fan	0/20,000 cfm
Radwaste solidification	Two 50 percent supply and two 50 percent exhaust fans. All supply air passes through high efficiency filters. All exhaust from this area is passed through HEPA filters	32,000/34,000

Area	Radiological Safety Features	Supply/Exhaust Air Flow Rate, cfm
Technical support center[(1)]	One 100 percent supply fan delivers 14,000 cfm through low/high efficiency filters. Under the normal mode, 13,000 cfm are recirculated together with 1000 cfm outside air through the low/high efficiency filters. Under the recirculation mode, 10,000 cfm are circulated through the low/high efficiency filters. 4000 cfm are bypassed through the HEPA/charcoal filters. No outside air is taken in. Under the pressurization mode, 10,000 cfm are circulated through the high/low efficiency filters, and 3000 cfm are bypassed through the HEPA/charcoal filters, together with 1000 cfm of outside air	14,000/1000 (normal) 14,000/0 (recirculation) 14,000/1000 (pressurization)

Area	Radiological Safety Features	Supply/Exhaust Air Flow Rate, cfm
Service area	Two 50 percent service area supply (SAS) and two 50 percent service area exhaust (SAE) fans. Supply passes through high efficiency filters, and exhaust passes through HEPA filters	57,800/41,300 11,200 recirculation
Remote shutdown panel room	One 100 percent supply fan	1400/200 exfiltration
Radiochemistry lab	Two 100 percent capacity exhaust fans. All supply air passes through high efficiency filters. All exhaust is passed through HEPA filters	6300/6300 Recirculation rate: 0

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(1) The TSC is located in the Reactor Building, but the ventilation equipment is located on the auxiliary building roof.

(2) Upon high radiation detection, the technical support center emergency filter unit (TSCEF) is activated.
The TSCEF, which can handle 4000 cfm, is designed to mix 3000 cfm of return air with 1000 cfm of outside air, or it can be switched to handle 4000 cfm in a recirculation mode.

(Historical Information)

TABLE 12.3-8

CONTROL ROOM VENTILATION SYSTEM DESIGN FEATURES

Operation Mode	Radiological Safety Features	Air Flow Rate, cfm	Exposure to Airborne Concentrations
Normal	Two 100 percent supply fans, two 100 percent return air fans, and one exhaust fan (1)	18,500 supply, 15,500 recirculation, 3000 outside air in normal mode, and 2500 exhaust	Background
Accident	Two 100 percent supply and two 100 percent return fans(2). Automatic/ manual switch to emergency intake and filtering system on high activity signal	14,500 recirculation air in emergency mode	Less than CFR allowable limits. Whole body <5 rem Thyroid <30 rem

(Historical Information)

TABLE 12.3-8 (Cont)

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(1) The exhaust fan does not operate during emergency.

(2) Upon high radiation detection, the Control Room Emergency Filter (CREF) System is activated. The CREF system, which handles 4000 cfm, has the capability of mixing 3000 cfm of return air with 1000 cfm of outside air, or this system can switch to 4000 cfm return air in a recirculation mode.

The supply air passes through high efficiency filters.
The CREF system air passes through charcoal and HEPA filters.

TABLE 12.3-9

AREA RADIATION MONITOR DETECTOR (1) (2) LOCATIONS

Instrument Number	Description	Location	Elevation	Area	Reference Drawing
Reactor Building					

Security Related Information
Table withheld Under 10 CFR 2.390

TABLE 12.3-9

AREA RADIATION MONITOR DETECTOR (1) (2) LOCATIONS

Instrument Number	Description	Location	Elevation	Area	Reference Drawing

Security Related Information
Table withheld Under 10 CFR 2.390

TABLE 12.3-9

AREA RADIATION MONITOR DETECTOR (1) (2) LOCATIONS

Instrument Number	Description	Location	Elevation	Area	Reference Drawing

Security Related Information
Table withheld Under 10 CFR 2.390

TABLE 12.3-9

AREA RADIATION MONITOR DETECTOR (1) (2) LOCATIONS

Instrument Number	Description	Location	Elevation	Area	Reference Drawing

Security Related Information
Table withheld Under 10 CFR 2.390

TABLE 12.3-10

ATMOSPHERIC DISPERSION FACTORS FOR EVALUATING POST-ACCIDENT ACCESS

Technical Support Center (TSC)

Time Period (hours)	Atmospheric Dispersion Factor (3 sec/m^3)	
	FRVS Vent Release	Turbine Building Release
0-2	2.07E-03	5.79E-03
2-8	1.54E-03	3.73E-03
8-24	5.78E-04	1.35E-03
24-96	4.35E-04	9.28E-04
96-720	3.55E-04	6.96E-04

Operations Support Center (OSC)

Time Period (hours)	Atmospheric Dispersion Factor (3 sec/m^3)	
	FRVS Vent Release	Turbine Building Release
0-2	8.29E-04	9.41E-04
2-8	6.34E-04	6.85E-04
8-24	2.60E-04	2.57E-04
24-96	1.84E-04	1.89E-04
96-720	1.41E-04	1.39E-04

Guardhouse

Time Period (hours)	Atmospheric Dispersion Factor (sec/m^3)	
	FRVS Vent Release	Turbine Building Release
0-2	9.28E-05	1.76E-04
2-8	6.70E-05	1.38E-04
8-24	2.64E-05	5.48E-05
24-96	1.89E-05	3.82E-05
96-720	1.51E-05	3.05E-05

Diesel Generator Area

Time Period (hours)	Atmospheric Dispersion Factor (sec/m^3)	
	FRVS Vent Release	Turbine Building Release
0-2	1.73E-03	7.73E-04
2-8	1.15E-03	5.06E-04
8-24	4.31E-04	1.84E-04
24-96	3.15E-04	1.26E-04
96-720	2.33E-04	9.65E-05

Service Area (FRVS-RMS skid, control access point, remote shutdown panel)

Time Period (hours)	Atmospheric Dispersion Factor (3 sec/m^3)		
	FRVS Vent Release	South Plant Vent Release	Turbine Building Release
0-2	1.29E-03	1.35E-02	3.53E-03
2-8	9.37E-04	1.08E-02	2.56E-03
8-24	3.41E-04	4.45E-03	9.32E-04
24-96	2.66E-04	3.03E-03	5.40E-04
96-720	2.07E-04	2.29E-03	3.82E-04

Figure F12.3-1 intentionally deleted.

Refer to Plant Drawing N-1031 in DCRMS

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Refer to Plant Drawing N-1032 in DCRMS

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Figure F12.3-4 intentionally deleted.

Refer to Plant Drawing N-1033 in DCRMS

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Refer to Plant Drawing N-1035 in DCRMS

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HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

May 3, 2002 F12.3-31

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May 3, 2002	F12.3-33

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HOPE CREEK UFSAR - REV 12	SHEET 1 OF 1
May 3, 2002	F12.3-34

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HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

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HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1 May 3, 2002 F12.3-36

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HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

May 3, 2002 F12.3-38

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HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1
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HOPE CREEK UFSAR - REV 12	SHEET 1 OF 1
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HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

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HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1
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May 3, 2002	F12.3-47

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12	SHEET 1 OF 1
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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

May 3, 2002 F12.3-52

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PSEG NUCLEAR L.L.C.
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

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HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

May 3, 2002 F12.3-54

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HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12 SHEET 1 OF 1

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HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12	SHEET 1 OF 1
May 3, 2002	F12.3-57

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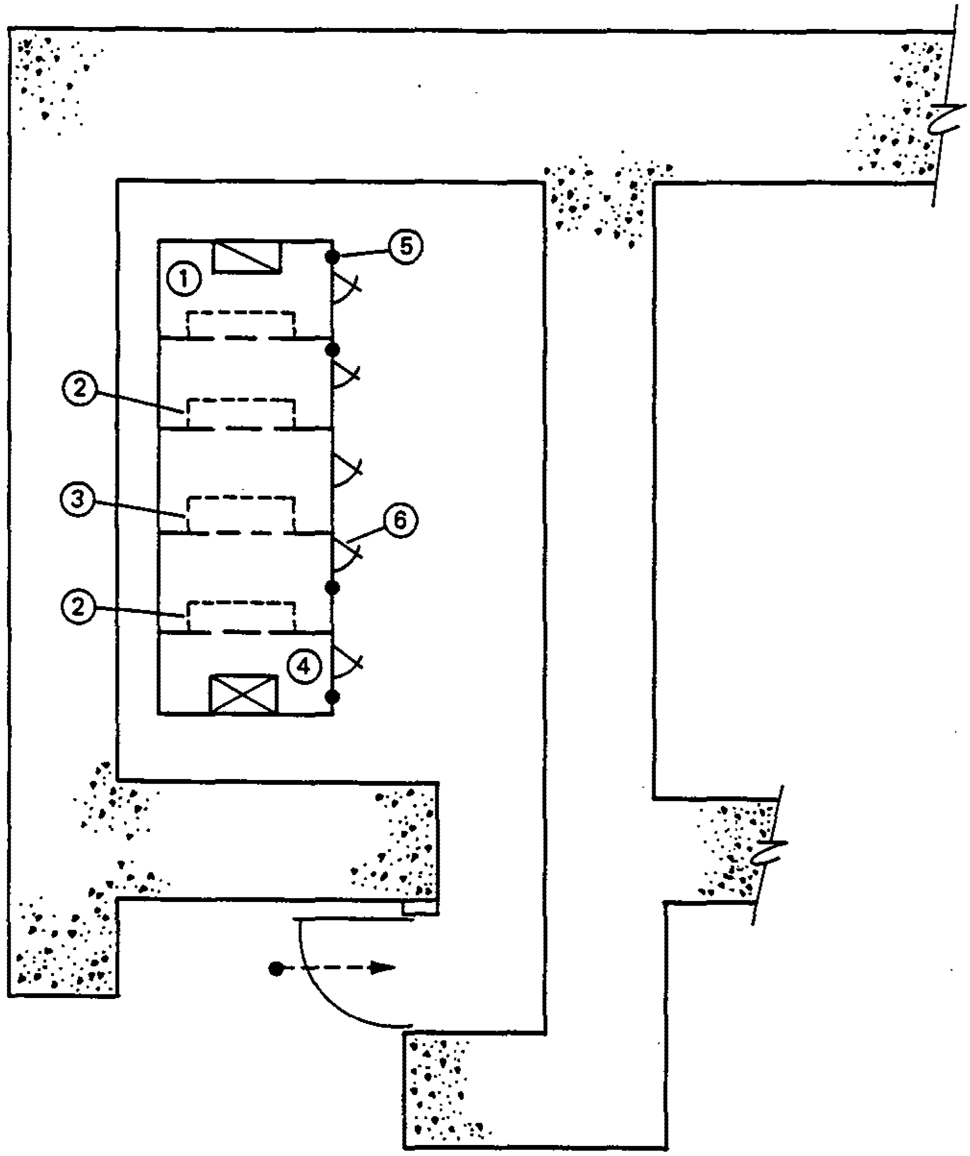
HOPE CREEK GENERATING STATION

HOPE CREEK UFSAR - REV 12	SHEET 1 OF 1
May 3, 2002	F12.3-57A



1 INLET WITH HEATING COIL AND PREHEATER
2 HEPA FILTERBANKS
3 RECHARGEABLE CHARCOAL FILTERS
4 PLENUM WITH OUTLET
5 TESTING CONNECTIONS
6 ACCESS DOORS

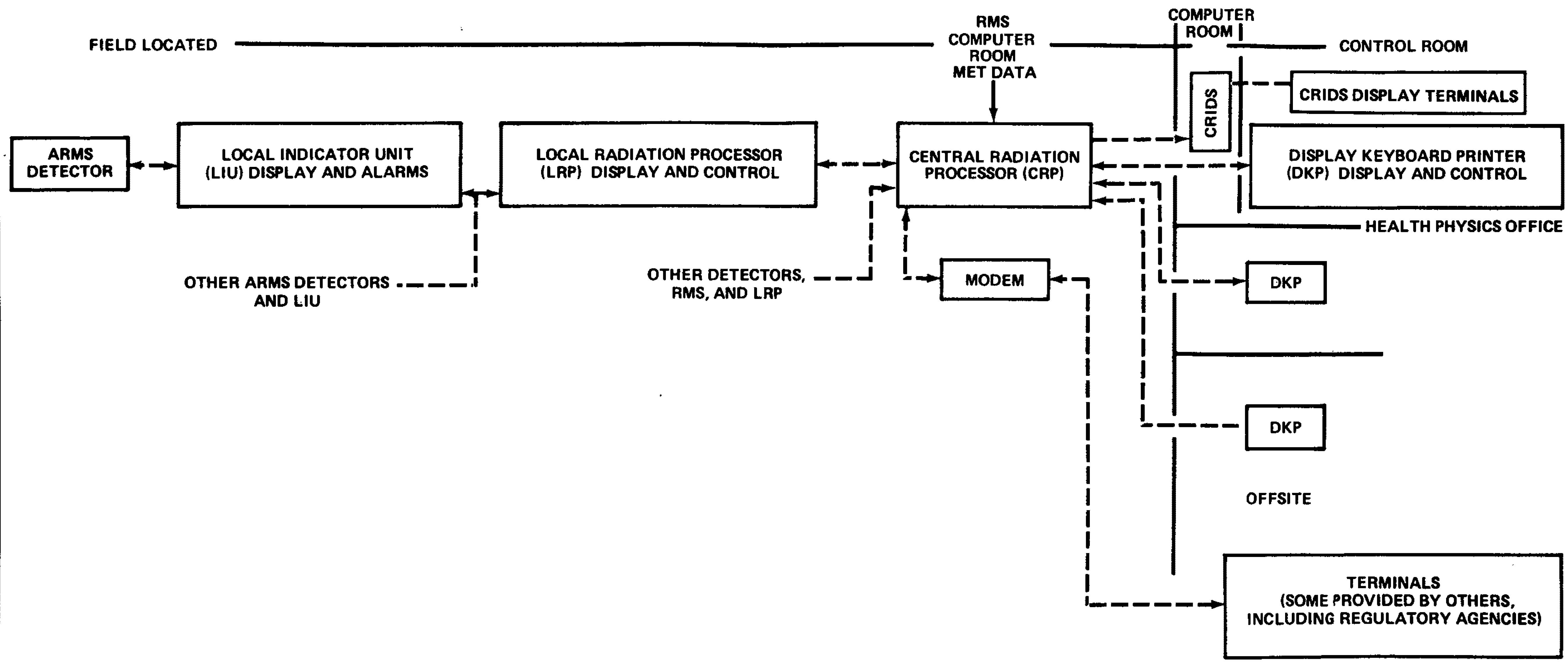
REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

