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DRAWINGS CITED IN THIS CHAPTER*

*The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

DRAWING*	SUBJECT
M-24-1 to -23	General Arrangements, Radiation Shielding Units 1 & 2
M-48A	Composite Diagram of Liquid Radwaste Treatment Processing Units 1 & 2

CHAPTER 12.0 - RADIATION PROTECTION

12.0 RADIATION PROTECTION

The design-basis shielding sources were determined using the conservative source model in Subsection 11.1.2.1.

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

12.1.1 Policy Considerations

It is the policy of Exelon Generation Company to maintain occupational radiation exposure as low as is reasonably achievable (ALARA), consistent with plant construction, maintenance, and operational requirements, and within the applicable regulations. Regulatory Guide 8.8, Sections C.1, C.3, and C.4 is used as a basis for developing the ALARA and radiation protection programs with the following exceptions: C1B page 8.8-6 - qualifications for radiation protection manager (RPM) job - the stations do not commit to requiring the RPM to take any type of certification exam.

Exelon Generation Company ALARA policy applies to total person-rems accumulated by personnel, as well as to individual exposures. Exelon Generation Company management provides the environment for this policy to function in a proper manner. Management's commitment to this policy is reflected in the design of the plant, the careful preparation of plant operating and maintenance procedures, the provision for review of these procedures and for review of equipment design to incorporate the results of operating experience, and most importantly, the establishment of an ongoing training program. Training is provided for all personnel (Subsection 13.2.1), so that each individual is capable of carrying out his responsibility for maintaining his own exposure ALARA consistent with discharging his duties and also that of others. The development of the proper attitudes and awareness of the potential problems in the area of health physics is accomplished by proper training of all plant personnel. The organizational structure related to assuring that occupational radiation exposure be maintained ALARA is described in Subsection 12.1.1.1.

12.1.1.1 Organization Structure

The operating organization structure of the Byron/Braidwood Stations (B/B) is described in Chapter 13.0. Reporting to the Radiation Protection Department Head are health physicists, supervisors, and technicians.

The Radiation Protection Department Head is responsible for the overall radiation protection and ALARA programs.

station manager. Periodic meetings are scheduled between the Radiation Protection Department Head the station manager to discuss radiation protection concerns. Also, Radiation Protection Department personnel periodically meet with the ALARA committee to discuss ALARA concerns. Several station departments (e.g., operations, maintenance, station management, etc.) participate in these meetings.

12.1.1.2 Personnel Activities and Responsibilities

The station Radiation Protection Manager is responsible for the health physics program and for handling and monitoring radioactive materials, including source and by-product materials. However, an Operations Supervisor, who holds at least a limited Senior Reactor Operator’s license, is responsible for handling new and spent fuel.

In the case of fuel handling operations that alter the configuration of the reactor core, supervisory personnel with either a limited Senior Reactor Operator (SRO) or SRO license are directly responsible for movement of the fuel.

In the case of fuel handling operations that do not alter the configuration of the reactor core, qualified management personnel (such as Reactor Services or others designated by the Operations Department), who report to an Operations Supervisor, are directly responsible for movement of the fuel.

12.1.1.3 Administration Concerns

The Byron/Braidwood administrative personnel have a considerable amount of experience, which was accumulated at operating stations. The health physics program is based on regulations and experience which includes or considers the following:

a. Detailed procedures are prepared and approved for radiation protection prior to reactor plant operation. Those procedures are a part of the station health physics program.

b. All incoming and outgoing shipments which may contain radioactive material are surveyed to assure compliance with 10 CFR 71, 10 CFR 73, and 49 CFR 100-180.

c. Radiological incidents are investigated and documented in order to minimize the potential for recurrence. Reports are made to the NRC in accordance with 10 CFR 20.

d. Periodic radiation, contamination, and airborne activity surveys are performed and recorded to document radiological conditions. Records of the surveys are maintained in accordance with 10 CFR 20.

e. Records of occupational radiation exposure are maintained and reports are made to the NRC as required by 10 CFR 20, and to individuals as required by 10 CFR 19.13.

f. Posted areas are segregated and identified in accordance with 10 CFR 20. A combination of administrative controls and physical barriers are utilized to control access to high and very high radiation areas in accordance with 10CFR20 and the Technical Specifications.

g. Personnel are provided with personnel radiation monitoring equipment to measure their radiation exposure in accordance with 10 CFR 20.

h. Process radiation, area radiation, portable radiation, and airborne radioactivity monitoring instrumentation are periodically calibrated as required.

i. Access control points are established to separate potentially contaminated areas from uncontaminated areas of the station.

j. Protective clothing is used as required to help prevent personnel contamination and the spread of contamination from one area to another.

k. Tools and equipment used in radiological posted areas are surveyed for contamination before removal to an uncontrolled area. Contaminated tools and equipment removed from a contaminated area are packaged as necessary to prevent the spread of contamination to uncontrolled areas.

l. Radiation work permits (RWP) are issued for certain jobs in accordance with the station radiation protection procedures. Jobs involving significant radiation exposure to personnel are preplanned. (Where conditions dictate a mock-up is used for practice to reduce exposure time on the actual job. The use of special tools and temporary shielding to reduce personnel exposure is evaluated on a job-by-job basis.)

m. A bioassay program is included as part of the health physics program. This program includes whole body counting and/or a urinalysis sampling program to measure the uptake of radioactive material.

n. An environmental radiological monitoring program is in operation to measure any effect of the station on the surrounding environment.

o. All significant radioactive effluent pathways from the station are monitored and records maintained.

p. There are no special lighting requirements for high radiation areas.

q. Known radiation sources are marked or identified as such in efforts to reduce personnel time in regions of the exposure field and increase personnel distance from the source of exposure. "Hot-spot" labels are utilized on some localized radiation sources as deemed appropriate.

12.1.2 Design Considerations

Careful design can contribute greatly to the reduction of occupational radiation exposures. Radiation protection design considerations include shielding radioactive components, reducing the need for maintenance, enhancing the accessibility of equipment, reducing the source strength relative to personnel through remote handling, minimizing leakage and streaming, providing adequate ventilation, and preflushing contaminated systems.

Byron/Braidwood radiation protection design considerations establish a practical basis for maintaining radiation exposures ALARA. The direction is established by a set of radiation protection design goals. Conservatively set criteria in facility and equipment design, experience from past designs and operating plants incorporated to improve the present design, and mechanisms established for design review, are implemented to fulfill the ALARA requirement. (Radiation protection design features which are provided to maintain personnel radiation exposures ALARA are described in Section 12.3.)

12.1.2.1 Radiation Protection Design Goals

Byron/Braidwood radiation protection design goals are directed to ensure compliance with the standards for radiation protection specified in 10 CFR 20. The following sequence of design goals was used as a basis for maintaining radiation exposures as low as is reasonably achievable.

a. Establish design dose rates for general access areas based upon Commonwealth Edison's experience and 10 CFR 20 regulations.

b. Determine the most severe mode of operation for each piece of equipment and section of pipe (Section 12.2).

c. Based upon source terms, determine the source for each piece of equipment or pipe (Section 12.2).

d. Determine shielding required to maintain design dose rates.

e. Determine advantages and disadvantages of equipment locations, orientation, and segregation.

f. Use predetermined guidelines and criteria for locating piping and penetrations (Section 12.3).

g. Make changes in design wherever practicable to achieve ALARA exposures.

12.1.2.2 Facility Design Considerations

Byron/Braidwood's radiation protection design goals are expanded to design objectives. These objectives are categorized into several radiation protection concerns, which are described in the following subsections. Station layout considers direct radiation (for this section, direct radiation is defined as scattered and unscattered gamma and/or neutron rays from a [several] nonairborne radiation source(s)), and ventilation considers airborne radioactivity (see Subsection 12.2.2.3). Health physics and access control are concerned with both direct and airborne radioactivity. Control of radioactive fluids and effluents is concerned with the processing and detection of radioactive materials. The assumptions of primary coolant activity listed according to isotope are given in Table 12.2-2. The majority of the other source terms were developed, from these activities.

The design objectives are coupled with operating experience to obtain an improved station design.

12.1.2.2.1 Station Layout (Shielding)

The shielding was arranged and designed to the following objectives:

a. A sufficient quantity of access paths (general access areas) are furnished to allow personnel access to equipment.

b. The radiation levels in general access areas are to be kept ALARA.

c. Sufficient shielding is provided to control the amount of direct radiation present in a general access area.

d. Radiation areas are classified into zones according to expected (maximum) radiation levels.

e. Segregation of radiation zones is employed whenever practicable.

f. Shielding must accommodate equipment removal and maintenance.

g. Radiation "hot spots" are to be expected along the face of some shielding walls due to penetration and embedded system piping (i.e., nonradioactive piping designed for the passage of air, steam, water, or oil). A radiation "hot spot" is a small area that has a higher dose rate than the surrounding areas. A "hot spot" has a set maximum value that is based upon the adjacent design dose rates.

h. The radiation protection design is to be based upon the design criteria given in Section 12.3.

12.1.2.2.2 Ventilation

The station ventilation systems aid in heat removal and control of airborne radioactivity. Ventilation systems are designed to direct potentially airborne radioactive material from occupied areas towards the station vent stack. The remaining HVAC systems have special functions (e.g., laboratory hood exhaust). The ventilation systems are described in greater detail in Section 9.4. The radiation protection aspects of the systems are discussed in Subsection 12.3.3.

12.1.2.2.3 Health Physics

The radiation protection design objectives for health physics are:

a. The station's radiation protection monitoring equipment is located (and is of sufficient quality) to detect excessive airborne radioactivity and high radiation levels.

b. Personnel radiation monitoring equipment is required to measure and record personnel radiation exposure.

c. Radioactive effluent release paths to the environment are monitored.

d. Facilities for analysis of radioactive samples are furnished.

e. Cleaning and decontamination facilities are provided for equipment and protective clothing.

f. Periodic radiation surveys are performed when required, such as for maintenance in radiation areas, receiving or shipping radioactive material, and decontamination and maintenance of equipment, parts, and tools.

12.1.2.2.4 Access Control

Access to radioactive equipment is designed so that with properly trained personnel, radiation exposures during all modes of station operation meet the ALARA requirements. Access to radiation areas is strictly controlled.

12.1.2.2.5 Control of Radioactive Fluids and Effluents

Radioactive fluids (liquids and gases) are contained and controlled to keep the release of radioactive materials to general access areas and the environment ALARA. This objective applies to drain liquids, airbornes, and process liquids and gases (e.g., reactor water, fuel pool water, radwaste water, and off-gas). The number of release paths is minimized in order to simplify control.

12.1.2.2.6 Safety Objectives

a. The 10 CFR 20 limits are maintained for operating personnel and the general public.

b. The 10 CFR 50 limits for the control room are met for a design-basis accident (DBA) and lesser accidents.

c. Radiation protection design objectives related to 10 CFR 100 for accidents analyzed using TID-14844 and 10 CFR 50.67 for AST are given in Chapter 15.0

12.1.2.3 Improvements in Facility Design Due to Past Experience and Operation

At the time of the design and construction of Byron and Braidwood, Commonwealth Edison operated five licensed BWRs and two PWRs (see Chapter 1.0). The operating experience obtained from these stations has been incorporated into the design of Byron/ Braidwood Stations. In addition, published information on radiation problems and radiation protection (in nuclear power stations) was used to anticipate and minimize occupational radiation exposure. Experienced operating personnel continually reviewed the station design as the design progressed, and provided recommendations based on their experience.

Routine survey data from Commonwealth Edison's operating stations has been used to correct or improve the design of Byron/ Braidwood Stations. Some design improvements directly attributed to experiences and operations are as follows; others are discussed in Section 12.3.

a. An adequate number of equipment decontamination areas have been included to reduce congestion and reduce maintenance time.

b. Concrete shield walls, floors, and ceiling are coated with a nonporous coating to enhance decontamination wherever a potential for leakage or spillage of radioactive material exists on these surfaces.

c. To the extent practicable, all valves servicing radioactive or potentially radioactive equipment are centrally located in shielded valve aisles apart from the equipment serviced; walk-in valve aisles are used. Where practicable, no valves are located in pipe tunnels.

d. All radioactive or potentially radioactive manually operated valves and associated piping are shielded from the valve operating area when practicable. Remote manual valve operators connected to manually operated or geared handwheels extending through the shielding to the valve operating area are used (see Figure 12.3-3). Valve operating personnel are thus protected from radiation due to radioactivity in the valves and associated fluid piping in the valve aisle.

e. To reduce the amount of radioactive material in valve aisles, radioactive pipe runs to and from valves aisles are minimized by maximizing the amount of radioactive runs behind the shield wall placed between the piece of equipment and the valves.

f. Motor and pneumatic operated valves (generally higher maintenance items than manually operated valves) which are in radioactive or potentially radioactive service, are located in areas shielded from the component serviced by the valve. This minimizes personnel exposure during valve maintenance and inspection.

g. Valves servicing radioactive or potentially radioactive equipment are installed and positioned relative to other valves so as to minimize maintenance time. Space is provided around valves so that compensatory shielding (such as lead blankets) can be used as needed.

h. Components associated with control of the instrument air supply to air operated valves, are not themselves radioactive or potentially radioactive, and are located in low radiation areas.

i. Controls are installed in the lowest practicable radiation zone; use of transducers is maximized in high radiation areas. Instrument readouts are located in areas which will result in the lowest personnel exposure, if consistent with other requirements such as instrument accuracy and precision.

j. Instrument readouts are designed and located to minimize the time and exposure necessary to take a reading. They are positioned in readily accessible, adequately lighted areas, at a convenient elevation for observation and parallax correction. They must face in a direction convenient for reading, have easily readable numbers and pointers; the application of scale multipliers is minimized.

k. Shielding separates pumps from their associated tanks or other vessels.

l. Space and adequate floor strength for temporary shielding is supplied where practicable.

More examples of how Commonwealth Edison's experience has contributed to the Byron/ Braidwood Stations design can be found in Section 12.3.

12.1.2.4 Equipment Design Considerations

Radiation protection design consideration of equipment involves shielding, equipment access, equipment selection, and equipment maintenance. Equipment design objectives deal with access to, and segregation of, radioactive equipment. The following are the equipment design objectives for radiation protection:

a. Equipment which processes fluids with low radioactivity are located in separate cubicles from equipment which processes highly radioactive fluids.

b. Galleries, hatches, and gratings are provided as needed to allow access to equipment from the top, especially if the piece of equipment is high above a floor.

c. Equipment is located in accessible parts of cubicles. Equipment frequently changed in whole or in part is readily accessible.

d. Cranes or lifting lugs are provided as needed for equipment servicing, maintenance, and removal.

e. Localized shielding or space and adequate structure for localized shielding is provided as part of the shielding design.

f. Unmortared removable block walls or easily removable floor or wall plugs are provided to minimize the radiation exposure in gaining access to highly radioactive components when removal (e.g., tube pulling) is required.