TYPICAL SHIELDING ARRANGEMENT FOR CHARCOAL AND PARTICULATE FILTER

UPDATED FSAR

FIGURE 12.3-58



REVISION 0
APRIL 11, 1988

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION

FUNCTIONAL BLOCK DIAGRAM AREA RADIATION MONITORING SYSTEM

UPDATED FSAR FIGURE 12.3-59

SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2.390

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Up ated FSAR heet 1 of 1
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SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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heet 1 of 1
Fig 12 3 2

SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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heet 1 of 1
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Figure F12.3-64 SH 1 and 2 intentionally deleted. Refer to Plant Drawing P-0047-1 SH 1 & 2 in DCRMS

SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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Figure F12.3-66 SH 1 and 2

intentionally deleted. Refer to Plant

Drawings P-0044-1 SH 1 and P-0046-1

SH 1 in DCRMS

SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2.390

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heet 1 of 1
Fig 12 3 8

SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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heet 1 of 1
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SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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heet 1 of 1
Fig 12 3 Ø

SECURITY RELATED
INFORMATION WITHHELD
UNDER 10 CFR 2 390

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heet 1 of 1
F g 2 3 71

12.4 DOSE ASSESSMENT

This section discusses the estimated radiation exposures to plant personnel both inside the plant structures and at locations outside of the plant structures.

Not all the methods for occupational radiation dose assessment discussed in Regulatory Guide 8.19 are employed for Hope Creek Generating Station. The assessment of doses incurred during routine operation and maintenance were performed on an area by area basis rather than a function by function basis. This approach was taken because when combined with operating experience, it ensures a more complete estimate of the person-hours spent in radiation areas. Doses attributable to the remaining activities discussed in Regulatory Guide 8.19 were assessed on the basis of historical exposure data. This approach was taken because of large uncertainties involved in estimating person-hours for them. The details of these approaches are described below.

12.4.1 Direct Radiation Dose Estimates for Exposures Within the Plant Structures

To estimate the total annual person-rem dose from direct radiation shine to personnel within the plant structures, seven broad categories or job functions were defined, and the annual person-rem dose for each category was evaluated. Where the functions and expected radiation levels are predictable or clearly defined, analytical methods were employed for the person-rem estimates. In other cases, the estimate basis is historical exposure data from operating boiling water reactor (BWR) power plants. Section 12.4.1.1 provides the definitions and components of each of the seven broad categories, and Section 12.4.1.2 briefly describes the estimation techniques used.

Resultant dose estimates are contained in Section 12.4.1.3, along with further discussion of the factors involved and the methodology used for each category and its related components.

(Historical Information)

12.4.1.1 Definition of Categories Used in Exposure Estimates

The categories used in estimating the total annual plant person-rem dose are:

1. Routine operations - This category is composed of the following four subcategories:

   a. Routine patrols for surveillance of the Reactor and Turbine Buildings, and the control, standby diesel generator (SDG), service, and radwaste areas of the Auxiliary Building

   b. Periodic tests and checks in the Reactor Building, Turbine Building, control and SDG areas, and the service and radwaste areas

   c. Control room operations - This category includes areas inside structures where plant personnel occupancy is assumed to be continuous during a workshift, e.g., the main control room and the radwaste control room.

   d. Administration and Corporate Engineering - This category includes all activities in the administration and warehouse buildings. Although these activities take place in nonradiation areas, they are included in the dose assessment because of the proximity of the buildings to the Turbine Building.

2. Routine maintenance - This includes all scheduled maintenance and the preventive maintenance performed in the radiation areas of the Reactor Building; Turbine Building; and radwaste, control, and SDG areas of the Auxiliary Building.

3. Inservice inspections and nondestructive examination (NDE) - These are inspections normally performed by plant personnel and outside contractors. Such inspections normally occur during outages on piping and components that cannot be checked while the plant is at power.

4. Special maintenance - All maintenance that is not scheduled. This type of maintenance is not planned in advance and normally is difficult to predict.

5. Radwaste processing - This includes any work with solid or liquid radwaste, e.g., movement of casks and liners; radwaste, condensate system, or fuel pool filter changes; resin moving; collection, transport, and baling of low level trash, etc. Maintenance of radwaste equipment is covered by the maintenance categories and is not included in this job function.

6. Refueling - This includes all activities with new fuel, spent fuel, or reactor components that are performed in the reactor and fuel pool areas during refueling and stocking operations.

7. Health physics - This covers all radiation protection activities and includes chemistry operations and sample collection.

12.4.1.2 Exposure Estimate Methodology

12.4.1.2.1 Analytical Method

The analytical method used for person-rem estimation is based upon the product of estimated exposure time and estimated ambient dose rate. Estimates of the occupancy time requirements for operations associated with equipment in plant radiation areas, e.g., maintenance time or surveillance time, are developed first. An applicable frequency of occurrence is then factored in to determine the annual exposure time for each operation.

(Historical Information)

Areas with no significant radiation sources are not included in the exposure estimate. Where radiation sources are present, the design maximum dose rate of 2.5 mrem/h is assumed for the Zone II areas. Radiation zones are defined in Table 12.3-1. Similarly, a dose rate of 15 mrem/h is assumed for the Zone III areas. All other estimated dose rates are based on either calculations or actual radiation levels encountered at operating plants, and are representative of conditions at plants with more than 5 years operation. The analytical method is used in determining the exposure estimates for the routine maintenance and routine operations categories.

12.4.1.2.2 Historical Method

In the historical method, the annual person-rem dose is estimated from the exposures received at operating BWR power plants. This method is used for all the remaining categories, i.e., special maintenance, inservice inspection, radwaste processing, refueling, and health physics. The data sources are the annual and semiannual BWR operating reports and plant correspondence with regulatory agencies. Included are a total of 137 reactor years of data for 22 nuclear stations listed in Table 12.4-1. The average licensed power level of these units is 780 MWe, with the smallest rated at 514 MWe. The data are collected and assembled using the following guidelines:

1. No data before the first calendar year following completion of at least 9 months of commercial operation are used.

2. In multiple unit plants, each unit is assumed to contribute equally to the annual exposures.

3. If exposure contributions from two or more job functions cannot be separated, a conservative approach is taken by assigning all the exposure to both functions.

(Historical Information)

Table 12.4-2 contains the results of the historical data compilation and includes both the number of reactor years and standard deviations associated with each job function. It is expected that these data will be representative of future operations at HCGS. The large standard deviations, which range from about 60 percent to 200 percent of the mean values, are indicative of the wide spread of data that have been reported within each exposure category.

12.4.1.3 Results of Annual Direct Radiation Dose Estimates

The annual person-rem estimates for each category and subcategory are detailed below. The methods used in their determination are as described previously, with any additional assumptions or required information also included below.

A summary of the direct radiation dose estimates is given in Section 12.4.1.3.8.

12.4.1.3.1 Routine Operations Dose Estimate

During normal operations, routine patrols and surveillances are performed by plant operators. The majority of items checked are rotating equipment, e.g., pumps, fans, etc, and each is viewed to verify the absence of leaks, excessive vibrations, or other abnormal conditions. For the surveillance person-rem exposure estimation, the following assumptions are made:

1. Dose rates for individual rooms are estimated as outlined in Section 12.4.1.2. Additionally, because of the high potential dose rates associated with certain equipment, routine surveillances of such equipment are performed from a remote location such as the equipment cell doorway or through a radiation shielding viewing window, and credit is taken for the lower ambient radiation level at that particular location.

2. Each patrol consists of only one person.

(Historical Information)

3. Walking speed during the patrols is 200 ft/min.

4. Frequency of patrols for cubicles with active components is once per shift for Zone II and Zone III areas, and once per day for Zone IV and Zone V areas. Frequency of patrols for areas with passive components is once per week.

5. Stay time is 5 minutes for areas with active components and 2 minutes for areas with passive components.

6. An average dose rate to account for both exposures during room entries and during walking was developed for each building.

The results of the routine operations and maintenance exposure estimate are contained in Tables 12.4-3, 12.4-5, and 12.4-9.

Similarly, the details and results of the exposure estimate for the periodic testing subcategory are also contained in Tables 12.4-3 and 12.4-6 through 12.4-8. Since periodic testing is assumed to occur during equipment shutdown, the estimated shutdown dose rates are used for determining the associated exposures.

Control room operations exposures are estimated from the anticipated control room radiation levels and the staffing requirements for the main and radwaste control rooms. Table 12.4-14 contains the details of the control room operations exposure estimate.

The remaining subcategory is administration and corporate engineering exposures. The design basis dose rate for the Administration Building is 0.125 mrem/h. Assuming 2000 hours of occupancy per year, the most exposed individual would receive no more than 0.25 rem per year. Conservatively assuming that the entire administration staff of 200 individuals receives this dose will result in a total annual exposure estimate of 50 person-rem.

(Historical Information)

Corporate engineering staff is located in the existing warehouse, which receives an exposure of 23 mrem/yr shown on Figure 12.4-1. An estimated 10 person-rem/yr is conservatively added for engineering personnel in the warehouse.

The total annual exposure estimate for the routine operations category is the sum of the four subcategory annual exposures as shown in Table 12.4-3 and is 95.7 person-rem.

12.4.1.3.2 Routine Maintenance Dose Estimate

A detailed review of plant radiation areas was performed to produce a listing of the types and quantities of selected equipment present in each area. Total annual maintenance person-hours are estimated for each equipment type identified based on a combination of operating experience and engineering judgment. Total estimated person-hours are shown in Table 12.4-4 and are intended to include all expected routine activities for each equipment type, such as valve repacking, valve relapping, pump seal replacement, fan overhaul, etc.

In any area, the total annual person-hours for routine maintenance are thus the summation of the quantity person-hour products for all equipment types found in the area. Multiplying the area's annual maintenance person-hours by the anticipated maximum area dose rate produces the estimated person-rem by area. As with periodic testing, the estimated shutdown dose rate is used for estimating maintenance exposures.

A "total annual" maintenance approach was used for each component, since currently available data generally does not contain sufficient information to provide a basis for person-hour breakdowns by individual maintenance activity. In addition, the area by area methodology employed makes estimate compilations by system unnecessary, because locations where high person-rem exposures are expected are clearly indicated. Tables 12.4-6 through 12.4-8 contain the details of the routine maintenance exposure estimate for each building. The sum from the three buildings is 288 person-rem.

(Historical Information)

12.4.1.3.3 Inservice Inspection Dose Estimate

The annual exposure estimate for inservice inspection is based upon the data from operating BWRs given in Table 12.4-2, and is 39 person-rem.

12.4.1.3.4 Special Maintenance Dose Estimate

The annual exposure estimate for special maintenance is based upon the data from operating BWRs given in Table 12.4-2, and is 351 person-rem.

12.4.1.3.5 Radwaste Processing Dose Estimate

Many of the operations in the plant associated with radwaste processing are performed remotely, and these are therefore not suitable for evaluation by the analytical estimation technique. Consequently, the annual person-rem estimate for these remotely operated radwaste processing activities is more properly taken from the historical BWR operating data of Table 12.4-2, which is a conservative estimate of the anticipated exposure, and is 53 person-rem.

12.4.1.3.6 Refueling Dose Estimate

The annual exposure estimate for refueling is based upon the data from operating BWRs given in Table 12.4-2, and is 34 person-rem.

12.4.1.3.7 Health Physics Dose Estimate

The annual exposure estimate for radiation protection monitoring, including chemistry operations and sampling, is based upon the data from operating BWRs given in Table 12.4-2, and is 53 person-rem.

(Historical Information)

12.4.1.3.8 Summary of Direct Radiation Dose Estimates

The annual dose estimates in the preceding seven sections are summarized and totaled in Table 12.4-5. As shown in this table, the estimate of total annual inplant exposure from direct radiation is 914 person-rem.

Dose estimates for inservice inspection, special maintenance, radwaste processing, refueling, and health physics activities are based on historical data from operating facilities. Any further breakdown of the dose estimate such as was made for routine operations and routine maintenance, for these types of activities still rely primarily on the historical information available. The resultant dose estimate is therefore not any more precise than the estimate based solely on reported radiation exposures.

In all five areas where historical data is used in the dose estimate, the HCGS design includes design features that reduce actual exposures received by plant personnel. However, due to the lack of sufficiently detailed information to allow the precise quantification of the dose reduction, a calculation of the reduction has not been attempted.

As examples of design features that result in dose reductions, the following design features are incorporated to facilitate inservice inspection, inspection, plant personnel surveillance, maintenance, test, and repair operations.

1. Quick removal insulation around the reactor pressure vessel (RPV) nozzles.

2. Access panels in the RPV skirt to allow access to the RPV bottom head welds.

3. The use of remote, track-mounted instrument vehicles for RPV weld inspection.

(Historical Information)

4. The use of remote, automatic weld inspection of the RPV nozzle welds.

5. Platforms at strategic locations to improve access and overall work conditions.

6. The provision of quick opening shield doors at RPV nozzle and pipe penetrations through the biological shield.

7. Use of radiation shielding windows during routine surveillances of plant areas.

12.4.2 Airborne Radioactivity Dose Estimates for Exposures Within the Plant Structures

The estimated exposures to plant personnel from airborne radioactivity are based upon the source distributions and radionuclide concentrations presented in Section 12.2 and Tables 12.2-138 through 12.2-144. Because of the limited geometry afforded by the finite compartment sizes within the plant, personnel exposures due to noble gas immersion are expected to be insignificant when compared to inhalation exposures and therefore are not estimated.

To determine whether exposure contributions from airborne radioactive particulates are significant, an evaluation for each area is made of the ratio of total particulate Derived Air Concentration (DAC) fractions to total radioiodine DAC fractions, which is equivalent to the ratio of particulate DAC-hours to iodine DAC-hours. The evaluations performed in this section are based on DACs listed in Appendix B of 10CFR20 as of 1988. For the turbine and reactor building enclosure areas, the particulate to iodine ratios are approximately 1:80 and 1:10, respectively, indicating that the particulate inhalation exposures are not significant in those areas. In the radwaste areas of the Auxiliary Building, however, the particulate to iodine ratio is approximately 1:4. Since over 50 percent of the total particulate DAC fraction is attributable to Co-60, both the thyroid inhalation dose due to radioiodines and the lung inhalation dose due to Co-60 are estimated for the radwaste areas, as the thyroid and the lung are the critical organs for iodines and Co-60, respectively.

(Historical Information)

Per NUREG-0016, Revision 1 the BWR GALE code, the plant's total annual tritium release is equally divided between the Turbine Building and the Reactor Building. The turbine building tritium release is assumed to occur continuously. Estimates of whole body doses due to exposure to airborne tritium in Turbine Building areas are given in Tables 12.4-10 and 12.4-13. The Reactor Building tritium release is assumed to occur at the refueling deck during refueling operations. Based on the turbine building tritium dose estimates and data in Tables 12.2-141 and 12.2-142, it is concluded that whole-body doses due to exposure to airborne tritium on the refueling deck will be negligible compared with the refueling dose estimate (Section 12.1.3.6). Hence, tritium exposures in the Reactor Building are not estimated.

Tables 12.4-10 through 12.4-13 contain the compilations of the estimated annual occupancy times and the estimated annual exposures for each of the areas identified in Section 12.2 as potential sources of airborne radioactivity. The occupancy times are based upon detailed reviews of each area and the determination of the operations that might occur in those areas. The exposures are based upon the estimated airborne concentrations in Tables 12.2-138 through 12.2-144, dose factors from Table C-1 of Regulatory Guide 1.109, and an assumed breathing rate of 3.47×10^{-4} m^3/s.

12.4.3 Exposures at Locations Outside Plant Structures

The radiation exposures at locations outside the plant structures are estimated for two areas: the site boundary (property line) and the guardhouse.

(Historical Information)

12.4.3.1 Direct Radiation Dose Estimates Outside Plant Enclosures

At locations outside plant enclosures, the direct radiation exposure is due to the following principal components:

1. The activity stored outside the plant structures in the condensate storage tank (CST).

2. Turbine shine due to the N-16 present in the reactor steam.

3. Radiation shine during transport of drummed radwaste and spent fuel assemblies to offsite facilities.

Based on the calculated surface dose rates for the CST, the radwaste transport casks, and the spent fuel shipping cask, the dose contribution at locations outside the plant structures due to these sources is considered negligible, compared to the N-16 shine from the turbine.

The N-16 present in the reactor steam in the primary steam lines, turbines, and moisture separators provides a dose contribution to locations outside the plant enclosure as a result of the high energy gamma rays that it emits as it decays. To reduce the turbine shine doses, radiation shielding is provided around the turbine train.

The resultant annual exposure due to turbine shine was calculated with the SKYSHINE computer program discussed in Reference 12.4-1. Point sources are used to represent the components on the turbine deck, and the source strengths are given in Table 12.4-15.

The calculated dose rate at the west side of the boundary, as shown on Figure 12.4-1, and listed in Table 12.4-16, is 0.0045 mrem/h. However, the west and south portions of the site boundary are adjacent to the Delaware River, where personnel occupancy will be very low.

(Historical Information)

The maximum dose rate for areas with potentially high occupancy occurs at the north side of the boundary and is 1.2 mrem/yr.

The dose rate in the guardhouse is calculated by the SKYSHINE program to be 0.051 mrem/h. Assuming a visitor stays at the guardhouse 1 day a year for 8 hours, the estimated dose for the visitor is 0.41 mrem/yr, as shown in Table 12.4-17.

12.4.3.2 <u>Airborne Radioactivity Dose Estimates Outside Plant Structures</u>

Doses at the site boundary due to released airborne radioactivity are given in Section 11.3.

12.4.4 References

12.4-1 M.G. Wells, et al, "SKYSHINE, A Computer Procedure For Evaluation Effect of the Structure Design on N-16 Gamma Ray Dose Rates," RRA-T7209, Radiation Research Associates Inc, Fort Worth, Texas.

TABLE 12.4-1

SUMMARY OF HISTORICAL DATA USED IN COMPILING EXPOSURES RECEIVED AT OPERATING BOILING WATER REACTORS

				Reactor Years Contributing to Data Base							
Plant	Unit	Power (MWe) (1)	Data of Commercial Operation	Routine Operations	Routine Maintenance	Inservice Inspection	Special Maintenance	Waste Processing	Refueling Operations	Health Physics	Annual Total (2)
Browns Ferry	1	1065	8/74	4	5	4	5	5	5	5	5
Browns Ferry	2	1065	8/75	4	5	4	5	5	5	5	5
Browns Ferry	3	1065	3/77	3	4	4	4	4	4	4	4
Brunswick	2	790	11/75	5	5	5	5	5	3	5	5
Brunswick	1	790	3/77	4	4	4	4	4	2	4	4
Cooper	-	764	7/74	6	6	6	6	6	6	6	6
Dresden	2	772	7/70	9	9	3	8	9	9	8	9
Dresden	3	773	11/71	9	9	3	8	9	9	8	9
Duane Arnold	-	515	2/75	6	6	6	6	6	6	6	6
Fitzpatrick	-	810	7/75	5	5	5	5	5	1	3	5
Edwin I. Hatch	1	757	12/75	5	5	5	5	4	3	5	5
Edwin I. Hatch	2	767	9/79	1	1	1	1	1	1	1	1
Millstone	1	654	3/71	7	7	6	7	6	6	7	7
Monticello	-	536	6/71	8	8	5	8	8	7	7	8
Nine Mile Point	1	610	12/69	7	7	6	7	6	7	7	7
Oyster Creek	1	620	12/69	9	9	8	9	8	6	8	9
Peach Bottom	2	1051	7/74	6	6	6	6	6	6	6	6
Peach Bottom	3	1035	12/74	6	6	6	6	6	6	6	6
Pilgrim	1	670	12/72	7	7	7	7	7	4	7	7
Quad Cities	1	769	2/73	8	8	3	7	8	8	8	8
Quad Cities	2	769	3/73	8	8	3	7	8	8	8	8
Vermont Yankee	-	504	11/72	7	7	7	7	7	6	7	7
Totals				134	137	107	133	133	118	133	137

(1) From NUREG-0713

(2) Data from years for which a breakdown by job function is not available not included

(Historical Information)

TABLE 12.4-2

OCCUPATIONAL EXPOSURES BY JOB FUNCTION FOR OPERATING BOILING WATER REACTORS

	Number of Reactor Years Averaged	Average Person-Rem per Reactor Year	Standard Deviation (Person-rem per reactor year)
Job functions			
Routine operations	134	70.9	53.4
Routine maintenance	137	309	330
Inservice inspection	107	39.2	45.4
Special maintenance	133	351	374
Waste processing	133	52.6	112
Refueling	118	34.3	51.8
Health physics	133	52.6	47.5
Total reported(1) annual exposure	137	881	569

(1) Total exposure by job function differs from the annual total exposure due to conservatisms employed in compilation of job function exposures.

(Historical Information)

TABLE 12.4-3

SUMMARY OF ROUTINE OPERATIONS EXPOSURE ESTIMATE

Operation	Annual Estimated Person-Rem
Routine surveillances	
Turbine Building	3.2
Reactor Building	7.9
Radwaste areas	1.6
	12.7
Periodic tests	
Turbine Building	3.1
Reactor Building	4.5
Radwaste areas	4.4
	12.0
Administration and engineering	
Administration	50
Corporate engineering	10
Control room operations	
Main control room	8.8
Radwaste control room	2.2
	11.0
Total	95.7

(Historical Information)

TABLE 12.4-4

ESTIMATE OF EXPECTED ROUTINE MAINTENANCE REQUIREMENTS

Equipment Type Or Activity	Estimated Person-hours Per Year
Valves	
Under 3 inches	3
3 to 6 inches	9
8 to 10 inches	12
12 to 16 inches	17
Over 16 inches	23
Valve operators	
Reach rod	1
Air operator	2
Motor operator	2
Pumps	
Reactor recirculation	150
Residual heat removal (RHR) or condensate	100
Any other pump	50
Motors	
200 hp or greater	20
Less than 200 hp	13
Motor-generator	35
Main generator	1000
Turbines	
Main turbine	2400
Any other turbine	240
Heat exchanger	30
Evaporator	50
Chiller	5
Unit heater	5
Unit cooler	5
Compressor	25
Fan or blower	25

(Historical Information)

TABLE 12.4-4 (Cont)

Equipment Type Or Activity	Estimated Person-hours Per Year
Main condenser (per shell)	150
Hoist or crane	60
Instrument panel	150
Instrument rack	150
Motor control center	5
Switchgear	120
Centrifuge	37
Agitator	5
Transfer cart or conveyor	15
Traversing in-core probe (TIP) drive	20
Control rod drive (CRD) hydraulic units (total per reactor)	600
HVAC filter unit (prefilter, HEPA filters, and charcoal filter)	20
Off-gas system (holdup pipe to plant vent - total for station)	780

(Historical Information)

TABLE 12.4-5

SUMMARY OF INPLANT DIRECT RADIATION EXPOSURE ESTIMATES

Category	Annual Estimated Person-Rem
Routine operations	96
Routine maintenance	288
Inservice inspection	39
Special maintenance	351
Radwaste processing	53
Refueling	34
Health physics	53
Total	914

(Historical Information)

TABLE 12.4-6

EXPOSURE ESTIMATES FOR TURBINE BUILDING AREAS

Elevation	Room#(3)	Estimated Dose Rate mrem/h)(1)	Periodic Testing: Estimated Annual Person -Hours	Periodic Testing: Estimated Annual Person -Rem	Routine Maintenance(2): Estimated Annual Person -Hours	Routine Maintenance(2): Estimated Annual Person -Rem
54'-0"	1102	2	42	8.4-02	1,127	2.3
54'-0"	1103	4	0	0	17	6.8-02
54'-0"	1104	2	24	4.8-02	708	1.4
54'-0"	1106	2.5	3	7.5-03	300	7.5-01
54'-0"	1107	5	84	4.2-01	1,410	7.1
54'-0"	1108	15	3	4.5-02	156	2.3
54'-0"	1111	2.5	13.5	3.4-02	860	2.2
54'-0"	1112	4	27	1.1-01	142	5.7-01
54'-0"	1113	-	0	0	0	0
54'-0"	1114	4	0	0	150	6.0-01
54'-0"	1115	4	0	0	150	6.0-01
54'-0"	1116	4	0	0	150	6.0-01
54'-0"	1117	4	48	1.9-01	429	1.7
54'-0"	1128	2	9	1.8-02	508	1.0
77'-0"	1206	15	22.5	3.4-01	143	2.1
77'-0"	1207	6	3	1.8-02	329	2.0
77'-0"	1209	10	1.5	1.5-02	54	5.4-01
77'-0"	1210	10	1.5	1.5-02	45	4.5-01
77'-0"	1211	10	1.5	1.5-02	54	5.4-01
77'-0"	1212	10	1.5	1.5-02	54	5.4-01
77'-0"	1213	10	3	3.0-02	54	5.4-01
77'-0"	1214	10	1.5	1.5-02	54	5.4-01
77'-0"	1215	10	1.5	1.5-02	54	5.4-01
77'-0"	1216	60	9	5.4-01	240	1.4+01
77'-0"	1218	6	13.5	8.1-02	345	2.1
77'-0"	1219	6	3	1.8-02	204	1.2
77'-0"	1220	4	6	2.4-02	146	5.8-1
77'-0"	1229	-	0	0	0	0
77'-0"	1230	4	0	0	318	1.3
102'-0"	1305	3	22.5	6.8-02	374	1.1
102'-0"	1308	3	30	9.0-02	388	1.2
102'-0"	1309	3	24	7.2-02	376	1.1
102'-0"	1310	-	0	0	0	0
102'-0"	1311	-	0	0	0	0
102'-0"	1312	-	0	0	0	0
102'-0"	1313	4	121.5	4.9-01	2,909	1.2+01
120'-0"	1405	4	12	4.8-02	149	6.0-01
137'-0"	1504	3	9	2.7-02	164	4.4-01
137'-0"	1505	3	10.5	3.2-02	105	3.2-01
137'-0"	1506	3	3	9.0-03	120	3.6-01
137'-0"	1508	3	3	9.0-03	458	1.4