12.1.2.5 Equipment Selection

The selection of equipment to handle and process radioactive materials is based upon system requirements and radiation protection requirements. Consideration is given to minimizing leakage, spillage, and maintenance requirements. Material and coating selection are chosen for decontamination properties as well as durability. Some components which may become contaminated are designed with provisions for flushing or cleaning. Reduced occupational radiation exposure is attained by utilizing operating experience and where practical, providing prudent equipment selections such as:

a. plug valves which require less maintenance in place of diaphragm valves;

b. diaphragm seal valves which require no packing;

c. longer-life graphite-filled packing, instead of standard packing;

d. fluid connections for the capability to back flush;

e. remote systems (or connections) for remote chemical cleaning where practicable;

f. air connections to tanks containing spargers to allow for air injection to uncake contaminates;

g. cross-ties between redundant equipment and/or related equipment capable of redundant operation to allow removal of contaminated equipment from service;

h. pumps with flanged connections to allow quick removal and installation;

i. mechanical seal flushing lines on pumps to reduce the accumulation of radioactive material in the seals;

j. remote filter handling equipment for radwaste disposal; and

k. drains on tanks flush with inside surface of the tanks.

12.1.2.6 Overall Impact of Design Considerations

Special attention has been given, as noted above and elsewhere in this chapter, to maintaining occupational radiation dose ALARA - while establishing the final design of Byron/Braidwood Stations. The design of facilities, equipment, structures, and access areas consider exposure obtained during routine operations (sampling, surveys, inspections, etc.), transient operations (changing power levels, startup, and shutdown), operational occurrences (identification, removal from service, etc.), maintenance, moving and storing radioactive materials, and accidents. These designs take into account equipment removal, decontamination, ventilation, orientation of equipment, in situ calibration and maintenance, sampling, monitoring, shielding, controlling contaminated fluids, minimizing leakage and spillage, and radiation exposure.

The station staff includes health physicists as described in Subsection 13.1.1.3. Experience in radiation protection has been incorporated into the design of Byron/Braidwood during review and comment stages. In addition, design reviews have been conducted by other competent health physicists.

The design philosophy established for Byron/Braidwood strives to maintain occupational radiation exposure ALARA and is in compliance with applicable regulations.

12.1.2.7 Radiation Protection Design Review

a. Reviewers of the Radiation Protection Design

The station owner has the responsibility for the radiation protection design review on the Byron and Braidwood Stations. Commonwealth Edison utilized Westinghouse and Sargent & Lundy to review the Byron/Braidwood Stations' radiation protection design.

Westinghouse employs system analysis engineers, competent in the area of health physics and radiation protection, to work with system design engineers. Although many groups within the Westinghouse Systems Division (SD) are available when required, the two major sections responsible for radiation protection review are Plant and Systems Evaluation Licensing, within the Nuclear Safety Department, and Radiation and Systems Analysis within the Engineering Department. The managers of these two sections report through the management of their respective departments to the SD General Manager, who is responsible for the overall design of RESAR-414 plants.

The A-E, Sargent & Lundy, performs ALARA Radiation Protection Design Reviews at key points in the balance of plant design. These reviews are independent of the owner's reviews and incorporate the instructions of the owner. The radiation protection design reviews conducted by Sargent & Lundy, cover access control, radiation shielding, radiation monitoring, radiation protection facilities, and control of airborne contamination in accordance with the ALARA concepts in Sections C.2 and C.4 of Regulatory Guide 8.8. The Sargent & Lundy ALARA review is conducted according to written procedures which establish a review committee and a committee chairperson. The chairperson is an experienced radiation protection specialist and is responsible for the design review; he assigns committee members and additional reviewers as necessary to review tasks in their area of expertise. The review committee issues a report summarizing its review and its conclusions. A summary of the qualifications of the personnel who participated in the most recent Sargent & Lundy ALARA Radiation Protection Design Review are given in Table 12.1-2. The review team consisted of the committee chairperson, at least three committee members and two additional reviewers.

Types of personnel that have been involved in the radiation protection review are given in Tables 12.1-1 and 12.1-2.

b. <u>Recordkeeping of the Radiation Protection Design Review Process</u>

Design information is logged and sent to the owner for comments. Portions of the design information involve radiation shielding, monitoring, laboratory facilities and other radiation considerations. These items are directed to the responsible radiation protection reviewer. Comments are sent through both project manager's divisions (owner and designer). Radiation protection comments and requested changes are forwarded to the engineer responsible for the radiation protection (RP) design. The RP designer responds to the comments and requests. He then files the comments, requests, and the response. The RP designer makes the required design changes. The project management divisions coordinate and document the changes.

The personnel with expertise in radiation protection within the groups stated above participate in the design review process in a systematic manner. The procedures to assure radiation protection functions needed to prevent or mitigate consequences of postulated accidents that could cause undue risk to the health and safety of the public are formally documented.

The NRC has reviewed the Westinghouse policy, design, and operational considerations related to assuring that occupational radiation exposures are ALARA for the RESAR-3S and RESAR-414 designs. They have concluded that Westinghouse has shown sufficient concern and familiarity with the ALARA principles in the areas of design considerations such that this aspect of radiation protection is acceptable. There are no substantial differences between RESAR-414, RESAR-3S and the Byron/Braidwood design in those areas that affect ALARA.

c. Radiation Protection Design Techniques to Reduce Person-Rem Exposure

1. The utilization of removable unmortared block wall sections (instead of mortared sections) for some equipment significantly reduces the number of person-hours spent in radiation areas.

2. Probe holes were placed in most removable hatches of filter and demineralizer cubicles. These holes allow radiation monitoring of the cubicles prior to removing its hatch. The radiation data from the monitor allows radiation protection personnel better control of occupational exposure.

3. Area radiation monitors (ARMs) were placed in valve aisles which serve two or more highly radioactive systems. These ARMs provide a warning on high-radiation and help to prevent high levels of unexpected exposure from the startup of an inactive system while performing maintenance on another system.

12.1.3 Operational Considerations

Operational radiation protection objectives deal with access to radiation areas, exposure to personnel, and decontamination. Working at or near highly radioactive components requires planning, special methods, and criteria directed toward keeping occupational radiation exposure ALARA. Job training and briefing for selected high exposure jobs contribute toward reduced exposures. Decontamination also helps to reduce exposure. Procedures and techniques are based upon operational criteria and experience. Procedures are discussed in Section 13.5.

12.1.3.1 Operational Objectives

The operational radiation protection objectives include the following:

a. knowledge of station design;

b. experienced personnel to direct and train other personnel;

c. detailed job planning and pre-job meetings for high exposure work;

d. job simulations to improve productivity on the job, thereby keeping exposure ALARA;

e. briefings after selected high exposure jobs to identify time consuming work and to identify problems;

f. improving procedures and techniques (defined in the following subsection) for future jobs;

g. use of radiation monitoring equipment to detect airborne radioactivity concentrations and high radiation levels and to measure and record personnel radiation exposure;

h. analysis of radioactive samples to monitor chemistry, check for radiation release, etc.;

i. use of cleaning and decontamination facilities for equipment and protective clothing; and

j. use of periodic radiation surveys are required.

12.1.3.2 Implementation of Procedures and Techniques

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

a. Section 12.1, "Ensuring That Occupational Radiation Exposures Are ALARA;"

b. Section 12.3, "Radiation Protection Design Features;" and

c. Section 12.5, "Health Physics Program."

12.1.3.3 Implementation of Exposure Tracking and Exposure Reduction Program

The Exelon Generation Company commitment to the ALARA principle is discussed in Subsection 12.1.1. The use of radiation work permits is discussed in Subsection 12.1.1.3.

Self-reading dosimeters are used at Byron/Braidwood stations to record estimates of daily exposures received by each individual worker. This information enables the Radiation Protection Department to spot significant individual exposures prior to processing other monitoring dosimetry. Work group person-rem summaries are generated by a computerized dose tracking program. The summaries serve to alert the station health physics staff and the corporate office of the trends in person-rem expenditures. Commonwealth Edison began a Radiation Evaluation Program (REP) in April of 1976. REP is a computer based occupational dose accounting system used to document, by work group, the dose expenditure resulting from work performed on various plant systems and components. In addition to each work group's dose and the plant component worked on, the program documents the total work effort in person-hours and include a brief description of the work performed.

The REP program applications are:

a. To provide timely radiological feedback information to the engineering and production departments and architect-engineer consultants for consideration in new plant design and to enable corrective action to be taken at existing stations.

b. To identify and compile dose histories on specific sources of occupational dose that might be reduced through improved station working and shielding procedures and training programs.

c. To provide data for comparison studies of specific sources of occupational exposure among similar Exelon Generation Company nuclear stations with relevant factors such as reactor equipment and plant layout, etc., taken into account.

d. To demonstrate an "active ALARA program."

The station has an ALARA Review Committee. This committee is composed of the manager of each affected department, the station manager, and Radiation Protection Department personnel. The charter of the committee is to advise the station manager on ALARA matters. The committee reviews annual exposure reduction goals as a part of its activities. The committee meets periodically as stated in subsection 12.1.1.1.

TABLE 12.1-1

NSSS RADIATION PROTECTION PERSONNEL

JOB TITLES	RADIATION PROTECTION REVIEW RESPONSIBILITIES	EDUCATION	EXPERIENCE
Manager of Energy and Environmental Analysis	Interfaces between the Engineering Department and the NRC. He reviews, coordinates, and supplies input for Chapters 1, 2, 11, 12, and 15 of the Safety Analysis Reports.	BS or higher in engineering or the physical sciences	5 years as a lead engineer or manager. Background in nuclear and chemical environmental engineering
Manager of Radiation and System Analysis	Provides radiation protection guidance. Analyzes plant radiation sources and exposure from and to components. Occupational radiation exposure design review.	MS or equivalent in mechanical, nuclear, or chemical engineering	6 years experience in nuclear plant system operation or design

TABLE 12.1-2

RADIATION PROTECTION PERSONNEL PARTICIPATING IN THE

AEs SUBMITTAL RADIATION PROTECTION DESIGN REVIEW

JOB TITLE	RESPONSIBILITIES	EDUCATION OF SPECIFIC REVIEWERS	EXPERIENCE OF* SPECIFIC REVIEWERS
Chairperson NSLD Radiation Protection Design Review Committee	Coordinate review by the committee Assign reviewers Assign review tasks Resolve disputes Approve committee conclusions Terminate review	Chairperson: B.S.E.E. Certified Health Physicist Registered Professional Engineer	Over 25 years experience in the nuclear industry and with the AEC.
Committee Members	Assigned a specific area of responsibility Summarize review responses Make recommendations and appraisals of plant's RP design	Members:	
		Ph.D., NE	Over 7 years in nuclear engineering and radiation engineering
		Ph.D., Health Physics	One year in health physics
		MS, NE Registered Professional Engineer	Over 13 years in nuclear engineering, radiation engineering, and health physics

*Experience at time of design review.

TABLE 12.1-2 (Cont'd)

JOB TITLE	RESPONSIBILITIES	EDUCATION OF SPECIFIC REVIEWERS	EXPERIENCE OF* SPECIFIC REVIEWERS
Reviewers (In addition to committee members)	Assigned a specific area of responsibility Review completeness of station's radiation protection Design. Identify deficiencies Make recommendations	<u>Reviewers</u>: Ph.D., NE MS, NE	 4 years in nuclear engineering and radiation engineering 3 years in nuclear engineering and health physics

12.2 RADIATION SOURCES

12.2.1 Contained Sources

The sources given in the subsection are based on the following parameters:

a. power = 3565 MW;

b. operation with defects in cladding or rods generating 1% of the core rated power;

c. reactor coolant mass = 2.42 x 10^8 grams; and

d. reactor coolant purification rate = 75 gpm at 130°F.

The design-basis shielding sources are more detailed and conservative than the realistic sources presented in Subsection 11.1.2. The conservative (design-basis) source model is described in Subsection 11.1.2.1.

The impact of a core power uprate on the design-basis shielding radiation source terms discussed above is provided in Section 12.2.4. The original licensed power level was 3411 MWt. The original source term and shielding analyses were performed at a power level of 3565 MWt. Byron and Braidwood Nuclear stations have uprated the core power level twice. First to a core power level of 3586.6 MWt, then to the Measurement Uncertainty Recapture uprate power level of 3645 MWt.

12.2.1.1 NSSS Sources

Reactor Coolant

Concentrations of activation products in the reactor water are given in Table 12.2-1 for Nitrogen-16 and in Table 12.2-2 for the other activation products and fission products.

Steam Generator

The activities on the primary side surfaces of the steam generator are used in determining access limitations in and around the steam generators at plant shutdown. Nominal values of deposited activity are listed in Table 12.2-3 for several operating times.

Pressurizer

The radioactive sources in the pressurizer, steam, and liquid phases, as well as the deposited sources are tabulated in Tables 12.2-4, 12.2-5, and 12.2-6 respectively.

Letdown Coolant Fission and Corrosion Products

The spectral source strengths in the purification letdown flow are tabulated in Table 12.2-7. The sources assume sufficient delay time from the reactor coolant loop for decay of the N-16 isotope and a fluid temperature of approximately 130°F (i.e., downstream of the letdown heat exchanger).

Volume Control Tank

The sources in the volume control tank are itemized in Table 12.2-8. These sources correspond to a nominal operating level in the tank of 160 ft^3 in the liquid phase and 240 ft^3 in the vapor phase.

Recycle Holdup Tank

The radiation sources in the recycle holdup tank exist in both the vapor and liquid phases. The vapor sources are used to determine the holdup tank shielding requirements, whereas the liquid sources are the basis for maximum evaporator activities.

For the vapor space inventory, it is assumed that all gases in the water entering the tank flashes in the vapor space. The Kr-88 isotope is the major isotope in terms of shielding requirements. The vapor sources are based on the time when the Kr-88 inventory is a maximum in the tank. The vapor sources are listed in Table 12.2-9.

The liquid phase activities are based on the assumption that all the gases remain in solution. The basis for this assumption is that the holdup tank liquid serves as feedwater to the recycle evaporator. The liquid phase sources are listed in Table 12.2-10.

Recycle Evaporator

The sources associated with the recycle evaporator package are specified in Table 12.2-11. In the package, gaseous activity is concentrated in the vent condenser portion while particulate and dissolved activity is concentrated in the evaporator section.

Residual Heat Removal Loop

The maximum specific source strengths in the residual heat removal loop are tabulated in Table 12.2-12. The residual heat removal loop is placed in operation approximately 4 hours after reactor shutdown and reduces the reactor coolant temperature to approximately 120°F within about 20 hours after shutdown. The sources are maximum values with credit taken for 4 hours of activity decay and purification.

Ion Exchangers, Chemical Volume Control System (CVCS)

The radiation sources in the ion exchangers of the CVCS are tabulated in Tables 12.2-13 through 12.2-16. The mixed bed retains the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation bed can be used intermittently to remove lithium for pH control, and supplements the mixed bed in removing Y, Cs, Mo, and the crud metals.

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

Recycle Evaporator Condensate Demineralizer

Sources for the recycle evaporator condensate demineralizer are given in Table 12.2-17.

Filters

Sources for the reactor coolant filter, seal water return filter, recycle evaporator feed filter, spent fuel pit filter, spent fuel pit skimmer filter, seal water injection filter, recycle evaporator concentrates filter, and recycle evaporator condensate filter, are given in Tables 12.2-19 through 12.2-22.

Reactor Core

The core gamma sources (after shutdown) are used to establish radiation shielding requirements during refueling operations and during shipment of spent fuel. The sources associated with the spent fuel are based on an average power assembly with an irradiation time of 10^8 seconds (3.1 years). These source strengths per unit volume of homogenized core are tabulated in Table 12.2-23 for various times after shutdown.

Irradiated Control Rods

The irradiated control rod sources are used in establishing radiation shielding requirements during refueling operations and during shipping of irradiated rods. The absorber material used in the control rods is silver-indium-cadmium (Ag-In-Cd) or hafnium.

The source strengths associated with the control rods are listed in Tables 12.2-24 and 12.2-24a for various times after shutdown. The values are per cm of height of a single rod for an irradiation period of 100,000 hours.

Refueling Water

Prior to refueling, the radioactivity in the reactor coolant is reduced by operating the purification system at the maximum letdown rate. Particulate (soluble) activity such as cesium, iodine, and the metals are removed by the mixed bed and cation bed demineralizers. Radioactive gases are removed by the volume control tank.

Sources for the spent fuel pit demineralizer are given in Table 12.2-18.

Based on a direct exposure dose rate of 2.5 mrem/hr at the surface of the refueling water, the accumulative isotopic concentration in the refueling water (after dilution) should not exceed one maximum permissible concentration (< MPC), as defined by 10 CFR 20, Appendix B, Table II, column 2 (the term MPC as used in this section refers to a 10CFR20 limit in effect prior to January 1, 1994) and Equation 12.2-1. The dilution consists of the complete mixing of the reactor coolant in the vessel with stored refueling water, approximately 1 to 10, respectively. The resulting mixture must satisfy the following relationship:

$$\frac{C_A}{MPC_A} + \frac{C_B}{MPC_B} + \frac{C_C}{MPC_C} + \frac{D_D}{MPC_D} + \ldots \leq 1 \qquad (12.2\text{-}1)$$

where the subscript identifies the isotope, C is the concentration in μCi/cc and MPC is one maximum permissible concentration in μCi/cc.

Table 12.2-25 gives the maximum allowable concentration (same as MPC) for some dominant isotopes.

The refueling pool purification system is designed to maintain the relationship given in Equation 12.2-1.

Irradiated Incore Detectors

Table 12.2-26 shows the incore fission chamber sources energy spectrum after three months of irradiation and one day decay.

The incore detector drive wire sources are used in establishing the radiation shielding requirements for the wires when the detectors are not in use and during shipment when the detectors have failed.

Table 12.2-27 lists the detector drive wire sources per cm of length of wire, assuming that the detector has been placed in the core for 1 year.

Process Piping

The radiation sources in process piping are derived from the activity in the process fluid plus an estimate of crud buildup. The concentrations in the process fluids are given in the following:

> Tables 12.2-1, 12.2-2, 12.2-4, 12.2-5, 12.2-8 through 12.2-18, 12.2-25, 12.2-29, 12.2-30, 12.2-31, and 12.2-33. The crud buildup levels are estimated using data from operating stations (References 3 and 4).

Gas Decay Tanks

The gas decay tank inventory used in the Chapter 15.0 accident analysis reaches a maximum while degassing the reactor coolant system during a cold shutdown. The gases removed by the volume control tanks are vented to the waste gas system. The maximum activity for the shielding design of a single gas decay tank is shown in Table 12.2-28, and represents the inventory present at the time Kr-85 is at a maximum. The shield thickness of the gas decay tank cubicle was determined using the maximum activity in the two tanks.

Spent Fuel Pit

The sources for the spent fuel pit fuel handling accident are given in Table 12.2-29.

12.2.1.2 Balance of Plant Shielding Design-Basis Sources

Shielding source terms are supplied here for components which contain liquid from processed reactor coolant and which are considered part of the radioactive waste management system.

Sources are presented in Tables 12.2-30 and 12.2-31 for the station drains and steam generator blowdown which are input sources to the liquid radwaste processing system. In addition, sources are presented for the spent resin storage tank in Table 12.2-43. The primary assumptions used in the calculation of source terms are presented in the following sections.

12.2.1.2.1 Blowdown Sources

Blowdown sources are based on 1 gpm primary-to-secondary tube leakage for one steam generator for 14 days. The total blowdown rate for four steam generators during reactor coolant leakage is approximately 135 gpm (see Subsection 10.4.8), so input into the blowdown stream is 1/135 times primary coolant activity.

12.2.1.2.2 Radwaste Processing Sources

Liquid radwaste processing sources are based on the radioactive sources contained in the drain tanks shown in Tables 12.2-30 and 12.2-31.

Turbine building floor drain tanks are based on main steam condensate activity which consists of blowdown sources (1/135 times primary coolant activity) multiplied by the steam generator partition factor (10^{-2} for iodines and 10^{-3} for noniodines), divided by 2 since the turbine building services two units.

The turbine building drains source bases along with the chemical drain tank, chemical/regeneration waste drain tank, auxiliary building floor and equipment drain tanks, and laundry drain tank source bases are shown on Table 12.2-32 as fractions of primary coolant obtained from Reference 1. Laundry drain tank sources are further adjusted by incorporating laundry drain sources measured at the Zion and Quad-Cities Stations. These sources are shown on Table 12.2-33.

The decontamination factors (DFs) used for filters, demineralizers, and evaporators are given in Table 12.2-34 for the elements found in the sources. Data is from References 1 and 2. The sources for the components of the radwaste processing system shown on Drawing M-48a are given in Tables 12.2-35 through 12.2-41. Included are shielding design-basis source terms for the blowdown mixed bed demineralizer, radwaste mixed bed demineralizer, concentrates holding tank, blowdown prefilter and afterfilter, radwaste afterfilter, turbine building equipment drain filter, turbine building floor drain filter, auxiliary building equipment drain filter, regeneration waste drain filter, chemical drain filter, laundry drain filter, radwaste evaporator, 30,000 gallon release tank, laundry drain tank, blowdown monitor tank, radwaste evaporator monitor tank, and the auxiliary building floor drain filter. Source terms for the steam generator blowdown prefilters were calculated using the volume of equipment 1/2WX02MA,B (housing-only prefilter vessels).

The design-basis operating time for radwaste processing equipment was taken as 30 days which is sufficient time for most radionuclides of concern to build up to equilibrium.

12.2.1.2.3 Spent Resin Storage Tank

The spent resin storage tank inventory assumes that the demineralizers with the highest potential activities will send their resins to the tank. The demineralizers considered and the assumed fractional contribution of each is shown in Table 12.2-42. The demineralizers are assumed to operate for one core cycle at full power and 1% failed fuel for both units. Demineralizer sources are given in Subsection 12.2.1.1. The spent resin storage tank radionuclide inventory is given in Table 12.2-43.

12.2.1.2.4 Radwaste Solidification System

The shielding source terms for the decanting and drumming equipment storage area are based on the drumming of wastes from the spent resin storage tank shown in Table 12.2-43, column 2, with no decay time assumed. Spent resins are the highest activity sources to be handled by the radwaste solidification system and result in a conservative shielding design basis.

The composition of a design-basis radwaste drum used in the shielding of the radwaste drum storage areas is shown in Table 12.2-44. The sources used for intermediate activity drum storage area shielding are spent resins decayed for 90 days, which assumes that resins are stored for this average time period in the spent resin storage tank prior to drumming. These sources result in a dose rate of 80 R/hr for a single drum at contact. Spent resin storage tank sources decayed for 90 days are given in Table 12.2-43.

The sources used for the low activity drum storage area shielding are radwaste evaporator concentrates shown in Table 12.2-39.

The decontamination factors (DFs) used for filters, demineralizers, and evaporators are given in Table 12.2-34 for the elements found in the sources. Data is from References 1 and 2. The sources for the components of the radwaste processing system shown on Drawing M-48A are given in Tables 12.2-35 through 12.2-41. Included are shielding design-basis source terms for the blowdown mixed bed demineralizer, radwaste mixed bed demineralizer, low activity spent resin storage tank holding tank, blowdown prefilter and afterfilter, radwaste afterfilter, turbine building equipment drain filter, turbine building floor drain filter, auxiliary building equipment drain filter, regeneration waste drain filter, chemical drain filter, laundry drain filter, radwaste evaporator, 30,000 gallon release tank, laundry drain tank, blowdown monitor tank, radwaste evaporator monitor tank, and the auxiliary building floor drain filter. Source terms for the steam generator blowdown prefilters were calculated using the volume of equipment 1/2WX02MA,B (housing-only prefilter vessels).

The design-basis operating time for radwaste processing equipment was taken as 30 days which is sufficient time for most radionuclides of concern to build up to equilibrium.

12.2.1.2.3 Spent Resin Storage Tank

The high activity spent resin storage tank inventory assumes that the demineralizers with the highest potential activities will send their resins to the tank. The demineralizers considered and the assumed fractional contribution of each is shown in Table 12.2-42. The demineralizers are assumed to operate for one core cycle at full power and 1% failed fuel for both units. Demineralizer sources are given in Subsection 12.2.1.1. The high activity spent resin storage tank radionuclide inventory is given in Table 12.2-43.

It is expected that only the resin from blowdown and radwaste mixed bed demineralizers will be sent to the low activity spent resin tank (formerly the concentrates holding tank).

The total amount of resin (Anion and Cation) per mixed bed for blowdown and radwaste systems is 113 ft^3 and 29 ft^3 respectively.

The demineralizers considered and the assumed fractional contribution of each is shown in Table 12.2-53.

The low activity spent resin tank expected radionuclide inventory is given in Table 12.2-54.

12.2.1.2.4 Radwaste Solidification System

The shielding source terms for the decanting and drumming equipment storage area are based on the drumming of wastes from the spent resin storage tank shown in Table 12.2-43, column 2, with no decay time assumed. Spent resins are the highest activity sources to be handled by the radwaste solidification system and result in a conservative shielding design basis.

The composition of a design-basis radwaste drum used in the shielding of the radwaste drum storage areas is shown in Table 12.2-44. The sources used for intermediate activity drum storage area shielding are spent resins decayed for 90 days, which assumes that resins are stored for this average time period in the spent resin storage tank prior to drumming. These sources result in a dose rate of 80 R/hr for a single drum at contact. Spent resin storage tank sources decayed for 90 days are given in Table 12.2-43.

The sources used for the low activity drum storage area shielding are radwaste evaporator concentrates shown in Table 12.2-39.

These sources result in a dose rate of 3 R/hr at contact. The models used for shielding design are given in Subsection 12.3.2.1.9.

12.2.1.2.5 Volume Reduction System

The volume reduction (VR) system is addressed in Section 11.4.

Source information and shielding design information has been intentionally deleted. Byron and Braidwood stations do not intend to use volume reduction system equipment.

12.2.1.3 Sources for HVAC Charcoal Filters

Sources for the main auxiliary building charcoal filters and the off-gas vent charcoal filters are given in Table 12.2-45. The Braidwood off-gas filter system has been modified such that all exhaust gases bypass the filter unit under all operating conditions. The following assumptions were made in the determination of the sources on the main auxiliary building charcoal filters.

a. Halogens become airborne only by evaporation from leaks of radioactive steam or water to the interior of the auxiliary building.

b. The steam or water contains reactor water sources.

c. The total leak rate is 20 gal/day for cold leakage and 1 gal/day for hot leakage. Partition factors are 0.001 and 0.1 for cold and hot leakage respectively.

d. Instantaneous complete mixing of the iodine throughout the volume of the auxiliary building occurs.

e. The exhaust rate through the filters is 20,000 ft^3/min. Holdup time in the filters is 0.25 seconds.

f. There are six main auxiliary building charcoal filters with efficiencies of 95%.

The following assumptions were made in the determination of the sources on the off-gas charcoal filters. (The Braidwood off-gas filter system has been modified such that all exhaust gases bypass the filter unit under all operating conditions.)

a. There is a primary to secondary leakage in one steam generator of 1 gpm with 1% failed fuel.

b. All four steam generators blow down at once. The blowdown rate for the one leaking steam generator is 90 gpm, while the blowdown rate for each of the three nonleaking steam generators is 15 gpm.

c. Partition factors are 0.01 for the steam generators, 5×10^{-4} for the main condenser, and 0.05 for the blowdown tank and condenser vents.

d. The main steam flow rate is 1.51 x 10^7 lb/hr.

e. The filter efficiency is 95%. Since the filters are in series, in the off-gas filter train, they may be considered together as one filter.

12.2.1.4 Old Steam Generator Storage Facility

The old steam generator storage facility (OSGSF) is a reinforced concrete building that provides long-term storage and shielding for the four, old Unit 1 steam generators. The facility is located in the owner-controlled area outside of the security perimeter fence (see Figure 2.1-7 for Byron and Figure 2.1-4 for Braidwood). Shielding analysis for the OSGSF used measured dose rates obtained at each generator region in conjunction with waste samples to identify the dominant gamma-emitting isotopes (see Table 12.2-55).

12.2.2 Airborne Radioactive Material Sources

With the exception of noble gases, sources of airborne radioactivity are generated from radioactive liquid sources by the mechanisms discussed in the following subsection. The generation of airborne radioactivity in radiation areas can affect the areas normally accessible to operating personnel, mainly pump and valve areas. The airborne radioactivity during normal operation for accessible areas is discussed in Subsection 12.2.2.3 The calculational model is given in Subsection 12.2.2.4.

In addition to affecting the station, the airborne radioactive material which exits in the filtration systems, enters the environment via the station vent stack.