(Historical Information)

TABLE 12.4-6 (Cont)

Elevation	Room# [3]	Estimated Dose Rate [1] (mrem/h)	Periodic Testing: Estimated Annual Person -Hours	Periodic Testing: Estimated Annual Person -Rem	Routine Maintenance [2]: Estimated Annual Person -Hours	Routine Maintenance [2]: Estimated Annual Person -Rem
137'-0"	1509	1.6	9	1.4-02	284	4.5-01
137'-0"	1510	1.6	10.5	1.7-02	339	5.4-01
137'-0"	1511	1.6	13.5	2.2-02	340	5.4-01
137'-0"	1512	4	7.5	3.0-02	107	4.3-01
137'-0"	1513	1.6	36	5.8-02	465	7.4-01
137'-0"	1514	-	0	0	0	0
171'-0"	1704	-	0	0	0	0
Total			634.5	3.1	14,779	69

(1) Assumes equipment in the room is shut down.
(2) The estimated exposures assume that all person-hours are expended in the areas in which they appear. Portions of these person-hours may be spent at lower radiation levels within the area. Components may also be removed to a lower background area for maintenance.
(3) For room locations, see Figures 12.3-16 through 12.3-21.

(Historical Information)

TABLE 12.4-7
EXPOSURE ESTIMATES FOR REACTOR BUILDING AREAS

Elevation	Room#[3]	Estimated Dose Rate[1] (mrem/h)	Periodic Testing Estimated Annual Person -Hours	Periodic Testing Estimated Annual Person -Rem	Routine Maintenance[2] Estimated Annual Person -Hours	Routine Maintenance[2] Estimated Annual Person -Rem
Drywell	4220	10	0	0	604	6.0
Drywell	4221	10	12	1.2-01	623	6.2
Drywell	4341	10	75	7.5-01	2,939	2.9+01
54'-0"	4101	5	10.5	5.3-02	262	1.3
54'-0"	4102	2	36	7.2-02	1,994	4.0
54'-0"	4106	2.5	18	4.5-02	328	8.2-01
54'-0"	4109	15	0	0	409	6.1
54'-0"	4110	5	24	1.2-01	610	3.1
54'-0"	4111	5	33	1.7-01	694	3.4
54'-0"	4113	15	18	2.7-01	366	5.5
54'-0"	4114	2.5	9	2.3-02	0	0
54'-0"	4115	2.5	18	4.5-02	328	8.2-01
77'-0"	4208	15	7.5	1.1-01	195	2.9
77'-0"	4214	15	7.5	1.1-01	172	2.6
102'-0"	4316	30	15	4.5-01	602	1.8+01
102'-0"	4318	15	0	0	52	7.8-01
102'-0"	4319	5	0	0	14	7.0-02
102'-0"	4320	2.5	228	5.7-01	963	2.4
102'-0"	4321	5	7.5	3.8-02	358	1.8
102'-0"	4327	5	0	0	42	2.1-01
102'-0"	4328	2.5	4.5	1.1-03	1,152	2.9
102'-0"	4329	5	3	1.5-02	271	1.4
132'-0"	4402	30	6	1.8-01	68	2.0
132'-0"	4403	28	7.5	2.1-01	132	3.7
132'-0"	4405	28	7.5	2.1-01	132	3.7
132'-0"	4406	2.3	9	2.1-02	176	4.0-01
132'-0"	4407	0	0	0	9	0
145'-0"	4502	2.3	13.5	3.1-02	279	6.4-01
145'-0"	4503	2.3	13.5	3.1-02	279	6.4-01
145'-0"	4505	30	3	9.0-02	77	2.3
145'-0"	4506	28	6	1.7-01	182	5.1
145'-0"	4509	50	0	0	27	1.4
162'-0"	4601	2.3	12	2.8-02	150	3.5-01
162'-0"	4603	15	0	0	44	6.6-01
162'-0"	4620	10	0	0	0	0
162'-0"	4621	10	0	0	0	0
162'-0"	4625	16	9	1.4-01	147	2.4
162'-0"	4624	16	9	1.4-01	120	1.9
162'-0"	4627	16	9	1.4-01	75	1.1

(Historical Information)

TABLE 12.4-7 (Cont)

			Periodic Testing		Routine Maintenance (2)	
Elevation	Room# (3)	Estimated Dose Rate (1) (mrem/h)	Estimated Annual Person -Hours	Estimated Annual Person -Rem	Estimated Annual Person -Hours	Estimated Annual Person -Rem
162'-0"	4628	16	9	1.4-01	72	1.0
162'-0"	4615	2.5	13	3.3-02	0	0
Total			653.5	4.5	14,947	127

(1) Assumes equipment in the room is shut down.
(2) The estimated exposures assume that all person-hours are expended in the areas in which they appear. Portions of these person-hours may be spent at lower radiation levels within the area. Components may also be removed to a lower background area for maintenance.
(3) For room location, see Figures 12.3-9 through 12.3-15.

(Historical Information)

TABLE 12.4-8

EXPOSURE ESTIMATES FOR THE RADWASTE AREAS [3]

Elevation	Room# [3]	Estimated Dose Rate [1] (mrem/h)	Periodic Testing Estimated Annual Person -Hours	Periodic Testing Estimated Annual Person -Rem	Routine Maintenance [2] Estimated Annual Person -Hours	Routine Maintenance [2] Estimated Annual Person -Rem
54'-0"	3101	42	0	0	9	3.8-01
54'-0"	3102	3	9	2.7-02	138	4.1-01
54'-0"	3107	2.5	0	0	9	2.3-02
54'-0"	3108	5	10.5	5.3-02	312	7.8-01
54'-0"	3109	5	0	0	9	1.1
54'-0"	3112	3	13.5	4.1-02	444	1.3
54'-0"	3113	50	0	0	9	4.5-01
54'-0"	3114	3	6	1.8-02	144	4.3-01
54'-0"	3115	3	6	1.8-02	147	4.4-01
54'-0"	3116	50	0	0	9	4.5-01
54'-0"	3117	2.5	0	0	44	1.1-01
54'-0"	3118	120	0	0	9	1.1
54'-0"	3119	120	0	0	15	1.8
54'-0"	3120	25	12	3.0-01	246	6.2
54'-0"	3121	10	4.5	2.3-02	231	2.4
54'-0"	3122	50	0	0	12	6.0-01
54'-0"	3123	50	0	0	12	6.0-01
54'-0"	3124	10	12	6.0-02	237	2.4
54'-0"	3128	10	15	1.5-01	291	2.9
54'-0"	3129	300	0	0	9	2.7
54'-0"	3130	300	0	0	9	2.7
54'-0"	3132	40	0	0	9	3.6-01
54'-0"	3133	10	6	6.0-02	96	9.6-01
54'-0"	3134	10	18	1.8-01	192	1.9
54'-0"	3135	50	0	0	3	1.5-01
54'-0"	3136	35	1.5	5.3-02	9	3.2-01
54'-0"	3137	35	1.5	5.3-02	9	3.2-01
54'-0"	3138	3	13.5	4.1-02	300	9.0-01
54'-0"	3141	2.5	12	3.0-02	189	4.7-01
54'-0"	3142	2.5	0	0	6	1.5-02
54'-0"	3143	3	9	2.7-02	447	1.3
54'-0"	3145	--	0	0	0	0
54'-0"	3146	--	0	0	0	0
54'-0"	3147	1	7.5	7.5-03	93	9.3-02
54'-0"	3148	10	0	0	3	3.0-02
54'-0"	3150	13	0	0	3	3.9-02
54'-0"	3151	2.5	10.5	2.6-02	138	3.5-01
54'-0"	3153	2.5	12	3.0-02	16	4.0-02
54'-0"	3155	--	0	0	0	0
54'-0"	3156	--	0	0	0	0
54'-0"	3158	--	0	0	0	0

(Historical Information)

TABLE 12.4-8 (Cont)

Elevation	Room# (3)	Estimated Dose Rate (mrem/h) (1)	Periodic Testing Estimated Annual Person-Hours	Periodic Testing Estimated Annual Person-Rem	Routine Maintenance (2) Estimated Annual Person-Hours	Routine Maintenance (2) Estimated Annual Person-Rem
54'-0"	3159	--	0	0	0	0
54'-0"	3160	--	0	0	0	0
54'-0"	3161	--	0	0	0	0
54'-0"	3164	--	0	0	0	0
54'-0"	3166	60	1.33	1.0-01	0	0
54'-0"	3167	60	14.8	8.9-01	136	8.0
54'-0"	3168	60	10.5	6.3-01	117	7.0
54'-0"	3170	3	6	1.8-02	300	9.0-01
54'-0"	3171	--	0	0	0	0
54'-0"	3173	--	0	0	0	0
54'-0"	3175	10	1.5	1.5-02	33	3.4-01
54'-0"	3176	25	15	3.8-01	162	4.1
54'-0"	3177	5	18	9.0-02	81	4.0-01
54'-0"	3180	-	0	0	0	0
54'-0"	3181	10	4.5	4.5-02	219	2.2
54'-0"	3182	--	0	0	0	0
54'-0"	3183	2.5	3	7.5-03	348	8.7-01
54'-0"	3186	--	0	0	0	0
54'-0"	3187	--	0	0	0	0
54'-0"	3189	--	0	0	0	0
54'-0"	3190	--	0	0	0	0
54'-0"	3191	4	15	6.0-02	335	1.3
54'-0"	3192	20	6	1.2-01	63	1.3
54'-0"	3193	4	3	1.2-02	167	6.6-01
54'-0"	3194	20	6	1.2-01	63	1.3
54'-0"	3195	4	3	1.2-02	167	6.6-01
54'-0"	3196	4	16.5	6.6-02	288	1.2
102'-0"	3317	--	0	0	0	0
102'-0"	3318	--	0	0	0	0
102'-0"	3319	--	0	0	0	0
102'-0"	3320	--	0	0	0	0
102'-0"	3321	35	3	1.1-01	0	0
102'-0"	3322	50	3	1.5-01	0	0
102'-0"	3323	50	3	1.5-01	0	0
102'-0"	3324	2.3	58.5	1.3-01	456	1.0
102'-0"	3325	2.3	18	4.1-02	315	7.2-01
102'-0"	3326	2.3	16.5	3.8-02	351	8.1-01
153'-0"	3603	14	0	0	63	8.8-01
153'-0"	3606	14	3	4.2-02	87	1.2
Off-gas System		--	0	0	780	[BS]2.04+01
Total			409.1	4.4	8,527	91.8

(Historical Information)

TABLE 12.4-8 (Cont)

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(1) Assumes equipment in the room is shut down.
(2) The estimated exposures assume that all person-hours are expended in the areas in which they appear. Portions of these person-hours may be spent at lower radiation levels within the area. Components may also be removed to a lower background area for maintenance.
(3) For room locations, see Figures 12.3-1 through 12.3-8.

(Historical Information)

TABLE 12.4-9

DIRECT EXPOSURE ESTIMATES DUE TO ROUTINE SURVEILLANCE

Building	Area	Annual Person-Hours (1)	Estimated Dose (Person-Rem/Yr) (2)
Turbine bldg	Condenser area	39.8	0.17
	Steam jet air ejector area	0 (remote viewing via mirrors)	0
	Mechanical vacuum pump area	0 (remote viewing via mirrors)	0
	Turbine hall	191.3	0.83
	Other areas	503.0	2.2
	Total	734	3.2
Reactor Bldg	Reactor water clean-up pump area	26.8	0.28
	Reactor water filter/ demineralizer area	14.6	0.15
	Refueling area	0	0
	Emergency core cooling systems	308.5	3.2
	Drywell area	0	0
	Other areas	417.0	4.3
	Total	767	7.9
Radwaste areas	Liquid radwaste areas	229.5	0.81
	Solid radwaste areas	75.5	0.26
	Other areas	151.0	0.53
	Total	456	1.6

(1) Includes transit time.
(2) Includes exposure received during transit and during room entries.

(Historical Information)

TABLE 12.4-10

ESTIMATED TURBINE BUILDING INHALATION EXPOSURES DUE TO AIRBORNE RADIOACTIVITY

Room (3)	Estimated Annual Person-Hours			Estimated Thyroid Dose (1) (person-rem/yr)	Estimated Tritium Dose (2) (person-rem/yr)
	Testing	Maintenance	Total		
Condenser area					
1113	0	0	0	0	0
1114	0	150	150	1.2-1	8.1-4
1115	0	150	150	1.2-1	8.1-4
1116	0	150	150	1.2-1	8.1-4
1117	48	429	477	3.8-1	2.6-3
1220	6	146	152	1.2-1	8.2-4
1310	0	0	0	0	0
1311	0	0	0	0	0
1312	0	0	0	0	0
1313	121.5	2,909	3,030.5	2.4	1.6-2
1405	12	149	161	1.3-1	8.7-4
Steam jet air ejector area					
1218	13.5	345	358.5	5.0-1	3.3-3
1219	3	204	207	2.9-1	1.9-3
Mechanical vacuum pump area					
1206	22.5	143	165.5	9.6-2	6.5-4
Turbine hall					
1512	7.5	107	114.5	3.2-2	2.2-4
1513	36	465	501	1.4-1	9.5-4
1514	0	0	0	0	0
Other areas					
1102	42	1,127	1,169	3.2-1	2.1-3
1103	0	17	17	4.6-3	3.1-5
1104	24	708	732	2.0-1	1.3-3
1106	3	300	303	8.2-2	5.5-4
1107	84	1,410	1,494	4.0-1	2.7-3
1108	3	156	159	4.3-02	2.9-4
1110	0	0	0	0	0
1111	13.5	860	873.5	2.4-1	1.6-3
1112	27	142	169	4.6-2	3.0-4
1128	9	508	517	1.4-1	9.3-4
1207	3	329	332	9.0-2	6.0-4

(Historical Information)

TABLE 12.4-10 (Cont)

Room (3)	Estimated Annual Person-Hours Testing	Maintenance	Total	Estimated Thyroid Dose (1) (Person-rem/yr)	Estimated Tritium Dose (2) (person-rem/yr)
1208	0	0	0	0	0
1209	1.5	54	55.5	1.5-2	1.0-4
1210	1.5	45	46.5	1.3-2	8.4-5
1211	1.5	54	55.5	1.5-2	1.0-4
1212	1.5	54	55.5	1.5-2	1.0-4
1213	3	54	57	1.5-2	1.0-4
1214	1.5	54	55.5	1.5-2	1.0-4
1215	1.5	54	55.5	1.5-2	1.0-4
1216	9	240	249	6.7-2	4.5-4
1230	0	318	318	8.6-2	5.7-4
1305	22.5	374	396.5	1.1-1	7.1-4
1308	30	388	418	1.1-1	7.5-4
1309	24	376	400	1.1-1	7.2-4
1504	9	164	173	4.7-2	3.1-4
1505	10.5	105	115.5	3.1-2	2.1-4
1506	3	120	123	3.3-2	2.2-4
1508	3	458	461	1.2-1	8.3-4
1509	9	284	293	7.9-2	5.3-4
1510	10.5	339	349.5	9.4-2	6.3-4
1511	13.5	340	353.5	9.2-2	6.4-4
	634.5	14,779	15,413.5	7.44	0.049

(1) Due to inhalation of radioiodines
(2) Whole body exposure due to tritium uptake
(3) For room locations see Figures 12.3-16 through 12.3-28.

(Historical Information)

TABLE 12.4-11

ESTIMATED REACTOR BUILDING INHALATION EXPOSURES DUE TO AIRBORNE RADIOACTIVITY

Room[(2)]	Estimated Annual Person-Hours Testing	Maintenance	Total	Estimated Thyroid Dose[(1)] (person-rem/yr)
Reactor water cleanup pump areas				
4319	0	14	14	8.8-3
4402	6	68	74	4.7-2
4403	7.5	132	139.5	8.8-2
4405	7.5	132	139.5	8.8-2
4505	3	77	80	5.0-2
4506	6	182	188	1.2-1
4509	0	27	27	1.7-2
Reactor water cleanup/filter demin area				
4406	9	176	185	9.8-2
4407	0	9	9	4.8-3
4502	13.5	279	292.5	1.6-1
4503	13.5	279	292.5	1.6-1
4620	0	0	0	0
4621	0	0	0	0
Emergency Core Cooling Systems				
4101	10.5	262	272.5	3.0-2
4102	36	1,994	2,030	2.2-1
4104	0	0	0	0
4105	0	0	0	0
4107	0	0	0	0
4109	9	409	409	4.6-2
4110	24	610	634	7.0-2
4111	33	694	727	8.1-2
4113	18	366	384	4.2-2
4114	9	0	9	9.9-4
4116	0	0	0	0
4118	0	0	0	0
4128	0	0	0	0
4208	7.5	195	202.5	2.2-2
4214	7.5	172	179.5	2.0-2
4215	0	0	0	0
4327	0	42	42	4.6-3

TABLE 12.4-11 (Cont)

Room[2]	Estimated Annual Person-Hours Testing	Maintenance	Total	Estimated Thyroid Dose[1] (person-rem/yr)
Others				
4106	18	328	346	3.3-2
4115	18	328	346	3.3-2
4316	15	602	617	5.9-2
4318	0	52	52	5.0-3
4320	228	963	1,191	1.1-1
4321	7.5	358	365.5	3.5-2
4326	0	0	0	0
4328	4.5	1,152	1,156.5	1.1-1
4329	3	271	274	2.6-2
4601	12	150	162	1.5-2
4603	0	44	44	4.2-3
4615	13	0	13	1.2-3
4625	9	147	156	1.5-2
4626	9	120	129	1.2-2
4627	9	75	84	8.0-3
4628	9	72	81	7.7-3
Drywell area				
4220	0	604	604	3.9
4221	12	623	525	4.1
4341	75	2,939	2,939	1.9+1
Total	653.5	14,947	15,600.5	28.7

(1) Due to inhalation of radioiodines.
(2) For room numbers see Figures 12.3-9 through 12.3-15.

(Historical Information)

TABLE 12.4-12

ESTIMATED RADWASTE AREAS INHALATION EXPOSURES DUE TO AIRBORNE RADIOACTIVITY

Room (3)	Estimated Annual Person-Hours Testing	Maintenance	Total	Estimated Thyroid Dose (1) (person-rem/yr)	Estimated Lung Dose (2) (person-rem/yr)
Liquid radwaste handling areas					
3103	0	0	0	0	0
3104	0	0	0	0	0
3105	0	0	0	0	0
3107	0	9	9	1.2-3	1.1-4
3108	10.5	312	322.5	4.2-2	3.9-3
3109	0	9	9	1.2-3	1.1-4
3113	0	9	9	1.2-3	1.1-4
3114	6	144	150	2.0-2	1.8-3
3115	6	147	153	2.0-2	1.8-3
3116	0	9	9	1.2-3	1.1-4
3121	4.5	231	235.5	3.1-2	2.8-3
3122	0	12	12	1.6-3	1.4-4
3123	0	12	12	1.6-3	1.4.4
3124	12	237	249	3.2-2	3.0-3
3136	1.5	9	10.5	1.4-3	1.3-4
3137	1.5	9	10.5	1.4-3	1.3-4
3138	13.5	300	313.5	4.1-2	3.8-3
3145	0	0	0	0	0
3146	0	0	0	0	0
3149	0	0	0	0	0
3150	0	3	3	3.9-4	3.6-5
3151	10.5	138	148.5	1.9-2	1.8-3
3153	12	16	28	3.6-3	3.4-4
3171	0	69	69	9.0-3	8.3-4
3172	0	0	0	0	0
3173	0	69	69	9.0-3	8.3.4
3174	0	0	0	0	0
3175	1.5	33	34.5	4.5-3	4.1-4
3176	15	162	177	2.3-2	2.1-3
3177	18	81	99	1.3-2	1.2-3
3191	15	335	350	4.6-2	4.2-3
3192	6	63	69	9.0-3	8.3-4
3193	3	167	170	2.2-2	2.0-3
3194	6	63	69	9.0-3	8.3-4
3195	3	167	170	2.2-2	2.0-3
3196	16.5	288	304.5	4.0-2	3.7-3
3316	0	0	0	0	0
3320	0	0	0	0	0
3321	3	0	3	3.9-4	3.6-5
3325	18	315	333	4.3-2	4.0-3

(Historical Information)

TABLE 12.4-12 (Cont)

Room (3)	Estimated Annual Person-Hours Testing	Maintenance	Total	Estimated Thyroid Dose (1) (person-rem/yr)	Estimated Lung Dose (2) (person-rem/yr)
Solid radwaste handling areas					
3118	0	9	9	1.4-4	1.4-5
3119	0	15	15	2.4-4	2.3-5
3120	12	246	258	4.1-3	3.9-4
3128	15	291	306	4.9-3	4.6-4
3129	0	9	9	1.4-4	1.4-5
3130	0	9	9	1.4-4	1.4-5
3132	6	9	9	1.4-4	1.4-5
3133	6	96	102	1.6-3	1.5-4
3134	18	192	210	3.4-3	3.2-4
3135	0	3	3	4.8-5	4.5-6
3147	7.5	93	100.5	1.6-3	1.5-4
3148	0	3	3	4.8-5	4.5-6
3309	0	0	0	0	0
3311	0	0	0	0	0
3322	3	0	3	4.8-5	4.5-6
3323	3	0	3	4.8-5	4.5-6
3326	16.5	351	367.5	5.9-3	5.5-4
3096	0	0	0	0	0
3103	0	9	9	2.5-4	2.3-5
3102	9	138	147	4.1-3	3.8-4
3112	13.5	444	457.5	1.3-2	1.2-3
3117	0	44	44	1.2-3	1.1-4
3141	12	189	201	5.6-3	5.2-4
3142	0	6	6	1.7-4	1.6-5
3143	9	447	456	1.3-2	1.2-3
3155	0	0	0	0	0
3156	0	0	0	0	0
3158	0	0	0	0	0
3159	0	0	0	0	0
3160	0	0	0	0	0
3161	0	0	0	0	0
3165	0	0	0	0	0
3166	1.3	0	1.3	3.6-5	3.4-6
3167	14.8	136	150.8	4.2-3	3.9-4
3168	10.5	117	127.5	3.6-3	3.3-4
3170	6	300	306	8.6-3	8.0-4
3180	0	0	0	0	0
3181	4.5	219	223.5	6.3-3	5.8-4
3182	0	0	0	0	0
3183	3	348	351	9.8-3	9.1-4
3187	0	0	0	0	0
3189	0	0	0	0	0

(Historical Information)

TABLE 12.4-12 (Cont)

Room (3)	Estimated Annual Person-Hours Testing	Maintenance	Total	Estimated Thyroid Dose (1) (person-rem/yr)	Estimated Lung Dose (2) (person-rem/yr)
3198	0	0	0	0	0
3199	0	0	0	0	0
3317	0	0	0	0	0
3318	0	0	0	0	0
3319	0	0	0	0	0
3324	58.5	456	514.5	1.4-2	1.3-3
3603	0	63	63	8.1-3	7.6-4
3606	3	87	90	1.2-3	1.1-3
Total	409.1	7,747	8,156.1	0.60	0.054

(1) Due to inhalation of radioiodines.
(2) Due to Co-60.
(3) For room locations, see Figures 12.3-1 through 12.3-8.

(Historical Information)

TABLE 12.4-13

AIRBORNE EXPOSURE ESTIMATES DUE TO ROUTINE SURVEILLANCE

Building	Area	Annual Person- Hours	Estimated Thyroid Dose Person -rem/yr(1)	Estimated Lung Dose Person -rem/yr(2)	Estimated Whole Body Dose Due to Tritium, Person-rem/yr(3)
Turbine Building	Condenser area	11.4	9.0-3	-	6.1-5
	Steam jet air ejectors	0 (remote view via mirrors)	0	-	0
	Mechanical vacuum pumps	0 (remote view via mirrors)	0	-	0
	Turbine hall	54.8	1.5-2	-	1.0-4
	Other areas	144.1	3.8-2	-	2.6-4
	Total	210.3	6.2-2	-	4.2-4
Reactor Building	Reactor water cleanup pump areas	6.6 (based on remote look-in)	4.1-3	-	-
	Reactor water cleanup filter/ demineralizers	3.6 (based on remote look-in)	1.9-3	-	-
	Refueling area	0	0	-	-
	Emergency core cooling systems	75.9	8.4-3	-	-
	Drywell area	0	0	-	-
	Other areas	102.6	9.8-3		
	Total	188.7	2.4-2		
Radwaste areas	Liquid radwaste areas	54.4	7.0-3	6.5-4	-
	Solid radwaste areas	17.9	2.9-4	2.7-5	-
	Other areas	35.8	1.0-3	9.3-5	-
	Total	108.1	8.3-3	7.7-4	-

(1) Due to inhalation of radioiodines.
(2) Due to Co-60.
(3) Whole body exposure due to tritium update.
(4) A "-" indicates no estimated dose rates.

(Historical Information)

TABLE 12.4-14

ESTIMATED EXPOSURE FOR OPERATORS IN CONTROL ROOMS

	Operators Per Shift	Shifts Staffed per Day	Annual Operator Person-Hours (1)	Estimated Dose Rate (mrem/h)	Annual Operator Person-rem
Main control room	4	3	35,040	0.25	8.8
Radwaste control room	1	3	8,760	0.25	2.2
Total	5		43,800		11.0

(1) Based on each operator spending 8 hours in the control room, and the control rooms being staffed 365 days per year.