12.2.2.1 Production of Radioactive Airborne Material

Radioactive materials become airborne through evaporation and by being attached to suspended water droplets and water vapor. The water vapor comes from leaks in high energy lines (pressurized hot water). Suspended water droplets are created by sprays (usually leaks) and splashing. Evaporation occurs wherever there is standing water. Some examples are:

Component	Airborne Method
fuel pool	evaporation
radwaste	evaporation (venting)
high energy line leak	vapor, evaporation

Component	Airborne Method
spray from high energy	vapor, droplets evaporation
low energy line break	evaporation
spill	droplets, evaporation

The major contributors to airborne radioactivity during normal operation are: (1) leaks in the chemical volume control system, (2) evaporation from fuel pool, (3) leaks in radwaste systems, (4) venting of radwaste tanks, and (5) leaks in the charcoal-HEPA exhaust systems. Minor contributions are: (1) cleaning and decontaminating tools and equipment, (2) contaminated wearing apparel, and (3) sample preparation and analysis.

Some abnormal occurrences can cause airborne radioactivity; they are: (1) spills (i.e., overflows and splashing), (2) failure of a ventilation system, (3) cracks in piping, (4) failures of pump and valve seals, and (5) malfunctioning equipment.

12.2.2.2 Sources in Areas Normally Accessible to Operating Personnel

Airborne radioactive material is expected to affect general access areas only during a ventilating system failure, or spillage of radioactive material in areas which are not sealed from general access areas. Airborne radioactive material is expected during refueling in maintenance areas, in labs (occasionally), and the hot instrument room.

The ventilation flow path is from areas of potentially low airborne radioactivity to ones of potentially high airborne radioactivity. The ventilation system has been designed to control the airborne radioactivity in the laboratories, maintenance areas, and the refueling floor of the reactor building. The concentration of airborne radioactivity is periodically determined by the radiation protection staff. The most significant radioactive isotopes are the halogens (primarily iodine). The iodines have the highest concentration in relation to the maximum permissible concentrations.

Maintenance accounts for a sizeable portion of the internal exposure of personnel because station personnel have to perform many of these functions in areas with relatively high airborne radioactivity. The airborne radioactivity is caused by leaks, spills, venting, etc. The airborne concentrations are calculated for the occurrences that are the most common, namely, leaks and venting. These concentrations are given in the next subsection. Infrequent anticipated operational occurrences and abnormal occurrences are handled in the manner established in the personnel internal exposure program (Subsection 12.5.3.3).

12.2.2.3 Calculated Concentrations During Operation

The calculated concentrations of airborne radioactive iodine in normally accessible cubicles are based upon the model given in Subsection 12.2.2.4. These concentrations are given in Tables 12.2-46, 12.2-47, and 12.2-48. The general access areas have very little if any airborne contaminants (i.e., $<10^{-12}$ μCi/cc above background) during normal operation except for those mentioned in Subsection 12.2.2.2. Concentrations in normally accessible areas are determined by periodic air sampling as specified in the health physics program.

12.2.2.4 Models and Parameters Used in Calculations of Airborne Radioactivity Concentration

For cubicles with non-radioactive supply air, the equation used to calculate the equilibrium airborne radioactivity concentrations during normal operation is as follows:

$$C_A = [(L\ C_\ell\ P)/(7.48(\lambda V + F))] \qquad (12.2\text{-}2)$$

where:

C_A = airborne concentration in each cubicle (μCi/cc)
L = leak rate (gpm)
C_ℓ = liquid concentration (μCi/cc)
P = fraction of activity released to air
7.48 = is conversion factor (7.48 gal/ft^3)
λ = decay constant (min^{-1})
V = enclosed volume (ft^3)
F = air exhaust flow rate (ft^3/min).

12.2.2.5 Stack Effluents

Ventilation system exhausts which are potentially radioactive are routed to the station vent stack. Each ventilation system is designed to exhaust into the station vent stack simultaneously with all other ventilation systems. Ventilation systems, containing potentially high airborne radioactivity concentrations are provided with filters specifically designed to hold-up or remove radioactive material (Section 9.4).

The dominating radioisotopes released through the station vent stack are the noble gases from the off-gas system and the vent filter system. The expected yearly releases during normal operation are discussed in Section 11.3.

12.2.3 Changes to Source Data Since PSAR

Airborne radioactive material sources were not specified in the Byron/Braidwood PSAR. The entire Subsection 12.2.2 has been added. Sources for reactor coolant have been updated by Westinghouse. However, there have been no changes necessary in shielding requirements as a result of these revisions.

12.2.4 Impact of Uprate on Radiation Source Terms

The original licensed power level was 3411 MWt. The original source terms for shielding analyses were performed at a power level of 3565 MWt. Byron and Braidwood Nuclear stations have uprated the core power level twice. First to a core power level of 3586.6 MWt, then to the Measurement Uncertainty Recapture uprate power level of 3645 MWt.

Core uprate will result in an approximate 0.6% increase in the inventory of core fission and activation products addressed in the original design basis source term/shielding analyses. The reactor coolant N-16 activity given in Table 12.2-1, the shutdown reactor core gamma sources presented in Table 12.2-23, the irradiated control rod sources in Tables 12.2-24 and 12.2-24a, the irradiated incore detector sources in Tables 12.2-26 and 12.2-27, and the spent fuel pit water activity for a fuel handling accident given in Table 12.2-29 will all increase by approximately 0.6% after core uprate. This small percentage increase is well within the uncertainty of the calculated design basis shielding source terms presented in this section (taking into consideration the accuracy of nuclear data, and the conservatism present in computation model simplification utilized in the original analyses). Consequently, these tables remain valid for uprate.

The deposited corrosion product activities on the primary side surfaces of the steam generators listed in Table 12.2-3 and on the pressurizer surface listed in Table 12.2-6 are based on the experience of large operating PWRs and are applicable for uprate. Radiation sources in filters given in Tables 12.2-19 through 12.2-22 are conservative maximum values based on operating experience and remain valid for uprate.

The original design basis RCS activity given in Table 12.2-2 (same as Table 11.1-2) is re-calculated at a core power level of 3658.3 MWt which is 2% above the uprated power level. A more realistic nominal coolant mass of 2.477E8 gms is utilized in lieu of the conservative low RCS water mass of 2.42E8 gms used in the original analyses. The uprated design-basis coolant activity provided in Table 11.1-13 is comparable to the original design-basis coolant activities presented in Table 12.2-2. Since the remaining radiation source term data presented in this section are derived from the design-basis coolant concentrations, they remain valid for uprate.

Power uprate has no significant impact on the plant design basis radiation source terms.

12.2.5 References

1. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ALAP for Radioactive Material in Light Water Power Reactor Effluents," WASH-1258, July 1973.

2. Radiation Analysis Design Manual, WCAP-7664, January 1973.

3. "Source Term Data for Westinghouse Pressurized Water Reactors," WCAP-8253, Pittsburgh, Pa., May 1974.

4. "Oconee Radiochemistry Survey Program," RDTPL75-4, Babcock & Wilcox, May 1975.

TABLE 12.2-1

REACTOR COOLANT NITROGEN-16 ACTIVITY

LOCATION IN PRIMARY COOLANT LOOP	TIME AFTER LEAVING THE CORE (sec)	ACTIVITY (μCi/g)
leaving core	0.0	136
leaving reactor vessel	1.3	113
entering steam generator	1.7	109
leaving steam generator	5.8	74
entering reactor coolant pump	6.5	69
entering reactor vessel	7.2	65
entering core	9.2	54

TABLE 12.2-2

REACTOR COOLANT SOURCES FOR SHIELDING DESIGN
(ORIGINAL DESIGN BASIS)

ISOTOPE	ACTIVITY (μCi/g)	ISOTOPE	ACTIVITY (μCi/g)
H-3	3.5 (maximum)		
Kr-85	8.8 (peak)	I-131	2.5
Kr-85m	2.1	I-132	2.8
Kr-87	1.2	I-133	4.0
Kr-88	3.7	I-134	0.6
Xe-131m	1.9	I-135	2.2
Xe-133	281.0	Te-132	0.3
Xe-133m	18.8	Te-134	2.9×10^{-2}
Xe-135	6.3	Cs-134	2.3
Xe-135m	0.4	Cs-136	2.8
Xe-138	0.7	Cs-137	1.5
		Cs-138	0.98
Br-84	4.3×10^{-2}	Ba-137m	1.4
Rb-88	3.7	Ba-140	4.3×10^{-3}
Rb-89	0.21	La-140	1.5×10^{-3}
Sr-89	3.3×10^{-3}	Ce-144	3.4×10^{-4}
Sr-90	1.7×10^{-4}	Pr-144	3.4×10^{-4}
Sr-91	1.9×10^{-3}		
Sr-92	7.4×10^{-4}	Mn-54	7.9×10^{-4}
Y-90	2.0×10^{-4}	Mn-56	3.0×10^{-2}
Y-91	6.1×10^{-3}	Co-58	2.6×10^{-2}
Y-92	7.2×10^{-4}	Co-60	1.0×10^{-3}
Zr-95	7.0×10^{-4}	Fe-59	1.1×10^{-3}
Nb-95	6.9×10^{-4}	Cr-51	9.5×10^{-4}
Mo-99	5.3		

This table is based on the following:

a. Reactor coolant mass = 2.42×10^{8} grams.

b. Operation with defects in cladding of rods generating 1% of the core rated power of 3565 MWt.

c. Reactor coolant purification rate = 75 gpm.

d. The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

e. See Table 11.1-13 for the design basis reactor coolant inventory based on the uprated power level.

TABLE 12.2-3

DEPOSITED CORROSION PRODUCT ACTIVITY ON STEAM GENERATOR PRIMARY SIDE SURFACES ($\mu Ci/cm^2$)

ISOTOPES	OPERATING TIME (months) 6	12	24	36
Mn-54	0.15	0.60	1.5	2.0
Mn-56	3.3	3.3	3.3	3.3
Co-58	4.5	10.2	11.0	11.0
Fe-59	1.4	3.0	3.0	3.0
Co-60	0.26	1.0	2.6	4.5

TABLE 12.2-4

PRESSURIZER LIQUID PHASE ACTIVITY

ISOTOPE	ACTIVITY (μCi/g)
N-16 (maximum)	1.3
Rb-88	1.1×10^{-2}
Mo-99	2.2
I-131	1.6
I-132	6.2×10^{-2}
I-133	7.0×10^{-1}
I-134	5.5×10^{-3}
I-135	1.4×10^{-1}
Cs-134	1.9
Cs-136	2.1×10^{-1}
Cs-137	1.3
Cs-138	5.5×10^{-3}
Ba-137m	1.2

TABLE 12.2-5

PRESSURIZER STEAM PHASE ACTIVITY

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)*
Kr-85	5.1×10^{1}
Kr-85m	1.0×10^{-1}
Kr-87	1.8×10^{-2}
Kr-88	1.2×10^{-1}
Xe-131m	4.7
Xe-133	3.6×10^{2}
Xe-133m	10.9
Xe-135	6.5×10^{-1}
Xe-135m	1.3×10^{-3}
Xe-138	2.2×10^{-3}

* at operating conditions.

TABLE 12.2-6

PRESSURIZER DEPOSITED ACTIVITY

ISOTOPE	ACTIVITY ($\mu Ci/cm^2$)
Cr-51	9.8×10^{-2}
Mn-54	1.5×10^{-1}
Mn-56	2.2×10^{-2}
Co-58	3.8
Co-60	2.1×10^{-1}
Fe-59	1.4×10^{-1}

TABLE 12.2-7

LETDOWN COOLANT ACTIVITY

GAMMA ENERGY (MeV/γ)	SPECIFIC SOURCE STRENGTH (MeV/cm^3-sec)
0.4	4.5×10^5
0.8	2.7×10^5
1.3	1.7×10^5
1.7	1.2×10^5
2.2	1.4×10^5
2.5	1.6×10^5
3.5	1.9×10^4

NOTE: Same isotopic composition as Table 12.2-2.

TABLE 12.2-8

VOLUME CONTROL TANK

VAPOR PHASE (240 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Kr-85	2.0	1.4×10^{1}
Kr-85m	9.1	6.2×10^{1}
Kr-87	3.4	2.3×10^{1}
Kr-88	1.62×10^{1}	1.1×10^{2}
Xe-131m	1.44×10^{1}	9.8×10^{1}
Xe-133	2.1×10^{3}	1.4×10^{4}
Xe-133m	1.4×10^{2}	9.5×10^{2}
Xe-135	3.8×10^{1}	2.6×10^{2}
Xe-135m	4.0×10^{-1}	2.7
Xe-138	7.0×10^{-1}	4.8

LIQUID PHASE (160 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/g$)	INVENTORY (Ci)
Kr-85	8.8 (peak)	4.0×10^{1}
Kr-85m	1.5	6.8
Kr-87	0.5	2.3
Kr-88	2.1	9.5
Xe-131m	1.9	8.6
Xe-133	2.76×10^{2}	1.2×10^{3}
Xe-133m	1.8×10^{1}	7.9×10^{1}
Xe-135	5.1	2.3×10^{1}
Rb-88	3.7×10^{-1}	1.7
Rb-89	2.1×10^{-2}	9.6×10^{-2}
Mo-99	5.3×10^{-1}	2.4
I-131	2.5×10^{-1}	1.1
I-132	2.8×10^{-1}	1.3
I-133	4.0×10^{-1}	1.8

TABLE 12.2-8 (Cont'd)

LIQUID PHASE (160 ft^3)

ISOTOPE	ACTIVITY (μCi/g)	INVENTORY (Ci)
I-134	5.6×10^{-2}	2.5×10^{-1}
I-135	2.2×10^{-1}	1.0
Cs-134	2.3×10^{-1}	1.1
Cs-136	2.8×10^{-1}	1.3
Cs-137	1.5×10^{-1}	6.8×10^{-1}
Cs-138	9.8×10^{-2}	4.4×10^{-1}
Ba-137m	1.4×10^{-1}	6.4×10^{-1}

TABLE 12.2-9

RECYCLE HOLDUP TANK

VAPOR PHASE SOURCES (8087 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Kr-85	5.0	1.1×10^3
Kr-85m	0.8	1.8×10^2
Kr-87	0.3	6.9×10^1
Kr-88	1.2	2.7×10^2
Xe-131m	1.0	2.3×10^2
Xe-133	1.4×10^2	3.2×10^4
Xe-133m	9.1	2.1×10^3
Xe-135	2.7	6.2×10^2
Xe-135m	2.4×10^{-2}	5.5
Xe-138	4.2×10^{-2}	9.6

TABLE 12.2-10

RECYCLE HOLDUP TANK

LIQUID PHASE SOURCES (6886 ft^3)