(Historical Information)

TABLE 12.4-15

SKYSHINE SOURCE TERMS[1]

Source Description	Normal Water Chemistry Condition γ/s
High pressure turbine	4.14+9
Low pressure turbines	3.15+9
28-inch ϕ high pressure turbine inlet pipe[2]	3.34+10
42-inch ϕ high pressure turbine exhaust pipe[2]	2.30+10
42-inch ϕ cross-around pipe[2]	6.82+10
Moisture separators[2]	5.79+11

(1) These are all the "exposed" sources above the turbine operating deck, Elevation 137 feet, and account for component self-shielding.

(2) All photons are assumed to be at 6.2 MeV.

(Historical Information)

TABLE 12.4-16

DIRECT RADIATION DOSE RATES AT SITE BOUNDARY LOCATIONS(1)(2)

Receptor Point(1)		Distance From Origin (ft)	Normal Water Chemistry Dose Rate (3)(4) (mrem/h)
7.	N	3000	1.4-04
8.	E	4500	7.8-05
9.	S	2150	1.1-03
10.	W	1120	4.5-03

(1) Locations of receptor points and origin are shown on Figure 12.4-1. The distances given are the closest location to the origin within each of the four sectors.

2) Operation at 100 percent of rated power.

(3) Normal Water Chemistry (NWC) dose rate estimate based on model.

(4) Hydrogen Water Chemistry (HWC) 35 SCFM hydrogen. Dose rates are determined to be within 10CFR20 and 40CFR190 limits (reference Engineering Evaluation H-1-AX-MEE-1318 Rev 1).

(Historical Information)

TABLE 12.4-17

DIRECT RADIATION DOSE RATES AT PLANT STRUCTURES[2]

Receptor Point[1]		Normal Water Chemistry Shine Dose Rate (4) (5) (mrem/h)
1.	Guardhouse (Security Bldg.)	0.051
2.	Cooling tower	0.0044
3.	Intake structure	0.0040
4.	Sewage treatment plant	0.0024
5.	Administration Building	0.16[3]
6.	Main parking lot	0.035

(1) Locations of receptor points and origin are shown on Figure 12.4-1.

(2) Operation at 100 percent of rated power.

(3) Dose rate on the building roof slab. The 11 1/2-inch thick concrete roof slab will keep dose rates due to turbine shine below 0.01 mrem/h inside the Administration Building.

(4) Normal Water Chemistry (NWC) dose rate estimate based on model.

(5) Hydrogen Water Chemistry (HWC) 35 SCFM hydrogen. Dose rates are determined to be within 10CFR20 and 40CFR190 limits (reference Engineering Evaluation H-1-AX-MEE-1318 Rev 1).

Security Related Information
Figure withheld Under 10 CFR 2.390

PSEG Nuclear, LLC
HOPE CREEK GENERATING STATION

ISODOSES AND
RECEPTOR POINT EXPOSURES

Updated FSAR
Revision 11
November 24, 2000

Sheet 1 of 1
Fig. 12.4-1

(Historical Information)

12.5 RADIATION PROTECTION PROGRAM

12.5.1 Program Description

The Radiation Protection Program provides evaluation and documentation of site radiological conditions and ensures that every reasonable effort is made to maintain personnel exposures “as low as reasonably achievable” (ALARA) in accordance with requirements of 10CFR20, Regulatory Guides, and Technical Specifications. The program is designed to protect the public and plant personnel from unnecessary exposure to radiation and radioactive materials. The personnel responsible for the Radiation Protection Program are, in order of authority, the Hope Creek Site Vice President, the Plant Manager, the Radiation Protection Manager, Radiation Protection Superintendent, Radiation Protection Supervisors, and Radiation Protection Technicians.

12.5.1.1 Authority and Responsibility

The Plant Manager is responsible for maintaining and implementing the Radiation Protection Program and receives direct reports from the Radiation Protection Manager concerning the status of the program.

The Radiation Protection Manager (RPM) is responsible for managing the Radiation Protection department to meet station operational needs and radiological safety standards.

The Radiation Protection Manager manages the Radioactive Material Control program and implements the ALARA program as described in administrative procedures.

Radiation Protection Supervisors are responsible for planning, conducting, and supervising daily radiation protection activities.

Radiation Protection Technicians implement the radiation protection program under the supervision of Radiation Protection Supervisors.

Radiation Protection personnel have the authority to halt any work activity when, in their professional judgment, worker safety is being jeopardized or unnecessary personnel exposures are occurring.

(Historical Information)

In the absence of Radiation Protection supervision, the authorities of the above positions may be delegated in accordance with station radiation protection procedures to qualified supervisors or technicians.

12.5.1.2 Experience and Qualifications

The Radiation Protection Manager is familiar with the design features of nuclear power stations and possesses both the technical competence to establish radiation protection programs and the supervisory capability to direct the work of the professionals and technicians required to implement such programs.

The qualifications of the designated “Radiation Protection Manager” meet or exceed the requirements of Regulatory Guide 1.8, September 1975.

At least one member of the Radiation Protection Supervisor staff shall be designated as the backup “Radiation Protection Manager” in accordance with paragraph 4.4.4(d) of ANSI/ANS 3.1-1981.

The Radiation Protection Supervisors are qualified in accordance with ANSI/ANS 3.1-1981. They shall have a minimum of four years of experience in applied radiation protection, including two years of experience in a nuclear power plant or a nuclear facility.

The qualifications of the Radiation Protection Technicians meet or exceed the personnel requirements of ANSI/ANS 3.1-1981. Radiation Protection Technicians are additionally trained and qualified in accordance with administrative procedures.

12.5.2 Facilities, Equipment, and Instrumentation

Radiation protection facilities, equipment, and instrumentation were designed and acquired to meet the requirements of Regulatory Guides 8.3, 8.4, 8.8, 8.9, 8.12, 8.14, 8.15, 8.28, and 1.97 (specifically Table 2, Type E Variables for Environs Radiation and Radioactivity).

(Historical Information)

12.5.2.1 Radiation Protection and Radiochemistry Facilities

12.5.2.1.1 Access Control

HCGS has two general area classifications for radiological control purposes: the restricted area and the radiological control area (RCA). The restricted area is any area where access is controlled to protect all individuals from exposure to radiation or radioactive material. In general, the HCGS restricted area corresponds to the area inside the station security fence (protected area). The RCA, which is within the restricted area, features positive control over access, activities, and egress. Access is limited in accordance with operational requirements and individual training (in radiation protection). The RCA may include radiation areas, high radiation areas, very high radiation areas, contaminated areas, radioactive material storage areas, and airborne radioactivity areas. Entry to and exit from the permanent RCA is normally through two designated access control points, the Radiation Protection Supervision may temporarily designate other entrances and exits for various plant conditions, as necessary. The access control points, shown on Plant Drawing P-0035-0 and Figure 12.5-3, are located at elevations 124 and 137 feet in the service and radwaste areas of the Auxiliary Building. Self-survey personnel monitoring equipment, such as automated whole-body contamination monitors or Geiger-Mueller (G-M) type friskers, are located at the exit from the RCA.

12.5.2.1.2 Radiation Protection Facilities

The radiation protection office and workrooms are located near the access control points. Portable radiation survey instrumentation, as well as air monitoring and sampling equipment, self-reading dosimeters, and miscellaneous radiation protection supplies, are stored in these rooms. Radiation protection equipment used for routine counting of smears and air samples, such as end window G-M counters, alpha and beta scintillation detectors, gas flow proportional counters, and/or gamma spectroscopy are located in the radiation protection count room. The radiation protection office area is equipped for survey record keeping and RWP preparation. Respiratory and protective clothing equipment are stored and issued in this area.

(Historical Information)

Decontamination facilities at the access control area consist of showers, sinks, and decontamination agents. Sinks and showers drain to tanks for processing through the liquid radioactive waste system. Large-area "pancake" end-window G-M friskers are located at these areas for personnel contamination monitoring.

Cleaning of protective clothing will be provided by a vendor laundry facility. Protective clothing, laundered offsite, is selectively monitored for contamination, sorted, and stored at the clean clothing issue areas or the laundry storage room.

Equipment decontamination facilities are located at the 102-foot elevation of the Auxiliary Building. One room, consisting of an ultrasonic cleaner, will be used primarily for tools and small equipment. The other room is equipped to handle larger components. Equipment that cannot be completely decontaminated may be worked on in the restricted machine shop located adjacent to the equipment decontamination facilities.

Both controlled and uncontrolled locker areas are located near the control points. The uncontrolled locker areas are used by workers not entering the RCA. Workers entering the RCA on a RWP change into clean protective clothing in the controlled locker room. Adjacent to the uncontrolled locker rooms are toilets and washrooms, shower rooms, and drying areas.

(Historical Information)

Several changes have been made since Plant Drawing P-0035-0 was last issued. These changes include use of some rooms for purposes other than shown on the drawing, and movement of some portable equipment to other rooms. These changes will not be shown on permanently issued drawings at this time since no structural changes have been made and it would not be cost effective to do so. The following describes those changes approved at the time which differ from Plant Drawing P-0035-0.

The current staffing plan projects 393 personnel during normal operation and 583 during outage periods. This includes clerical and management personnel with some facilities within the Administration Building for their use. A 10 percent female population is assumed in the normal operation workforce exclusive of management and clerical personnel.

The total number of lockers planned for installation is 630 controlled and 570 uncontrolled. The capability for adding 20 controlled and 140 uncontrolled lockers within these locker areas at a later date also exists.

The following is a current listing of the use of various rooms. These changes resulted, in part, from the relocation of lockers and some portable equipment:

1. Room 3403 is designated for data communication and telephone equipment. Rooms 3415, 3416, 3434, 3437, and 3441 have been deleted to expand the access control point. Room 3446 has been converted to an auxiliary personnel decontamination area.

2. Room 3447 will contain a self-contained respiratory cleaning and drying unit.

3. Room 3509 will be used as an RWP posting area for plant staff and outage personnel.

4. Rooms 3523 and 3524 will be used as the control point dress-out area. Room 3525 will be used as the Radiation Protection Technician work area.

5. Room 3531 will be used for radiation protection supervisory offices due to the facilities for dosimetry being relocated to the In-Processing Center.

(Historical Information)

6. Rooms 3535, 3536, 3537, and 3538 have been deleted to expand the access control point.

7. Room 3546 will be used for Radiation Work Permit (RWP) generation and records.

8. Rooms 3552 through 3560 (except 3555) will be used as facilities for females. Uncontrolled lockers in this area exceed 10 percent of the staffing levels for normal and outage periods.

9. Room 3555 will be used as a Radiation Protection counting room. The redundancy of equipment between this room and room 3423 will ensure that normal and emergency needs are met.

12.5.2.1.3 Radiochemistry Facilities

The radiochemistry facilities at the 124-foot elevation are part of the general chemistry facilities and consist of a hot chemistry laboratory, and a counting room. The counting room is surrounded by 18-inch concrete shield walls. An emergency shower and eyewash is available in the radiochemistry laboratory.

12.5.2.1.4 The Security Center Facility

The security center facility contains the site access security control station guardhouse. This facility serves as the access and security control point to the site areas of both the Salem and Hope Creek units. Portal monitors, and/or large area "pancake" end-window G-M friskers, are maintained at this location as the final monitoring check prior to leaving the restricted areas.

Plant Drawings A-B102-0 and A-B103-0 show the access security control station.

12.5.2.1.5 In-Processing Center

This facility is common for both Salem and Hope Creek plants to support activities at both stations. This facility includes provisions for dosimetry issue and recordkeeping, whole body counting, and respirator fit testing.

(Historical Information)

12.5.2.2 Instruments and Equipment

Instrumentation for detecting and measuring radiation consists of counting room equipment, portable instrumentation, and air samplers. Capabilities for detecting alpha, beta, gamma, and neutron radiation are provided. Sufficient inventory is provided to accommodate use, repairs, and calibration. Instrument calibrations include electronic and radiation calibration of dose rate and count rate instruments, counting scalers, and portal monitors. It also includes the flow calibration of portable air samplers, such as AMS-3's, Hi-Vols, H-809 Lo-Vols, RAP and RAS pumps utilizing anemometers, flow meters and D/P gauges.

Electronic calibrations are performed utilizing various types of measuring and test equipment, such as current pulsers, digital volt meters, oscilloscopes, decade boxes and capacitance testers.

Radiation calibrations are performed utilizing several licensed and exempt radiation sources, depending on the type radiation the instrument detects. For gamma survey instruments, Cs-137, with an energy of 662 KeV, is the isotope chosen because it is comparable to the average gamma energy of the isotopes present in a typical primary reactor coolant sample.

Three Cs-137 sources are used for calibrations. The irradiators used are the 100 mCi Shepherd Model 28-5 and a 400 Ci Shepherd Model 89 box type, which is primarily used for instruments, and a 3 Ci Shepherd Model 81-8A tower, which is used to irradiate Optically Stimulated Luminescent (OSLs) and self-reading dosimeters (SRDs).

(Historical Information)

All irradiation sources are shielded consistent with the station ALARA program. These sources are controlled and secured under the appropriate administrative, radiation protection and physical security procedures.

Beta measuring survey instruments are calibrated and their efficiencies for beta detection determined with a depleted Uranium slab. This mylar covered slab has a standard dose rate and is commonly used for this application.

Beta counting equipment is radiation calibrated using electroplated Tc-99 planchet-sized sources.

Alpha counting scalers and alpha survey instruments are calibrated with electroplated sources, as well. These are either planchet sized or provided in detector sized jigs.