ISOTOPE	ACTIVITY (μCi/g)	INVENTORY (Ci)
H-3	3.5	6.8×10^{2}
Kr-85	8.8 (peak)	1.7×10^{3}
Kr-85m	2.1	4.1×10^{2}
Kr-87	1.2	2.3×10^{2}
Kr-88	3.7	7.2×10^{2}
Xe-131m	1.9	3.7×10^{2}
Xe-133	2.8×10^{2}	5.5×10^{4}
Xe-133m	1.9×10^{1}	3.6×10^{3}
Xe-135	6.3	1.2×10^{3}
Rb-88	3.7×10^{-2}	7.2
Rb-89	2.1×10^{-3}	4.0×10^{-1}
Mo-99	5.3×10^{-2}	1.0×10^{1}
I-131	2.5×10^{-2}	4.9
I-132	2.8×10^{-2}	5.6
I-133	4.0×10^{-2}	7.8
I-134	5.6×10^{-3}	1.1
I-135	2.2×10^{-2}	4.3
Cs-134	2.3×10^{-2}	4.4
Cs-136	2.8×10^{-2}	5.4
Cs-137	1.5×10^{-2}	2.9
Cs-138	9.8×10^{-3}	1.9
Ba-137m	1.4×10^{-2}	2.7

TABLE 12.2-11

RECYCLE EVAPORATOR

VENT CONDENSER SECTION (7900 cm^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Kr-85	293	2.3
Kr-85m	70	5.5×10^{-1}
Kr-87	41	3.2×10^{-1}
Kr-88	124	9.8×10^{-1}
Xe-131m	64	5.1×10^{-1}
Xe-133	9343	$7.4 \times 10^{+1}$
Xe-133m	630	5.0
Xe-135	209	1.7
Xe-135m	13	1.0×10^{-1}
Xe-138	23	1.8×10^{-1}

EVAPORATOR SECTION (CONCENTRATES) MASS = 2.08×10^6 grams

ISOTOPE	ACTIVITY ($\mu Ci/g$)	INVENTORY (Ci)
I-131	0.74	1.5
I-132	0.12	2.6×10^{-1}
I-133	0.87	1.8
I-134	0.01	2.1×10^{-2}
I-135	0.25	5.2×10^{-1}
Mo-99	1.48	3.1
Cs-134	0.69	1.5
Cs-137	0.47	9.8×10^{-1}
Ba-137m	0.44	9.5×10^{-1}

TABLE 12.2-12

RESIDUAL HEAT REMOVAL LOOP

RESIDUAL HEAT REMOVAL LOOP SOURCES

ISOTOPE	ACTIVITY (μCi/g)	ISOTOPE	ACTIVITY (μCi/g)
Mo-99	4.0	Kr-85	7.6
I-131	1.6	Kr-85m	0.96
I-132	0.56	Kr-87	0.12
I-133	2.3	Kr-88	1.6×10^{1}
I-135	1.0	Xe-133	2.4×10^{2}
Te-132	0.17	Xe-133m	2.6
Cs-134	1.8	Xe-135	4.0
Cs-136	2.2		
Cs-137	1.2		
Ba-137m	1.1		

TABLE 12.2-13

MIXED BED DEMINERALIZER (30 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Br-84	0.6	5.1×10^{-1}
Rb-88	28.9	2.5×10^{1}
Rb-89	1.3	1.1
Sr-89	1.04×10^{2}	8.8×10^{1}
Sr-90	15.2	1.3×10^{1}
Sr-91	0.5	4.2×10^{-1}
Sr-92	5.2×10^{-2}	4.4×10^{-2}
Y-90	7.5	6.4
Y-91	10.8	9.2
Y-92	0.12	1.0×10^{-1}
Zr-95	26.7	2.3×10^{1}
Nb-95	39.7	3.4×10^{1}
Mo-99	1.31×10^{3}	1.1×10^{3}
I-131	1.25×10^{4}	1.1×10^{4}
I-132	1.84×10^{3}	1.6×10^{3}
I-133	2.18×10^{3}	1.9×10^{3}
I-134	13.4	1.1×10^{1}
I-135	3.77×10^{2}	3.2×10^{2}
Te-132	5.4×10^{2}	4.6×10^{2}
Te-134	0.54	4.6×10^{-1}
Cs-134	1.01×10^{4}	8.4×10^{3}
Cs-136	7.4×10^{2}	6.2×10^{2}
Cs-137	6.6×10^{3}	5.6×10^{3}
Cs-138	12.9	1.1×10^{1}
Ba-137m	6.2×10^{3}	5.2×10^{3}
Ba-140	34.3	2.9×10^{1}
La-140	35.8	3.0×10^{1}
Ce-144	21.8	1.9×10^{1}
Pr-144	21.8	1.9×10^{1}
Mn-54	48.0	4.1×10^{1}
Mn-56	1.2	1.0
Co-58	5065	4300
Co-60	85.0	7.2×10^{1}
Fe-59	18.0	1.5×10^{1}

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-13

MIXED BED DEMINERALIZER (39 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Br-84	0.6	6.63×10^{-1}
Rb-88	28.9	3.19×10^{1}
Rb-89	1.3	1.44
Sr-89	1.04×10^{2}	1.15×10^{2}
Sr-90	15.2	1.68×10^{1}
Sr-91	0.5	5.22×10^{-1}
Sr-92	5.2×10^{-2}	5.74×10^{-2}
Y-90	7.5	8.28
Y-91	10.8	1.19×10^{1}
Y-92	0.12	1.33×10^{-1}
Zr-95	26.7	2.95×10^{1}
Nb-95	39.7	4.38×10^{1}
Mo-99	1.31×10^{3}	1.45×10^{3}
I-131	1.25×10^{4}	1.38×10^{4}
I-132	1.84×10^{3}	2.03×10^{3}
I-133	2.18×10^{3}	2.41×10^{3}
I-134	13.4	1.48×10^{1}
I-135	3.77×10^{2}	4.16×10^{2}
Te-132	5.4×10^{2}	5.96×10^{2}
Te-134	0.54	5.96×10^{-1}
Cs-134	1.01×10^{4}	1.12×10^{4}
Cs-136	7.4×10^{2}	8.17×10^{2}
Cs-137	6.6×10^{3}	7.29×10^{3}
Cs-138	12.9	1.42×10^{1}
Ba-137m	6.2×10^{3}	6.85×10^{3}
Ba-140	34.3	3.79×10^{1}
La-140	35.8	3.95×10^{1}
Ce-144	21.8	2.41×10^{1}
Pr-144	21.8	2.41×10^{1}
Mn-54	48.0	5.3×10^{1}
Mn-56	1.2	1.33
Co-58	5065.0	5590.0
Co-60	85.0	9.39×10^{1}
Fe-59	18.0	1.99×10^{1}

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-14

CATION BED DEMINERALIZER (20 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Y-90	11.4	6.5
Y-91	16.2	9.2
Mo-99	1.96×10^3	1.1×10^3
Cs-134	1.5×10^4	8.4×10^3
CS-136	1.1×10^3	6.2×10^2
Cs-137	9.89×10^3	5.6×10^3
Ba-137m	9.2×10^3	5.2×10^3

NOTE:The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-15

THERMAL REGENERATION DEMINERALIZER (70 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
I-131	81.0	1.6×10^2
I-132	3.4	6.8
I-133	18.5	3.7×10^1
I-134	0.4	7.9×10^{-1}
I-135	4.5	8.9

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-16

RECYCLE EVAPORATOR FEED DEMINERALIZER (30 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
I-131	1.22×10^3	1.0×10^3
I-132	1.73×10^1	1.46×10^2
I-133	1.90×10^2	1.6×10^2
I-134	9.6×10^{-1}	8.2×10^{-1}
I-135	2.98×10^1	2.5×10^1
Cs-134	1.70×10^3	1.5×10^3
Cs-137	1.1×10^3	8.8×10^2
Ba-137m	1.04×10^3	8.3×10^2

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-17

RECYCLE EVAPORATOR CONDENSATE DEMINERALIZER (20 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
I-131	3.8	2.2
I-132	5.6×10^{-2}	3.1×10^{-2}
I-133	0.72	4.1×10^{-1}
I-134	4.4×10^{-3}	2.5×10^{-3}
I-135	0.13	7.4×10^{-2}

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-18

SPENT FUEL PIT DEMINERALIZER (30 ft^3)

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)	INVENTORY (Ci)
Co-58	4.2	3.6
I-131	182.4	150.0
Cs-134	14.5	12.3
Cs-137	5.7	4.8
Ba-137m	5.3	4.5

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-19

REACTOR COOLANT FILTER

ISOTOPE	INVENTORY (Ci)
Co-58	8.9
Co-60	2.35
Cs-134	15.0
Cs-137	9.78
Ba-137m	9.16

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-20

SEAL WATER RETURN FILTER, RECYCLE EVAPORATOR FEED FILTER, SPENT FUEL PIT FILTER, AND SPENT FUEL PIT SKIMMER FILTER

ISOTOPE	INVENTORY (Ci)
Co-58	1.78
Co-60	0.47
Cs-134	3.00
Cs-137	1.96
Ba-137m	1.84

SEAL WATER INJECTION FILTER

ISOTOPE	INVENTORY (Ci)
Co-58	1.17
Co-60	0.30
Cs-134	2.0
Cs-137	1.29
Ba-137m	1.21

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-21

RECYCLE EVAPORATOR CONCENTRATE FILTER

ISOTOPE	INVENTORY (Ci)
Co-58	2.2×10^{-2}
Co-60	5.9×10^{-3}
Cs-134	3.7×10^{-2}
Cs-137	2.4×10^{-2}
Ba-137m	2.2×10^{-2}

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-22

RECYCLE EVAPORATOR CONDENSATE FILTER

ISOTOPE	INVENTORY (Ci)
I-131	2.15×10^{-2}
I-132	3.1×10^{-4}
I-133	4.1×10^{-3}
I-134	2.5×10^{-5}
I-135	7.1×10^{-4}

NOTE: The average sources expected during normal operation are assumed to be 20% of the maximum values listed. This average is based on operating experience to date with Westinghouse's PWRs using Zircaloy-clad fuel.

TABLE 12.2-23

CORE SHUTDOWN SOURCES - (MeV/cm^3-sec)

TIME AFTER SHUTDOWN

PHOTON ENERGY (MeV)	4 HOURS	12 HOURS	1 DAY	1 WEEK	1 MONTH	3 MONTHS
0.4	3.1×10^{11}	2.3×10^{11}	1.9×10^{11}	9.2×10^{10}	3.8×10^{10}	1.3×10^{10}
0.8	1.3×10^{12}	9.8×10^{11}	8.0×10^{11}	4.0×10^{11}	2.3×10^{11}	1.2×10^{11}
1.3	3.9×10^{11}	2.9×10^{11}	2.5×10^{11}	1.6×10^{11}	1.2×10^{11}	5.8×10^{10}
1.7	5.1×10^{11}	3.8×10^{11}	3.3×10^{11}	2.3×10^{11}	6.2×10^{10}	2.9×10^{9}
2.2	7.2×10^{10}	2.6×10^{10}	1.5×10^{10}	8.5×10^{9}	6.7×10^{9}	5.0×10^{9}
2.5	8.9×10^{10}	4.7×10^{10}	3.7×10^{10}	2.5×10^{10}	7.9×10^{9}	3.5×10^{8}
3.5	8.2×10^{9}	2.0×10^{9}	1.3×10^{9}	9.6×10^{8}	2.0×10^{8}	1.5×10^{7}

TABLE 12.2-24

IRRADIATED Ag-In-Cd CONTROL ROD SOURCES (Ci/cm/rod)

TIME AFTER SHUTDOWN

ISOTOPE	0	1 WEEK	1 MONTH	6 MONTHS	1 YEAR
Ag-110m	50	49	46	30	18

TABLE 12.2-24a

HAFNIUM CONTROL ROD SOURCE STRENGTHS

400-DAY IRRADIATION

ENERGY GROUP (MeV)	SOURCE STRENGTH AT TIME AFTER SHUTDOWN (Mev/cm-s) 1 DAY	1 WEEK	1 MONTH	6 MONTHS	1 YEAR
0.20 - 0.40	2.2×10^{10}	2.0×10^{10}	1.4×10^{10}	1.3×10^{9}	1.0×10^{8}
0.40 - 0.90	1.9×10^{11}	1.7×10^{11}	1.2×10^{11}	1.0×10^{10}	5.0×10^{8}
0.90 - 1.35	2.6×10^{10}	2.5×10^{10}	2.2×10^{10}	8.9×10^{9}	2.9×10^{9}

15-YEAR IRRADIATION

ENERGY GROUP (MeV)	SOURCE STRENGTH AT TIME AFTER SHUTDOWN (Mev/cm-s) 1 DAY	1 WEEK	1 MONTH	6 MONTHS	1 YEAR
0.20 - 0.40	4.7×10^{10}	4.3×10^{10}	2.9×10^{10}	3.8×10^{9}	6.2×10^{8}
0.40 - 0.90	3.5×10^{11}	3.2×10^{11}	2.2×10^{11}	1.9×10^{10}	9.3×10^{8}
0.90 - 1.35	2.8×10^{11}	2.7×10^{11}	2.4×10^{11}	9.7×10^{10}	3.1×10^{10}

*Source strengths are expressed per cm^3 of absorber. Density of the hafnium absorber is 13.31 g/cm^3.

TABLE 12.2-25

REFUELING WATER ACTIVITY CONCENTRATIONS

RESULTING IN 2.5 mrem/hr AT THE SURFACE

SOURCE OF ACTIVITY	DOMINANT ISOTOPE	MAXIMUM ALLOWABLE CONCENTRATION (μCI/cm^3)
A. Fission Product Gases	Xe-133	0.15
B. Fission Product Particulates	Cs-137	0.005
C. Corrosion Products	Co-58	0.005
D. Fission Product Halogens	I-131	0.01

TABLE 12.2-26

INCORE INSTRUMENTS - FISSION CHAMBER SOURCES*

(Bases: Irradiation Period = 3 Months
Decay Period = 1 Day)

GAMMA ENERGY GROUP MeV	ACTIVITY (MeV/sec)
0.4	8.1×10^{9}
0.8	4.7×10^{10}
1.3	7.5×10^{8}
1.7	1.9×10^{10}
2.2	5.0×10^{8}
2.5	1.7×10^{9}
3.5	5.0×10^{7}

*Spectrum represents irradiated Ag-110m.

TABLE 12.2-27

DRIVE WIRE SOURCES

(Bases: Irradiation Period = 1 Year
No Decay)

ISOTOPE	ACTIVITY (μCi/cm)
Mn-54	2.78×10^{4}
Mn-56	6.48×10^{5}
Fe-59	2.04×10^{4}
Co-58	5.41×10^{3}
Co-60	3.08×10^{3}

TABLE 12.2-28

SINGLE WASTE GAS DECAY TANK ACTIVITIES

(Single Unit)

ISOTOPE	ACTIVITY (Ci)
Kr-85	6.3×10^{2} (peak)
Kr-85m	1.3×10^{2}
Kr-87	2.0×10^{1}
Kr-88	1.7×10^{2}
Xe-131m	2.2×10^{2}
Xe-133	3.2×10^{4}
Xe-133m	2.2×10^{3}
Xe-135	5.4×10^{2}
Xe-135m	less than 1
Xe-138	less than 1

TABLE 12.2-29

SPENT FUEL PIT WATER ACTIVITY FOR A FUEL HANDLING ACCIDENT

ISOTOPE	ACTIVITY ($\mu Ci/cm^3$)
I-131	13.1
I-132	7.16
I-133	10.4
I-134	2.34
I-135	5.31
Kr-85m	1.65
Kr-85	1.77
Kr-87	0.969
Kr-88	4.47
Xe-131m	0.120
Xe-133	23.9
Xe-33m	2.51
Xe-135	1.88
Xe-135m	0.177
Xe-138	1.06

NOTE: These activities are the maximum gap activities of one fuel assembly distributed in the fuel pit water.

TABLE 12.2-30

SHIELDING DESIGN-BASIS INFLUENT RADIOACTIVITY CONCENTRATION IN LIQUID WASTE PROCESSING STREAMS

ISOTOPE	STEAM GENERATOR BLOWDOWN** (μCi/gm)	CHEMICAL DRAINS (μCi/gm)	REGEN. WASTE DRAINS (μCi/gm)	LAUNDRY DRAINS (μCi/gm)
H-3	2.6-02*	3.5-03	3.5-02	3.5-06
Na-24	2.4-07	2.0-03	-	2.9-06
Cr-51	7.1-06	9.6-07	9.6-06	9.6-10
Mn-54	5.9-06	7.9-07	7.9-06	3.5-05
Mn-56	2.2-04	3.0-05	3.0-04	3.0-08
Co-58	1.9-04	2.6-05	2.6-04	7.3-05
Fe-59	8.5-06	1.1-06	1.1-05	1.5-06
Co-60	7.0-02	1.0-06	1.0-05	2.6-05
Br-84	3.2-04	4.3-05	4.3-04	4.3-08
Rb-88	2.8-02	3.7-03	3.7-02	3.7-06
Rb-89	1.6-03	2.1-04	2.1-03	2.1-07
Sr-89	2.5-05	3.3-06	3.3-05	3.3-09
Sr-90	1.3-06	1.7-07	1.7-06	2.0-06
Y-90	1.5-06	2.0-07	2.0-06	2.0-10
Sr-91	1.4-05	1.9-06	1.9-05	1.9-09
Y-91	4.5-05	6.1-06	6.1-05	6.1-09
Sr-92	5.5-06	7.4-07	7.4-06	7.4-10
Y-92	4.5-06	7.2-07	7.2-06	7.2-10
Y-93	-	8.0-05	-	-
Zr-95	5.2-06	7.0-07	7.0-06	7.0-10
Nb-95	5.1-06	6.9-07	6.9-06	6.9-10
Mo-99	4.0-02	5.3-03	5.3-02	5.3-06
Sb-124	-	1.2-05	-	-
I-131	1.9-02	2.5-03	2.5-02	5.0-04
Te-132	1.6-03	2.3-04	2.3-03	2.3-07
I-132	2.1-02	2.8-03	2.8-02	2.8-06
I-133	3.0-02	4.0-03	4.0-02	4.0-06
Te-134	2.1-04	2.9-05	2.9-04	2.9-08
I-134	4.4-03	6.0-04	6.0-03	6.0-07
Cs-134	1.7-02	2.3-03	2.3-02	1.0-05
I-135	1.6-02	2.2-03	2.2-02	2.2-06
Cs-136	2.1-02	2.8-03	2.8-02	2.8-06
Cs-137	1.1-02	1.5-03	1.5-02	2.0-05
Ba-137m	1.0-02	1.4-03	1.4-02	1.9-05
Cs-138	7.1-03	9.8-04	9.8-03	9.8-07
Ba-140	3.2-05	4.3-06	4.3-05	4.3-09
La-140	1.1-05	1.5-06	1.5-05	1.5-09
Ce-141	-	1.7-07	-	-
Ce-144	2.5-06	3.4-07	3.4-06	3.4-10
Pr-144	2.5-06	3.4-07	3.4-06	3.4-10

* 2.6-02 means 2.6 x 10^{-2}

** Shielding was determined based on equipment 1/2WX02MA,B (housing-only prefilter vessels).

TABLE 12.2-31

SHIELDING DESIGN-BASIS INFLUENT RADIOACTIVITY CONCENTRATIONS IN LIQUID WASTE PROCESSING STREAMS

ISOTOPE	AUX. BLDG. EQUIPMENT DRAINS (μCi/gm)	AUX. BLDG. FLOOR DRAINS (μCi/gm)	TURBINE BUILDING EQUIPMENT DRAINS (μCi/gm)	TURBINE BLDG. FLOOR DRAINS (μCi/gm)
H-3	7.0-01*	7.0-02	1.3-04	1.3-04
C-14	4.0-06	8.0-06	-	-
Na-24	3.0-04	2.0-05	-	-
Cr-51	1.9-04	1.9-05	3.6-09	3.6-09
Mn-54	1.6-04	1.6-05	2.9-09	2.9-09
Fe-55	2.0-04	1.0-03	-	-
Mn-56	6.0-03	6.0-04	1.1-07	1.1-07
Co-58	5.2-03	5.2-04	9.6-08	9.6-08
Fe-59	2.2-04	2.2-05	1.5-09	1.5-09
Co-60	2.0-04	2.0-05	3.6-09	3.6-09
Ni-63	4.0-05	8.0-05	-	-
Br-84	8.6-03	8.6-04	1.6-06	1.6-06
Rb-88	7.4-01	7.4-02	1.4-05	1.4-05
Rb-89	4.2-02	4.2-03	7.8-07	7.8-07
Sr-89	6.6-04	6.6-05	1.2-08	1.2-08
Sr-90	3.4-05	3.4-06	6.3-10	6.3-10
Y-90	4.0-05	4.0-06	7.4-10	7.4-10
Sr-91	3.8-04	3.8-05	7.0-09	7.0-09
Y-91	1.2-03	1.2-04	2.3-08	2.3-08
Sr-92	1.5-04	1.5-05	2.7-09	2.7-09
Y-92	1.4-04	1.4-05	2.7-09	2.7-09
Zr-95	1.4-04	1.4-05	2.6-09	2.6-09
Nb-95	1.4-04	1.4-05	2.6-09	2.6-09
Mo-99	1.1-00	1.1-01	2.0-05	2.0-05
Ru-103	2.0-05	1.0-06	-	-
Sb-124	6.0-05	1.6-05	-	-
I-131	5.0-01	5.0-02	9.3-05	9.3-05
Te-132	4.5-02	4.5-03	8.3-07	8.3-07
I-132	5.6-01	5.6-02	1.0-04	1.0-04
I-133	8.0-01	8.0-02	1.5-04	1.5-04
Te-134	5.8-03	5.8-04	1.1-07	1.1-07
I-134	1.2-01	1.2-02	2.2-05	2.2-05

*7.0-01 means 7.0×10^{-1}

TABLE 12.2-31 (Cont'd)

ISOTOPE	AUX. BLDG. EQUIPMENT DRAINS (μCi/gm)	AUX. BLDG. FLOOR DRAINS (μCi/gm)	TURBINE BUILDING EQUIPMENT DRAINS (μCi/gm)	TURBINE BLDG. FLOOR DRAINS (μCi/gm)
Cs-134	4.6-01	4.6-02	8.4-06	8.4-06
I-135	4.4-01	4.4-02	8.1-05	8.1-05
Cs-136	5.6-01	5.6-02	1.0-05	1.0-05
Cs-137	3.0-01	3.0-02	5.6-06	5.6-06
Ba-137m	2.8-01	2.8-02	5.2-06	5.2-06
Cs-138	2.0-01	2.0-02	3.6-06	3.6-06
Ba-140	8.6-04	8.6-05	1.6-08	1.6-08
La-140	3.0-04	3.0-05	5.6-09	5.6-09
Ce-144	6.8-05	6.8-06	1.3-09	1.3-09
Pr-144	6.8-05	6.8-06	1.3-09	1.3-09

TABLE 12.2-32

SOURCE BASES FOR DRAIN TANKS

TANK NAME	QUANTITY	TOTAL MAXIMUM DAILY FLOW (gal/day)	FRACTION OF PRIMARY COOLANT CONTAINED
Turbine Bldg. Floor Drain Tank	2	12,000	3.704×10^{-5} for Iodines 3.704×10^{-6} for non-Iodines
Turbine Bldg. Equipment Drains Tank	2	12,000	3.704×10^{-5} for Iodines 3.704×10^{-6} for non-Iodines
Aux. Bldg. Floor Drain Tank	2	16,000	.02
Aux. Bldg. Equipment Drain Tank	2	16,000	.2
Chemical Drain Tank	1	6,000	.001
Chemical/Regeneration Waste Drain Tank	1	10,000	.01
Laundry Drain Tank	1	4,000	1×10^{-6} + Table 12.2-33 Sources.

TABLE 12.2-33

LAUNDRY DRAIN SOURCES USED

IN SHIELDING SOURCE CALCULATION

ISOTOPE	ISOTOPIC ACTIVITY (μCi/cc)
Na-24	2.9×10^{-6}
Mn-54	3.5×10^{-5}
Co-58	7.3×10^{-5}
Fe-59	1.5×10^{-6}
Co-60	2.6×10^{-5}
Sr-90	2.0×10^{-6}
I-131	5.0×10^{-4}
Cs-134	1.0×10^{-5}
Cs-137	2.0×10^{-5}
Ba-137m	1.9×10^{-5}

NOTE: In addition, 1×10^{-6} x primary coolant activity is added to the above inventory.

TABLE 12.2-34

DECONTAMINATION FACTORS USED IN SHIELDING SOURCE

CALCULATION OF LIQUID RADWASTE PROCESSING SYSTEM

AND BLOWDOWN SYSTEM COMPONENTS

ATOMIC NUMBER	ELEMENT	COMPONENT FILTER	DEMINERALIZER	EVAPORATOR
1	H	1	1	1
6	C	1	1	10000
11	Na	1	1	10000
24	Cr	10	10	10000
25	Mn	10	10	10000
26	Fe	10	10	10000
27	Co	10	10	10000
28	Ni	10	10	10000
35	Br	1	10	10000
36	Kr	1	1	1
37	Rb	1	1	10000
38	Sr	1	10	10000
39	Y	1	1	10000
40	Zr	10	10	10000
41	Nb	10	10	10000
42	Mo	1	1	10000
44	Ru	10	10	10000
51	Sb	1	1	10000
52	Te	1	10	10000
53	I	1	10	1000
54	Xe	1	1	1
55	Cs	1	1	10000
56	Ba	1	10	10000
57	La	1	10	10000
58	Ce	10	10	10000
59	Pr	10	10	10000

TABLE 12.2-35

SHIELDING DESIGN-BASIS RADIONUCLIDE CONTENT

IN LIQUID RADWASTE PROCESSING SYSTEM COMPONENTS (Curies)

ISOTOPE	BLOWDOWN MIXED BED DEMINERALIZER	RADWASTE MIXED BED DEMINERALIZER
Cr-51	5.3×10^{-3}	2.3×10^{-6}
Mn-54	5.1×10^{-3}	2.6×10^{-6}
Mn-56	2.2×10^{-3}	-
Co-58	1.6×10^{-1}	7.5×10^{-5}
Fe-59	6.5×10^{-3}	2.9×10^{-6}
Co-60	5.0×10^{-3}	2.6×10^{-6}
Br-84	7.0×10^{-3}	2.1×10^{-6}
Rb-88	4.8×10^{-2}	5.1×10^{-5}
Rb-89	1.4×10^{-3}	1.5×10^{-6}
Sr-89	2.0×10^{-1}	9.1×10^{-5}
Sr-90	1.1×10^{-2}	5.7×10^{-6}
Y-90	8.2×10^{-3}	4.9×10^{-6}
Sr-91	5.2×10^{-3}	1.3×10^{-6}
Sr-92	5.8×10^{-4}	-
Y-92	5.8×10^{-4}	-
Zr-95	4.3×10^{-3}	2.0×10^{-6}
Nb-95	4.5×10^{-3}	2.3×10^{-6}
Mo-99	6.9×10^{-2}	7.4×10^{-5}
I-131	9.6×10^{-1}	3.0×10^{-1}
Xe-131m	2.1×10^{-1}	1.2×10^{-3}
Te-132	5.4	1.4×10^{-3}
I-132	6.0	2.9×10^{-3}
I-133	2.3×10^{1}	5.7×10^{-2}
Xe-133m	5.6×10^{-1}	1.4×10^{-3}
Xe-133	1.9×10^{1}	5.5×10^{-2}
Te-134	6.1×10^{-3}	1.8×10^{-6}
I-134	1.6×10^{-1}	4.3×10^{-4}
Cs-134	3.9×10^{-3}	4.2×10^{-6}

BASES:

1. Time period of collection is 14 days for the blowdown demineralizer and 30 days for the radwaste demineralizer.

2. 1% failed fuel.

TABLE 12.2-35 (Cont'd)

ISOTOPE	BLOWDOWN MIXED BED DEMINERALIZER	RADWASTE MIXED BED DEMINERALIZER
I-135	4.2	1.0×10^{-2}
Xe-135m	1.2	3.0×10^{-3}
Xe-135	4.2	9.9×10^{-3}
Cs-136	2.0×10^{-3}	2.1×10^{-6}
Cs-137	2.0×10^{-2}	2.1×10^{-5}
Cs-138	1.3×10^{-2}	1.4×10^{-5}
Ba-140	2.0×10^{-1}	7.1×10^{-5}
La-140	1.9×10^{-1}	7.2×10^{-5}
Ce-144	2.2×10^{-3}	1.1×10^{-6}
Pr-144	2.2×10^{-3}	1.1×10^{-6}

BASES:

1. Time period of collection is 14 days for the blowdown demineralizer and 30 days for the radwaste demineralizer.

2. 1% failed fuel.