Neutron measuring instruments are calibrated with an encapsulated MRC 5.7 Ci AmBe source which is setup for irradiation on the source range in a borated polyethylene block jig.

Various low activity exempt check-sources ("button-type") are also used to verify instrument response.

All instruments are calibrated in compliance with the American National Standards Institute (ANSI) N323 - 1978, which establishes calibration methods for portable radiation protection instruments used for the detection and measurement of ionizing radiation and radioactive surface contamination.

(Historical Information)

Sufficient chemical supplies, chemistry laboratory equipment, and analytical instruments are available to perform the required sample preparations and analyses in support of radiation protection functions.

12.5.2.2.1 Chemistry Laboratories

The chemistry facility consists of two laboratories: one that handles low level or background level samples and a second that handles medium and high radiation samples.

The laboratories are equipped with constant air flow fume hoods. The fume hoods permit preparation and processing of contaminated samples under controlled conditions.

12.5.2.2.2 Counting Rooms

Plant system samples processed in the chemical laboratories for activity analysis and isotopic identification are transported to the chemistry counting room. Samples direct from the plant, such as air samples and smears, are transported to the radiation protection counting room. Equipment is available in both counting rooms for gross alpha, gross beta, and gross gamma activity measurements and for determination of the activity levels of specific isotopes. Both counting rooms are temperature controlled and the voltage supply is regulated for instrument stability.

Major instrumentation in both counting rooms includes a computer programmed multi-channel analyzer using germanium detectors, or other detectors appropriate for specific isotopic identification; gas flow proportional counters, with and/or without windows, for filter samples, smears, and planchetted water samples; and a gross beta-gamma counter. The chemistry counting room also contains a liquid scintillation counter for beta emitters such as tritium and carbon-14.

(Historical Information)

Background and efficiency checks are performed routinely. Counter plateaus are established to determine operating voltages. Calibrations for the isotopes are based on National Bureau of Standards (NBS) related guidelines. Conventional radionuclide reference standards will be used for calibrations.

12.5.2.2.3 Portable Survey Instruments and Equipment

Portable survey equipment is used primarily for conducting area surveys and for monitoring personnel throughout the plant. Some portable equipment is reserved for emergency use and is located in lockers at the access control point, the control room, the technical support center, and the offsite emergency operations facility.

The criteria for selection of the portable instruments include:

1. Ability of instrument to perform in its intended use with reliability and accuracy

2. Ease of calibration and repair

3. Interchangeability of components

4. Weight and size for user acceptance

5. Standard readouts and controls/adjustments to simplify training of users.

Portable instruments for routine plant use are provided to permit alpha, beta, gamma, and neutron radiation measurements, and for obtaining samples of surface and airborne contamination. Portable instruments for emergency use are provided to permit alpha, beta, and gamma radiation measurements for obtaining samples of surface and airborne contamination. PSEG Nuclear maintains and calibrates Radiation Protection equipment in accordance with manufacturer's recommendations, ANSI standards, and requisite regulations. Examples of types of Health Physics equipment include: gas flow Proportional counter, electronic dosimeter, area radiation monitor (ARM), ion chamber, dose rate meter, GM dose rate meter, neutron monitoring equipment, whole body contamination monitor, GM survey count rate instrument, alpha monitoring equipment, tool and equipment monitor, air sampler, and continuous air monitor. The frequency and methods of calibration are described in the applicable procedures.

12.5.2.2.4 Personnel Dosimetry

Personnel monitoring will be provided by the use of Optically Stimulated Luminescent (OSL) dosimeters, electronic dosimeters (EDs), direct reading pocket dosimeters, or calculations from area survey data and exposure times. Personnel monitoring is provided per 10CFR parts 20 and 34. The form of personnel monitoring depends on the type of radiation and the expected radiation level. Table 12.5-2 lists the planned quantities, sensitivities, and ranges of the TLDs and self-reading dosimeters to be used at HCGS as of 1989.

OSLs are normally used as the Dosimeter of Legal Record (DLR) for individuals who require monitoring. OSLs are used for beta, gamma, and neutron exposures and are normally processed off-site by a vendor service and evaluated on site. Whenever neutron dose determination is required, calculations using the area neutron dose rate or area neutron to gamma ratio and dosimeter readings may be used as a backup or a tracking method prior to permanent dosimetry processing, or may be used in place of neutron dosimetry if it is more accurate. Radiation protection practices include the movement of the normal whole-body badge or the use of multiple badges, in addition to the whole-body badge, when exposure to specific parts of the body may be greater than the general area dose rate. These badges will be issued at the access control point, when required by a RWP.

Electronic dosimeters are normally used to monitor gamma exposure. The results are used for specific ALARA job exposure evaluation, as well as to indicate current individual exposure status. Electronic dosimeter readings can also be used as permanent records, especially for individuals who do not require monitoring, or in case of lost or compromised DLR results. Electronic dosimeters are available at the access control point. Electronic dosimeters are response checked prior to issue, and are calibrated in accordance with industry standards.

(Historical Information)

Internally deposited radioactive material is evaluated with a whole body counter. The counter is sufficiently sensitive to detect in the thyroid, lungs, or gastrointestinal tract a fraction of the annual limit on intake of the relevant gamma emitting radionuclides. The whole body counter is calibrated on an annual basis using phantoms and standard sources containing various radionuclides covering the range of energies normally expected. The detectors are used in conjunction with a multi-channel analyzer, a computer, and printer, to obtain a permanent record. Lapel air samplers are available to aid in the assessment of airborne radiological environments.

12.5.2.2.5 Miscellaneous Instruments and Equipment

The following miscellaneous radiation protection equipment are available at one or more locations in the plant: contamination control supplies such as glove bags, containment tents, absorbent wipes, absorbent paper, rags, step off pads, rope, plastic sheets, plastic bags, tape, contaminated area signs, and protective clothing. Appropriate supplies are assembled into kits and situated throughout the plant to aid in the control of contaminated spills. Temporary shielding, such as concrete blocks, lead bricks, lead sheets, and lead wool blankets, is also available to reduce radiation levels.

An apparatus for quantitative fit testing of individuals involved in the respiratory protection program is available.

Portal monitors, whole-body friskers, and/or hand-held friskers with sensitive large-area "pancake" end-window G-M probes are positioned at the RCA exit points. The purpose of these devices is to control the spread of contamination. Other devices that prove to be of equal or greater sensitivity will be considered for use instead of those listed above. Personnel may use friskers within the plant to monitor themselves at any time, especially when leaving a contaminated area. Portal monitors, with friskers as backup, are used within the guardhouse as personnel leave the restricted area.

(Historical Information)

Portable ventilation systems equipped with HEPA filters or HEPA and sorbent filters, are available to minimize airborne contamination in highly contaminated areas.

Continuous air monitors (CAMs) monitor airborne concentration at specific work locations. These CAMs record trends or sudden changes in the airborne concentrations. While they are not intended for quantitative analysis, the fixed filter type can be used as a low volume grab air sample. The filter medium is removed and analyzed in more detail in the radiation protection counting room.

12.5.2.2.6 Personnel Protective Equipment

Special protective equipment such as coveralls, plastic suits, shoe covers, gloves, head covers, and respirators, including approved air purifying respirators, self-contained breathing apparatus (pressure demand), and airline respirators and hoods, are stored in various plant locations and clothing change areas. This equipment is used to prevent both deposition of radioactive material internally or on body surfaces and the spread of contamination. Most areas of the plant are kept free of contamination so that no special protective equipment is needed. Contaminated areas are identified with posted signs. Radiation signs and radiation work permits (RWPs) are the primary means of defining the equipment required to enter these contaminated areas.

A variety of combinations of protective equipment may be prescribed, depending on the nature and level of possible contamination. For example, cotton clothes may be adequate, but in wet areas, plastic rain suits or bubble suits may be prescribed. Respirators may be used if airborne hazards exist, or if surface contamination could cause an airborne hazard as defined in the radiation protection procedures.

The use of Delta Protection Mururoa V4 F1 and V4 MTH2 respiratory protection suits has been authorized for use at Hope Creek with an assigned protection factor (APF) of 2,000 (Reference NRC to PSEG letter: "HOPE CREEK GENERATING STATION AND SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 -REQUEST FOR AUTHORIZATION TO USE RESPIRATORY PROTECTION EQUIPMENT (TAC NOS. MD9199, MD9200, AND MD9201)", dated January 27, 2009). Approval was based on testing which demonstrated the suits met European Standard EN 1073-1 (January 1998), "Protective Clothing Against Radioactive Contamination, Part 1: Requirements and Test Methods for Ventilated Protective Clothing Against Particulate Radioactive Contamination." This standard is generally consistent with the pertinent acceptance criteria provided in Los Alamos National Laboratory Report LA-10156-MS, which is used to test and authorize the use of air-supplied suits at Department of Energy sites. The certification-testing was broadly based covering a range of various functional areas. Both models passed all required tests, and both provided a measured average protection level (fit factor) of 50,000. The following information on the Mururoa V4 F1 and V4 MTH2 suits is included to comply with commitment CM.CC.2008-121:

The manufacturer's instructions for use and storage of the Delta Protection Mururoa V4F1 and V4 MTH2 suits will be adhered to and integrated into the respiratory protection program, with the exception of the requirement to have a stand-by rescue person. New lesson plans will be developed to train workers on Mururoa's features, donning, use and removal, cautions and use of mouth strip and tear off strips for routine and emergency egress. Radiation Protection personnel will be provided additional training for selection, approval, issue, equipment set-up, operation and maintenance instructions for the Mururoa suit. The Mururoa V4F1 and V4 MTH2 suits will be discarded after a single use and will not be used in atmospheres that are immediately dangerous to life and health (IDLH). Any defects discovered with the Mururoa suit will be entered into the Corrective Action Program and reported to the manufacturer, as necessary. Industry notifications, when required, will be made through the Operating Experience Program.

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(Historical Information)

12.5.3 Procedures

Radiation protection procedures, as described in this section, are implemented by Hope Creek radiation protection instructions, administrative procedures, ALARA procedures, and emergency plan procedures. The procedures are written to meet the guidelines of Regulatory Guides 1.8, 1.33, 1.39, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.20, 8.26, 8.27, 8.29 and Hope Creek Technical Specifications.

12.5.3.1 Radiological Surveys

Area survey procedures describe the purpose and techniques of detecting and measuring levels of radiation and contamination. Contamination may be on surfaces or airborne. Area surveys are conducted throughout the plant. Such surveys may be routine or may be related to specific jobs. An area survey may be performed before, during, and/or after various work activities. Area surveys are performed by radiation protection personnel.

12.5.3.1.1 Radiation Detection

The preferred instrument for beta-gamma dose rate measurements is an ion chamber. G-M probes are preferred for measurement of low radiation levels or where environmental conditions such as temperature or humidity could cause erratic responses from ion chambers.

The preferred neutron measurement instrument is a rem counter, or equivalent, that has the ability to measure neutron dose rate in rem per hour.

These radiation detection methods are supplemented by continuous area and process radiation monitoring equipment with alarm capabilities, as described in Sections 11.5 and 12.3.4.

12.5.3.1.2 Surface Contamination Detection

A variety of techniques are used to detect and measure radioactive contamination. Procedures prescribe the use of smears (small paper discs) and large area wipes (Maslin-type cloths) to wipe a surface to pick up removable contamination. Fixed contamination is determined by scanning a surface with portable survey meters.

(Historical Information)

G-M probes are used for beta-gamma measurements and alpha detectors are used to distinguish the alpha component.

12.5.3.1.3 Airborne Contamination

Airborne contamination is normally determined by using air samplers to draw a known volume of air through a filter paper or charcoal cartridge. A charcoal cartridge is used with filter paper where iodine is of concern. The filter paper and charcoal cartridge are analyzed by gross beta-gamma count and/or gamma spectrometry. The gamma spectrometry identifies the particulate and radioiodine isotopic activity. Gross beta-gamma count data is used to judge the need for gamma spectrometry. High volume air samplers and low volume air samplers having nominal sample rates of 25 scfm and 2 scfm, respectively, are available. The high volume air sampler is used primarily to quickly obtain grab samples before, during, and after work activities. The low volume air sampler is used primarily to obtain the average air concentration for the work period.

In-plant sampling for radioiodine will be performed in accordance with the PSEG Nuclear Emergency Plan and Emergency Plan Implementing Procedures. Equipment to perform this in-plant radioiodine sampling will be kept in Emergency Lockers or in the vicinity of Emergency Lockers, in accordance with Section 9 of the Emergency Plan and Emergency Plan Inventory Procedure(s).

(Historical Information)

Prior to analysis, all silver zeolite cartridges analyzed inplant will be purged using bottled nitrogen gas or clean air (i.e., free of noble gases) to ensure absence of noble gases (i.e., xenon). Purging the cartridges will be performed in a well ventilated area or under a laboratory hood.

Analyses of in-plant and onsite silver zeolite cartridges for radioiodine will normally be performed using high purity germanium detectors (HPGe). These detectors will be located in the Chemistry Laboratory (124 ft el) and the Radiation Protection Count Room (137 ft el). Analyses of offsite silver zeolite cartridges will be performed using the HP210 frisker probe with an E-140 portable count rate meter or equivalent. The HPGe detectors at the Salem Station Chemistry Laboratory may be used to analyze silver zeolite cartridges for in-plant or onsite samples should the background radiation levels in the Chemistry Laboratory or Radiation Protection Count Room be too high to perform analyses with the HPGe.