TABLE 12.2-36

SHIELDING DESIGN-BASIS RADIONUCLIDE CONTENT

IN LIQUID RADWASTE PROCESSING SYSTEM COMPONENTS (Curies)

ISOTOPE	CONCENTRATES HOLDING TANK
C-14	1.9×10^{-2}
Na-24	5.1×10^{-2}
Cr-51	4.8×10^{-3}
Mn-54	4.5×10^{-3}
Fe-55	1.9
Mn-56	5.4×10^{-3}
Co-58	1.4×10^{-1}
Fe-59	5.7×10^{-3}
Co-60	5.6×10^{-3}
Ni-63	1.9
Br-84	1.6×10^{-2}
Rb-88	7.8×10^{-1}
Rb-89	3.8×10^{-2}
Sr-89	1.7×10^{-1}
Sr-90	9.6×10^{-3}
Y-90	1.0×10^{-2}
Sr-91	1.2×10^{-2}
Y-91m	6.8×10^{-3}
Y-91	3.2×10^{-1}
Sr-92	1.4×10^{-3}
Y-92	3.1×10^{-3}
Y-93	9.6×10^{-4}
Zr-95	3.7×10^{-3}
Nb-95	3.8×10^{-3}
Mo-99	1.4×10^{2}
Tc-99m	1.1×10^{2}
Ru-103	3.7×10^{-3}
Sb-124	1.1×10^{-1}
I-131	9.9×10^{1}
Xe-131m	1.3×10^{-1}
Te-132	6.3
I-132	2.6×10^{1}
I-133	4.7×10^{1}
Xe-133m	4.1
Xe-133	1.8×10^{1}
Te-134	1.4×10^{-2}
I-134	3.8×10^{-1}

BASES

1. Time collection period is 30 days.
2. 1% failed fuel.

TABLE 12.2-36 (Cont'd)

ISOTOPE	CONCENTRATES HOLDING TANK
Cs-134	1.3×10^{2}
I-135	9.7
Xe-135m	1.7
Xe-135	8.7
Cs-136	1.3×10^{2}
Cs-137	8.4×10^{1}
Ba-137m	7.8×10^{1}
Cs-138	3.7×10^{-1}
Ba-140	1.9×10^{-1}
La-140	1.6×10^{-1}
Ce-141	7.4×10^{-6}
Ce-144	1.9×10^{-3}
Pr-144	1.9×10^{-3}

BASES

1. Time collection period is 30 days.
2. 1% failed fuel.

TABLE 12.2-37

SHIELDING DESIGN-BASIS
RADIONUCLIDE CONTENT IN RADWASTE FILTERS (in Curies)

ISOTOPE	BLOWDOWN PREFILTER*	BLOWDOWN AFTERFILTER	RADWASTE AFTERFILTER	TURBINE BUILDING EQUIPMENT DRAIN FILTER	TURBINE BUILDING FLOOR DRAIN FILTER
Cr-51	7.4×10^{-2}	5.3×10^{-4}	2.3×10^{-7}	3.1×10^{-6}	3.1×10^{-6}
Mn-54	7.1×10^{-2}	5.1×10^{-4}	2.6×10^{-7}	3.5×10^{-6}	3.5×10^{-6}
Fe-55	-	-	1.8×10^{-5}	-	-
Mn-56	3.0×10^{-2}	2.2×10^{-4}	-	7.0×10^{-7}	7.0×10^{-7}
Co-58	2.2	1.6×10^{-2}	7.5×10^{-6}	1.0×10^{-4}	1.0×10^{-4}
Fe-59	9.1×10^{-2}	6.5×10^{-4}	2.9×10^{-7}	4.0×10^{-6}	4.0×10^{-6}
Co-60	9.1×10^{-2}	6.5×10^{-4}	3.4×10^{-7}	4.6×10^{-6}	4.6×10^{-6}
Ni-63	-	-	1.8×10^{-6}	-	-
Zr-95	6.0×10^{-2}	4.3×10^{-4}	2.7×10^{-7}	2.7×10^{-6}	2.7×10^{-6}
Nb-95	6.3×10^{-2}	4.5×10^{-4}	2.2×10^{-7}	3.1×10^{-6}	3.1×10^{-6}
Ru-103	-	-	3.4×10^{-8}	-	-
Sb-124	-	-	6.9×10^{-11}	-	-
I-131	1.6×10^{-4}	1.5×10^{-5}	1.7×10^{-7}	8.1×10^{-7}	8.1×10^{-7}
Ce-144	3.1×10^{-2}	2.2×10^{-4}	1.1×10^{-7}	1.5×10^{-6}	1.5×10^{-6}
Pr-144	3.1×10^{-2}	2.2×10^{-4}	1.1×10^{-7}	1.5×10^{-6}	1.5×10^{-6}

BASES:
1. Maximum daily flow rates given in Table 12.2-32 except for the blowdown stream which has maximum flow rate of 135 gpm.
2. 1% failed fuel.
3. Time period of collection is 14 days for the blowdown filters and 30 days for the radwaste and turbine building filters.
4. Primary to secondary steam generator leakage of 1 gpm.

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* Shielding was determined based on equipment 1/2WX02MA,B (housing-only prefilter vessels).

TABLE 12.2-38

SHIELDING DESIGN-BASIS

RADIONUCLIDE CONTENT IN RADWASTE FILTERS (in Curies)

ISOTOPE	AUX. BLDG. EQUIP. DRAIN FILTER	AUX. BLDG. FLOOR DRAIN FILTER	REGENERATION WASTE DRAIN FILTER	CHEMICAL DRAIN FILTER	LAUNDRY DRAIN FILTER
Cr-51	2.2×10^{-1}	2.2×10^{-2}	6.9×10^{-3}	4.2×10^{-4}	-
Mn-54	2.5×10^{-1}	2.5×10^{-2}	7.8×10^{-3}	4.7×10^{-4}	7.9×10^{-3}
Fe-55	3.3×10^{-1}	1.6	-	-	-
Mn-56	5.1×10^{-2}	5.1×10^{-3}	1.6×10^{-3}	9.5×10^{-5}	-
Co-58	7.4	7.4×10^{-1}	2.3×10^{-1}	1.4×10^{-2}	2.6×10^{-2}
Fe-59	2.9×10^{-1}	2.9×10^{-2}	9.0×10^{-3}	5.4×10^{-4}	-
Co-60	3.3×10^{-1}	3.3×10^{-2}	1.0×10^{-3}	6.1×10^{-4}	4.1×10^{-3}
Ni-63	6.5×10^{-2}	1.3×10^{-1}	-	-	-
Zr-95	2.0×10^{-6}	2.0×10^{-7}	6.1×10^{-3}	3.7×10^{-4}	-
Nb-95	2.2×10^{-1}	2.2×10^{-2}	6.9×10^{-3}	4.1×10^{-4}	-
Mo-99	9.3×10^{-3}	9.3×10^{-4}	4.6×10^{-4}	4.6×10^{-5}	-
Ru-103	2.5×10^{-3}	1.3×10^{-3}	-	-	-
Sb-124	-	-	-	2.9×10^{-11}	-
I-131	4.4×10^{-3}	4.4×10^{-4}	2.2×10^{-4}	2.2×10^{-5}	-
I-132	5.0×10^{-3}	5.0×10^{-4}	2.4×10^{-4}	2.4×10^{-5}	-
I-133	$7.0 \text{ X } 10^{-3}$	$7.0 \text{ X } 10^{-4}$	$3.5 \text{ X } 10^{-4}$	$3.5 \text{ X } 10^{-5}$	-
I-134	$1.1 \text{ X } 10^{-3}$	$1.1 \text{ X } 10^{-4}$	$5.3 \text{ X } 10^{-5}$	$5.3 \text{ X } 10^{-6}$	-
I-135	$3.9 \text{ X } 10^{-3}$	$3.9 \text{ X } 10^{-4}$	$1.9 \text{ X } 10^{-4}$	$1.9 \text{ X } 10^{-5}$	-
Cs-137	2.6×10^{-3}	2.6×10^{-4}	1.3×10^{-4}	1.3×10^{-5}	-
Cs-138	1.7×10^{-3}	1.7×10^{-4}	-	-	-
Ce-144	1.1×10^{-1}	1.1×10^{-2}	3.4×10^{-3}	2.0×10^{-4}	-
Pr-144	1.1×10^{-1}	1.1×10^{-2}	3.4×10^{-3}	2.0×10^{-4}	-

BASES:

1. Maximum daily flow rates given in Table 12.2-32.
2. 1% failed fuel.
3. Time period of collection is 30 days.
4. Primary to secondary steam generator leakage of 1 gpm.

TABLE 12.2-39

SHIELDING DESIGN-BASIS RADIONUCLIDE CONTENT

IN THE LIQUID RADWASTE PROCESSING SYSTEM COMPONENTS (Curies)

ISOTOPE	RADWASTE EVAPORATOR
C-14	1.96×10^{-2}
Na-24	5.30×10^{-2}
Cr-51	4.97×10^{-3}
Mn-54	4.52×10^{-3}
Fe-55	1.97
Mn-56	5.56×10^{-3}
Co-58	1.44×10^{-1}
Fe-59	5.93×10^{-3}
Co-60	5.77×10^{-3}
Ni-63	1.96×10^{-1}
Br-84	1.68×10^{-2}
Rb-88	8.12×10^{-1}
Rb-89	3.99×10^{-2}
Sv-89	1.81×10^{-1}
Y-89m	1.81×10^{-5}
Sr-90	9.82×10^{-3}
Y-90	1.06×10^{-2}
Sr-91	1.21×10^{-2}
Y-91m	7.05×10^{-3}
Y-91	3.35×10^{-1}
Sr-92	1.44×10^{-3}
Y-92	3.18×10^{-3}
Y-93	1.0×10^{-3}
Zr-95	3.86×10^{-3}
Nb-95m	3.57×10^{-5}
Nb-95	3.97×10^{-3}
Mo-99	1.42×10^{-2}
Tc-99m	1.15×10^{-2}
Tc-99	5.58×10^{-6}
Ru-103	3.8×10^{-3}
Sb-124	1.11×10^{-1}
I-131	1.03×10^{-2}
Xe-131m	1.33×10^{-1}
Te-132	6.53
I-132	2.73×10^{1}
I-133	4.91×10^{1}
Xe-133m	4.21
Xe-133	1.85×10^{1}
Te-134	1.49×10^{-2}
I-134	3.97×10^{-1}
Cs-134	1.33×10^{2}
I-135	1.01×10^{1}
Xe-135m	1.72
Xe-135	9.01

TABLE 12.2-39 (Cont'd)

ISOTOPE	RADWASTE EVAPORATOR
Cs-135	2.66×10^{-8}
Cs-136	1.30×10^{2}
Cs-137	8.67×10^{1}
Ba-137m	8.10×10^{1}
Cs-138	3.88×10^{-1}
Ba-140	1.98×10^{-1}
La-140	1.60×10^{-1}
Ce-141	7.70×10^{-6}
Ce-144	1.94×10^{-3}
Pr-144	1.94×10^{-3}

TABLE 12.2-40

SHIELDING DESIGN-BASIS RADIONUCLIDE CONTENT

IN LIQUID RADWASTE PROCESSING SYSTEM COMPONENTS (Curies)

ISOTOPE	30,000-GALLON RELEASE TANK	LAUNDRY DRAIN TANK
H-3	3.51×10^{-3}	2.65×10^{-5}
Na-24	2.59×10^{-4}	2.19×10^{-5}
Cr-51	1.73×10^{-8}	7.26×10^{-10}
Mn-54	3.10×10^{-4}	2.61×10^{-5}
Mr-56	5.41×10^{-7}	2.27×10^{-8}
Co-58	6.47×10^{-4}	5.51×10^{-5}
Fe-59	1.33×10^{-5}	1.10×10^{-6}
Co-60	2.31×10^{-4}	1.97×10^{-5}
Pr-84	4.32×10^{-5}	3.25×10^{-7}
Rb-88	6.67×10^{-4}	2.80×10^{-5}
Rb-89	3.78×10^{-5}	1.59×10^{-6}
Sr-89	5.95×10^{-7}	2.50×10^{-8}
Sr-90	1.78×10^{-4}	1.51×10^{-5}
Y-90	3.61×10^{-8}	1.51×10^{-9}
Sr-91	3.43×10^{-7}	1.44×10^{-8}
Y-91	1.10×10^{-6}	4.62×10^{-8}
Sr-92	1.33×10^{-7}	5.60×10^{-9}
Y-92	1.30×10^{-7}	5.45×10^{-9}
Zr-95	1.26×10^{-8}	5.30×10^{-10}
Nb-95	1.24×10^{-8}	5.22×10^{-10}
Mo-99	9.56×10^{-4}	4.01×10^{-5}
I-131	6.28×10^{-3}	4.00×10^{-3}
Te-132	4.07×10^{-5}	1.71×10^{-6}
I-132	2.55×10^{-4}	2.10×10^{-5}
I-133	4.01×10^{-3}	3.03×10^{-5}
Te-134	5.23×10^{-6}	2.20×10^{-7}
I-134	6.02×10^{-4}	4.54×10^{-6}
Cs-134	9.16×10^{-4}	7.57×10^{-5}
I-135	2.21×10^{-3}	1.66×10^{-5}
Cs-136	5.04×10^{-4}	2.12×10^{-5}
Cs-137	1.91×10^{-3}	1.51×10^{-4}
Ba-137m	1.78×10^{-3}	1.41×10^{-4}
Cs-138	1.77×10^{-4}	7.42×10^{-6}
Ba-140	7.75×10^{-7}	3.25×10^{-8}
La-140	2.70×10^{-7}	1.14×10^{-8}
Ce-144	6.13×10^{-9}	2.57×10^{-10}
Pr-144	6.13×10^{-9}	2.57×10^{-10}

TABLE 12.2-41

SHIELDING DESIGN-BASIS RADIONUCLIDE CONTENT

IN LIQUID RADWASTE PROCESSING SYSTEM COMPONENTS (Curies)