12.5.3.1.4 Survey Frequency and Techniques

Each area found to have a radiation dose rate such that an individual could receive 5 mrem in any one hour, is conspicuously posted as a radiation area, in accordance with 10CFR20. Every reasonable effort is made to minimize inadvertent entries into such areas. Routine surveys of all radiation areas are taken to ensure that each area is surveyed on a regular basis. Areas subject to variations in radiation levels and occupancy times may be surveyed on a more frequent basis. When reactor conditions are operationally stable, survey frequency in radiation areas may be reduced to spot checks at boundaries to minimize radiation protection personnel exposures.

(Historical Information)

Each area found to have a radiation dose rate equal to or greater than 100 mrem/h is posted as a high radiation area and access is controlled in accordance with 10CFR20 and technical specifications. Routine surveys within such areas are not normally performed with conventional portable survey instruments. Every reasonable effort is made to use readings from the Radiation Monitoring System (RMS) area radiation monitors to identify changes of radiation levels. Measurement of maximum and general radiation levels within high radiation areas is normally performed with remote probe survey instruments, long reach survey instruments, retrievable TLDs, or dosimeters. When practicable, findings from these surveys are correlated to the appropriate RMS area radiation monitor readings and reactor operating conditions. Correlation readings and/or perimeter readings are taken to ensure that each high radiation area is surveyed on a regular basis. In addition, the frequency of radiation surveys taken at entrances to high radiation areas is dependent upon occupancy in the vicinity and variation in radiation levels. If surveys at entrances or RMS readings show significant change, additional surveys may be performed to update the surveys for the area. In order to minimize occupational exposure of surveyors, high radiation area survey frequency may be reduced when operating conditions are stable.

Areas in and around the RCA not considered potential radiation areas are selectively surveyed to establish that every reasonable effort has been made to keep measurable radiation at ALARA levels. Portable instrument surveys are performed to ensure that a representative number of non-radiation areas are surveyed once per month. Areas subject to significant radiation change or variation are surveyed on a more frequent basis, as appropriate. Any area not previously noted that is found to be a radiation area is promptly posted with a "Caution Radiation Area" sign and reported to Radiation Protection Supervision. If the radiation field cannot be eliminated, every reasonable effort is made to minimize the dose rate and inadvertent entry. The area is placed on the radiation area survey list.

(Historical Information)

Areas within the Hope Creek security fence not covered by normal portable instrument surveys are selectively monitored by area TLDs to document integrated exposures and are supplemented by randomly scheduled instrument surveys.

Procedures for area surveys describe the use of instruments, effective survey techniques, and documentation of data. Procedures allow consideration for potential as well as actual radiological hazards.

12.5.3.2 Radiation Work Permits

Where radiation dose rates, airborne concentrations, or surface contamination levels exceed station administrative control levels, a radiation work permit (RWP) is issued prior to scheduled work. This is accomplished by submittal of a RWP request form to radiation protection. Radiation protection evaluates the radiological conditions associated with the work to be performed. Based on this analysis, radiation protection may issue an extended radiation work permit if the work is of a repetitive nature and no significant short term accumulation of person-rem is expected. Radiation protection specifies the appropriate protective clothing, equipment, and monitoring, including dosimetry. Work area survey frequency is established by radiation protection. All personnel performing work under a particular RWP must be familiar with permit conditions and must sign in on a register sheet. The register sheet identifies the individual, his time in and out, and his self-reading dosimeter value in and out. Radiation protection may terminate a RWP if radiological conditions change.

Radiation protection supervision selectively reviews completed RWPs before they are filed. RWPs serve as a data source for dose comparison on repeat jobs and can be used to determine the effectiveness of ALARA efforts.

An effective RWP program facilitates reporting requirements such as the annual report required by Hope Creek Technical Specifications.

(Historical Information)

12.5.3.3 Handling and Storage of Radioactive Material

Radiation protection personnel are notified of intended releases from the RCA shipment and/or receipt of radioactive material, special nuclear material, or potential radioactive material. This is done to ensure that the radiation protection personnel are aware of all radioactive materials onsite, so that required surveys can be performed, and to verify that correct labeling, placarding, and documentation has been performed on all potential radioactive materials.

Calibration sources for radiation instrumentation and sources used to prepare secondary standards are stored in vault(s). These vaults are kept locked. The locks are under the control of radiation protection management.

Small quantities of sealed or unsealed sources may be stored for convenience in shielded cabinets, caves, or safes in secured areas. Such sources are used in the chemistry laboratories, instrument calibration facilities counting rooms, or when checking instruments' response throughout the plant.

12.5.3.4 Whole-Body Counting

Whole body counting is normally performed on personnel recently exposed to radioactivity who begin employment on site. Annual recounts may be performed. Whole body counts may be performed when nasal or facial contamination is discovered, after a suspected internal exposure, or as determined necessary by a Radiation Protection Supervisor.

In addition to whole-body counting, urinalysis and fecal analysis may be used for a more definite analysis of actual internally deposited radioactive material.

12.5.3.5 Control of Access and Stay Time in Radiological Areas

The security checkpoint at the fence line perimeter is a continuously staffed central guardhouse. Individuals assigned an OSL ensure that the proper OSL is worn upon entering the Protected Area.

(Historical Information)

Any individual without clearance to enter the restricted area must be accompanied by a person who is authorized to do so. The training, retraining, and testing requirements for unescorted access are provided in Section 12.5.3.6.

12.5.3.5.1 Control of Radiation and High Radiation Areas

Radiation, high radiation, and very high radiation areas are identified by posted radiation signs. Supplemental signs may be used to inform individuals of requirements for entry in addition to RWP requirements. Where appropriate, yellow and magenta rope or tape is used to limit access or to divert personnel to a specific control point for access. RWPs are used to describe work activities in an area, to prescribe radiation protection clothing and equipment, and to document entry and exit of each individual. Station procedures describe the purpose and application of the RWPs. Administrative guidelines for personnel exposure are established by procedure. The initial administrative control level is nominally 2000 mrem per year, which is less than the exposure dose limits in 10CFR20. Deviation from these guidelines must be requested and approved. Procedures provide the steps for approval of dose extensions. Additionally, entry to high radiation areas and very high radiation areas is normally controlled by locked doors or gates. Keys for these high radiation doors are under administrative control of the radiation protection supervision and the operations superintendent. Each key used and the individual using the key are recorded. Certain work activities exposed to high dose rates may be monitored continuously to prevent personnel from inadvertently exceeding the recommended stay time determined at the start of such work. Personnel are advised to observe the reading on their electronic dosimeter frequently.

(Historical Information)

12.5.3.5.2 Contamination Control

Radioactive contamination can exist in radiologically controlled areas. Access to contaminated areas is confined to specific locations. Floor coverings called step off pads define these locations. Personnel wishing to enter contaminated areas must review the RWP to determine the radiation protection clothing required. The RWP or related survey data sheets contain stay time determining information based on actual or potential airborne direct radiation, and/or contamination levels. Derived Air Concentration (DAC) hour calculations are maintained for each individual whose internal exposure meets or exceeds administrative assessment criteria. These calculations include respiratory protection factors when applicable. Procedures provide instructions for selection and application of protective clothing and respirators under various specific conditions. Procedures are also established for inspection, cleaning, and maintenance of such protective equipment. As personnel leave the work area, they remove the protective clothing and respirators before stepping onto the step-off pad. Personnel must frisk themselves to ensure that no contamination has been transferred to their bodies or clothing. If the frisker alarms, radiation protection is required to be notified. Radiation protection will take appropriate actions to minimize further spread of contamination, and direct appropriate decontamination of effected areas and personnel.

When personnel contamination is noted, a radiation protection investigation appropriate to the incident will be performed. A contamination incident found to have caused a suspected intake of radioactive material will be promptly reported to appropriate supervision. When applicable, recommended methods to prevent recurrence will be forwarded to the Plant Manager for concurrence and implementation by his directive.

Potentially contaminated tools, trash, and equipment to be transported from a contaminated area must be surveyed or bagged at the step off pad. Contaminated material must be placed in an externally non-contaminated container and appropriately labeled. The material may be placed in storage, taken to another contaminated area for reuse, or designated for appropriate disposal or decontamination. Clean material may be released for use within the station. Refer to Section 12.5.3.3 for unconditional release if the material is to be removed from the restricted area.

(Historical Information)

The presence of radioactive contamination, whether surface or airborne, inhibits mobility of personnel within the plant. Protective clothing that must be worn creates inconveniences and introduces other factors that affect performance. For these reasons, plus the obvious potential external and internal radiological hazards, decontamination is initiated judiciously to confine the contamination levels thus minimizing protective requirements. Special coatings that aid in decontamination are applied to walls and floors. The ventilation flow pattern is from clean areas to contaminated areas. Process equipment is isolated in various cavities or cells. Components can be vented in a controlled manner, usually through filters to the plant vent, where flow rate and radioactivity are monitored. Highly contaminated equipment drains are piped to sumps to avoid the use of floor drains and attendant spillage of fluids onto the floor.

12.5.3.6 Radiation Protection Training Programs

Radiation protection training programs ensure that personnel who have unescorted access to the restricted areas possess an adequate understanding of radiation protection to maintain occupational radiation exposures as low as reasonably achievable. Special training or retraining is administered upon recommendation of the Radiation Protection Manager. The appropriate personnel schedule the necessary training and maintain training records.

12.5.3.6.1 Employee Training

Each individual, whether employee or contractor, must receive general employee training (GET) to be eligible for unescorted access to the Restricted Area. GET covers general site personnel response to emergency situations, basic radiation protection, quality assurance program requirements, security, and a general safety orientation. Those desiring unescorted access to the RCA must also receive radiation worker training (RWT). RWT covers detailed radiation protection principles and provides radiation worker instruction in exposure controls for direct radiation, airborne and contamination sources, the purposes served by different types of protective clothing, and how to properly don, remove, and dispose of such clothing. The type of access badges to be issued are based on this documentation. Retraining is administered annually and includes lessons learned items.

Individuals who have not received GET (i.e., visitors) shall be escorted by GET qualified individuals and shall receive pertinent site instructions with each entry. Training is not required for members of the public entering the Restricted Area.

(Historical Information)

Individuals who have not received RWT for unescorted access to the RCA shall be escorted by individuals who are currently so qualified and shall receive radiation protection training commensurate with the purpose for entry, prior to such entry, as determined by radiation protection supervision.

All individuals working in the Restricted Area are given instruction concerning prenatal radiation exposure, as defined in Regulatory Guide 8.13. Occupationally exposed visitors who enter the Restricted Area also receive these instructions commensurate with their purpose for entry.

12.5.3.6.3 Respiratory Equipment Training and Fit Test

Certain individuals may be required to wear specific respiratory equipment in the performance of their responsibilities. A separate training class will be conducted for them in the purpose, use, and limitations of specific respiratory protective equipment used at the site.

To be eligible for duty which requires the use of respiratory equipment, an individual must pass the respiratory fit test, must have received the respiratory equipment training, and must be medically certified as being capable of working safely while wearing respiratory equipment. A quantitative fit test will be used to prove a satisfactory respirator fit for different types of respirators. Procedures will be established that describe the technique and define acceptance criteria.

12.5.3.6.3 Radiation Protection Personnel Training

A radiation protection training program for radiation protection technicians is provided which meets the requirements of ANSI 3.1-1981. This program instructs new radiation protection technicians in operational and analytical radiation protection procedures and theories and fundamentals of radiation safety and familiarizes them with plant layout and systems. During successful completion of the modular training program for technicians, individuals receive further comprehensive instruction for the specific requirements of their positions, including training on Emergency Plan duties.

(Historical Information)

12.5.3.7 Radiation Protection Records

Radiation protection records, which are generated from procedural requirements of Sections 12.5.3.1 through 12.5.3.6, are maintained and retained to meet regulatory and technical specification requirements.

12.5.3.8 ALARA Program

Basic ALARA philosophies, policies and responsibilities are discussed in Section 12.1.

The PSEG Nuclear LLC ALARA program affects all elements of the radiation protection program. In addition to the ALARA program guidance in PSEG Nuclear LLC administrative procedures, many specific ALARA topics are covered in radiation protection department procedures such as the use of portable shielding, completion of ALARA reviews, and exposure reduction methods.

The program also specifies responsibilities, requirements, and documentation for such issues as:

1. Pre-job planning
2. Job performance evaluation
3. Post job review
4. Process review
5. ALARA review of procedures
6. Station ALARA committee.

TABLE 12.5-1

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TABLE 12.5-2

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SEE FIGURE 13.1-6d

PUBLIC SERVICE ELECTRIC AND GAS COMPANY HOPE CREEK NUCLEAR GENERATING STATION
RADIATION PROTECTION ORGANIZATION
HCGS UFSAR Revision 8, September 25, 1996   Figure 12.5-1

Figure F12.5-2 intentionally deleted.

Refer to Plant Drawing P-0035-0 in DCRMS

SECURITY RELATED
INFORMATION WITHHELD
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Figure F12.5-4 intentionally deleted.

Refer to Plant Drawing A-B102-0 in DCRMS

Figure F12.5-5 intentionally deleted.

Refer to Plant Drawing A-B103-0 in DCRMS