ISOTOPE	BLOWDOWN MONITOR TANK	RADWASTE EVAPORATOR MONITOR TANK
H-3	1.88	2.00×10^{1}
C-14	-	9.10×10^{-8}
Na-24	1.82×10^{-5}	1.74×10^{-5}
Cr-51	5.14×10^{-7}	5.49×10^{-10}
Mn-54	4.23×10^{-7}	4.52×10^{-10}
Fe-55	-	5.5×10^{-6}
Mn-56	1.61×10^{-5}	1.71×10^{-8}
Co-58	1.39×10^{-5}	1.49×10^{-8}
Fe-59	5.90×10^{-7}	6.29×10^{-10}
Co-60	5.37×10^{-7}	5.72×10^{-10}
Ni-63	-	9.08×10^{-9}
Br-84	2.31×10^{-3}	2.46×10^{-6}
Rb-88	1.98	2.12×10^{-3}
Rb-89	1.13×10^{-1}	1.20×10^{-4}
Sr-89	1.77×10^{-4}	1.89×10^{-7}
Sr-90	9.11×10^{-6}	9.72×10^{-9}
Y-90	1.07×10^{-4}	1.14×10^{-7}
Sr-91	1.02×10^{-4}	1.09×10^{-7}
Y-91	3.27×10^{-3}	3.49×10^{-6}
Sr-92	3.97×10^{-5}	4.23×10^{-8}
Y-92	3.86×10^{-4}	4.12×10^{-7}
Y-93	-	6.07×10^{-7}
Zr-95	3.75×10^{-7}	4.00×10^{-10}
Nb-95	3.70×10^{-7}	3.94×10^{-10}
Mo-99	2.84	3.03×10^{-3}
Ru-103	-	2.25×10^{-10}
Sb-124	-	6.60×10^{-7}
I-131	1.34×10^{-1}	1.43×10^{-3}
Te-132	1.21×10^{-2}	1.29×10^{-5}
I-132	1.49×10^{-1}	1.59×10^{-3}
I-133	2.14×10^{-1}	2.29×10^{-3}
Te-134	1.55×10^{-3}	1.66×10^{-6}
I-134	3.22×10^{-2}	3.43×10^{-4}
Cs-134	1.23	1.31×10^{-3}
I-135	1.18×10^{-1}	1.26×10^{-3}
Cs-136	1.50	1.60×10^{-3}
Cs-137	8.04×10^{-2}	8.57×10^{-4}
Ba-137m	7.48×10^{-2}	7.97×10^{-3}
Cs-138	5.25×10^{-1}	5.60×10^{-4}
Ba-140	2.30×10^{-4}	2.46×10^{-7}
La-140	8.04×10^{-5}	8.57×10^{-8}
Ce-141	-	2.9×10^{-11}
Ce-144	1.82×10^{-7}	1.94×10^{-10}
Pr-144	1.82×10^{-7}	1.94×10^{-10}

TABLE 12.2-42

ASSUMED DEMINERALIZER RESIN

INVENTORY IN SPENT RESIN TANK

FOR SHIELDING SOURCES CALCULATION

QUANTITY	COMPONENT NAME	VOLUME OF RESIN IN EACH (ft^3)	FRACTIONAL CONTRIBUTION TO SRST INVENTORY
4	Letdown Mixed Bed Demineralizers	30	.16
2	Cation Demineralizers	20	.053
2	Recycle Evaporator Feed Demineralizer	30	.08
2	Recycle Evaporator Condensate Demineralizer	20	.053
7	Boron Thermal Regeneration Demineralizers	70	.65

TABLE 12.2-42

ASSUMED DEMINERALIZER RESIN

INVENTORY IN HIGH ACTIVITY SPENT RESIN TANK

FOR SHIELDING SOURCES CALCULATION

QUANTITY	COMPONENT NAME	VOLUME OF RESIN IN EACH (ft^3)	FRACTIONAL CONTRIBUTION TO SRST INVENTORY
4	Letdown Mixed Bed Demineralizers	35	.182
2	Cation Demineralizers	20	.052
2	Recycle Evaporator Feed Demineralizer	30	.078
2	Recycle Evaporator Condensate Demineralizer	20	.052
7	Boron Thermal Regeneration Demineralizers	70	.636

TABLE 12.2-43

SPENT RESIN TANK

SHIELDING DESIGN-BASIS

RADIONUCLIDE CONTENT (in Curies)

ISOTOPE	ACTIVITY (Curies) NO DECAY	ACTIVITY (Curies)* 90-DAY DECAY
Br-84	2.0	-
Rb-88	9.8×10^{1}	-
Rb-89	4.55	-
Sr-89	3.54×10^{2}	1.03×10^{2}
Sr-90	5.2×10^{1}	5.2×10^{1}
Sr-91	1.7	-
Sr-92	1.8×10^{-1}	-
Y-90	3.84×10^{1}	5.1×10^{1}
Y-91	5.51×10^{1}	1.9×10^{1}
Y-92	4.1×10^{-1}	-
Zr-95	9.1×10^{1}	3.5×10^{1}
Nb-95	1.3×10^{2}	6.5×10^{1}
Mo-99	6.6×10^{3}	-
I-131	4.6×10^{4}	2.0×10^{1}
I-132	6.5×10^{3}	-
I-133	8.0×10^{3}	-
I-134	5.3×10^{1}	-
I-135	1.4×10^{3}	-
Te-132	1.6×10^{3}	-
Te-134	1.8	-
Cs-134	5.4×10^{4}	5.0×10^{4}
Cs-136	3.7×10^{3}	3.2×10^{1}
Cs-137	3.5×10^{4}	3.5×10^{4}
Cs-138	4.38×10^{1}	-
Ba-137m	3.3×10^{4}	3.2×10^{4}
Ba-140	1.16×10^{2}	8.9×10^{-1}
La-140	1.2×10^{2}	1.0
Ce-144	7.4×10^{1}	6.0×10^{1}
Pr-144	7.4×10^{1}	6.0×10^{1}
Mn-54	1.63×10^{2}	1.3×10^{2}
Mn-56	4.1	-
Co-58	2.2×10^{3}	9.2×10^{2}
Co-60	2.9×10^{2}	2.9×10^{2}
Fe-59	6.1×10^{1}	1.5×10^{1}

* neglected below 10^{-1} activity

TABLE 12.2-43

HIGH ACTIVITY SPENT RESIN TANK

SHIELDING DESIGN-BASIS

RADIONUCLIDE CONTENT (in Curies)

ISOTOPE	ACTIVITY (Curies) NO DECAY	ACTIVITY (Curies)* 90-DAY DECAY
Br-84	2.0	-
Rb-88	9.8×10^{1}	-
Rb-89	4.55	-
Sr-89	3.54×10^{2}	1.03×10^{2}
Sr-90	5.2×10^{1}	5.2×10^{1}
Sr-91	1.7	-
Sr-92	1.8×10^{-1}	-
Y-90	3.84×10^{1}	5.1×10^{1}
Y-91	5.51×10^{1}	1.9×10^{1}
Y-92	4.1×10^{-1}	-
Zr-95	9.1×10^{1}	3.5×10^{1}
Nb-95	1.3×10^{2}	6.5×10^{1}
Mo-99	6.6×10^{3}	-
I-131	4.6×10^{4}	2.0×10^{1}
I-132	6.5×10^{3}	-
I-133	8.0×10^{3}	-
I-134	5.3×10^{1}	-
I-135	1.4×10^{3}	-
Te-132	1.6×10^{3}	-
Te-134	1.8	-
Cs-134	5.4×10^{4}	5.0×10^{4}
Cs-136	3.7×10^{3}	3.2×10^{1}
Cs-137	3.5×10^{4}	3.5×10^{4}
Cs-138	4.38×10^{1}	-
Ba-137m	3.3×10^{4}	3.2×10^{4}
Ba-140	1.16×10^{2}	8.9×10^{-1}
La-140	1.2×10^{2}	1.0
Ce-144	7.4×10^{1}	6.0×10^{1}
Pr-144	7.4×10^{1}	6.0×10^{1}
Mn-54	1.63×10^{2}	1.3×10^{2}
Mn-56	4.1	-
Co-58	2.2×10^{3}	9.2×10^{2}
Co-60	2.9×10^{2}	2.9×10^{2}
Fe-59	6.1×10^{1}	1.5×10^{1}

* neglected below 10^{-1} activity

TABLE 12.2-44

COMPOSITION OF A SINGLE 55-GALLON

RADWASTE DRUM FOR SHIELDING ANALYSIS

OF DRUM STORAGE AREAS

1. Spent Resin

MIXTURE COMPONENT	DENSITY (lb/ft^3)	VOLUME (ft^3)	WEIGHT (lb)
Radioactive water and spent resins	75	4.5	340
Cement	94	2.85	270
		7.35	610

NOTE: Drum composition from the volume reduction system has been intentionally deleted from this table. Braidwood and Byron stations do not intend to use this equipment.

TABLE 12.2-45

DESIGN-BASIS SHIELDING SOURCES FOR

MAIN AUXILIARY BUILDING CHARCOAL AIR

FILTER AND OFF-GAS VENT FILTER*

ISOTOPE	MAIN AUX. BLDG. CHARCOAL AIR FILTER	OFF-GAS VENT FILTER**
Br-84	6.4×10^{-8}	8.0×10^{-6}
I-131	1.8×10^{-3}	1.2×10^{-1}
I-132	7.1×10^{-6}	7.3×10^{-4}
I-133	3.0×10^{-4}	3.0×10^{-2}
I-134	1.7×10^{-6}	1.8×10^{-4}
I-135	5.3×10^{-5}	5.2×10^{-3}

* Values given are in curies per filter.

** Charcoal filters in series are considered to be one filter.

TABLE 12.2-46

AUXILIARY BUILDING RADIOACTIVE AIRBORNE DESIGN-BASIS CONCENTRATION

EXPRESSED IN MPC*

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 330'-0"					
Auxiliary building Floor Drain Pump Room	2.90-4**	550	2.46-4	1.33-1	1.58-3
Auxiliary Building Floor Drain Sump	3.10-4	400	3.62-5	1.95-2	2.32-4
El. 344'-6					
Recycle evaporator room	1.32-3	4300	2.16-3	2.18+0	5.87-2
El. 346'-0					
Auxiliary Building Collection Drain Sump Room	2.80-4	1910	6.85-6	4.47-3	4.38-5
Auxiliary Building Equipment Drain Tank Room	2.90-4	1570	8.63-5	4.65-2	5.53-4

* Maximum Permissible Concentration, consistent with regulations that were in effect at the time of analysis

** Read as 2.90×10^{-4}

TABLE 12.2-46

AUXILIARY BUILDING RADIOACTIVE AIRBORNE DESIGN-BASIS CONCENTRATION

EXPRESSED IN MPC*

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 330'-0"					
Auxiliary building Floor Drain Pump Room	2.90-4**	550	2.46-4	1.33-1	1.58-3
Auxiliary Building Floor Drain Sump	3.10-4	400	3.62-5	1.95-2	2.32-4
El. 344'-6					
Recycle evaporator room	1.32-3	4300	2.16-3	2.18+0	5.87-2
El. 346'-0					
Unit 1 Auxiliary Building Collection Drain Sump Room	2.8-4	1910	6.85-6	4.47-3	4.38-5
Unit 2 Auxiliary Building Collection Drain Sump Room/ Hot Machine Shop	2.8-4	1910	6.85-6	4.47-3	4.38-5
Auxiliary Building Equipment Drain Tank Room	2.90-4	1570	8.63-5	4.65-2	5.53-4

* Maximum Permissible Concentration, consistent with regulations that were in effect at the time of analysis

** Read as 2.90×10^{-4}

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 346'-0" (Cont'd)					
Heat Exchanger Valve Aisle	2.20-4	1600	1.70-3	1.73-1	2.05-1
Letdown Chiller Heat Exchanger Room	8.99-5	750	1.49-3	1.52-1	1.79-1
Letdown Reheat Heat Exchanger Room	7.00-5	750	1.16-3	1.18-1	1.40-1
Moderating Heat Exchanger Room	1.10-4	750	1.82-3	1.84-1	2.19-1
Recycle Evaporator Feed Pump Valve Aisle	3.99-5	3200	1.55-4	1.58-2	1.87-4
Recycle Evaporator Feed Pump Room	1.30-4	1600	1.16-3	1.18-1	1.40-3
Recycle Holdup Tank Room - OA	1.95-4	8000	1.01-4	1.99-2	1.22-4
Recycle Holdup Tank Room - OB	3.15-4	5750	2.26-4	4.46-2	2.73-4
Regenerative Waste Drain Tank Room	2.70-4	4300	9.77-6	5.20-1	9.52-4
Residual Heat Removal Pump Room	3.50-4	1000	4.33-3	4.94-1	3.13-1

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 346'-0" (Cont'd)					
Waste Gas Decay Tank Valve Aisle	1.37-3	17500	2.00-6	5.19-1	1.28-5
Waste Gas Decay Tank Room	9.99-5	4400	2.00-6	6.90-1	1.28-5
El. 355'-4", 358'-2"					
Waste Gas Decay Tank & Recycle Evaporator Pipe Tunnel	1.45-3	26100	7.14-4	2.07+0	2.26-2
El. 357'-0"					
Residual Heat Removal Heat Exchanger Room	1.90-4	1400	1.68-3	1.92-1	1.26-1
El. 364'-0"					
Auxiliary Building Floor Drain Pump Room	3.40-4	1000	1.59-5	8.58-3	1.02-4
Auxiliary Building Floor Drain Tank Room	9.99-5	750	6.23-6	3.36-3	2.62-4
Blowdown Condenser - Unit 1	4.49-4	4760	1.64-6	-	4.02-6
Blowdown Condenser - Unit 2	4.29-4	2760	2.70-6	-	1.76-5
Centrifugal Charging Pump Room - A	3.99-4	1000	4.95-3	4.60-1	5.99-2

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 364'-0" (Cont'd)					
Centrifugal Charging Pump Room - B	3.40-4	750	1.15-2	1.10+0	5.93-1
Chemical Drain Tank Room	9.99-5	1860	1.26-7	6.78-5	8.04-7
Chemical Drain Tank Room	3.40-4	1000	7.94-7	4.29-4	5.08-6
Positive Displacement Charging Pump Room	3.89-4	1000	4.83-3	4.49-1	5.84-2
Chemical/Regeneration Waste Drain Pump Room	2.30-4	1000	2.85-5	2.90-3	3.44-3
Chemical/Regeneration Waste Drain Tank Room	6.99-5	2500	3.47-6	3.53-3	4.20-4
Safety Injection Pump Room - A	3.99-4	1000	5.20-3	5.92-1	3.74-1
Safety Injection Pump Room - B	2.40-4	750	9.89-3	1.06+0	8.11-1
El. 364'-0", 383'-0", 401'-0"					
Spray Additive Tank & Pipe Penetration Area	3.99-3	8350	5.93-3	6.05-1	5.25-1
El. 374'-6"					
Recycle Holdup Tank Pipe Tunnel	1.20-4	2500	2.99-4	5.90-2	3.61-4

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 375'-0"					
Pipe Tunnel (Q, 15-18)	6.89-4	1250	1.21-2	1.34+0	1.13+0
El. 383'-0"					
Filter Valve Aisle (M-Q, 11-12)	2.41-3	3360	1.68-3	9.05-1	1.07-2
Filter Valve Aisle (M-P, 13-15)	8.49-4	2010	9.88-4	5.33-1	6.05-3
Filter Pipe Tunnel 1	5.79-4	1760	3.33-3	1.79+0	2.00-2
Filter Pipe Tunnel 2	6.59-4	1600	3.52-3	1.90+0	2.09-2
Filter Pipe Tunnel 3	6.89-4	3810	1.27-3	6.89-1	8.02-3
Heat Exchanger Valve Aisle	2.50-4	1500	2.06-3	2.10-1	2.49-1
Letdown Heat Exchanger Room - A	1.80-4	1300	1.71-3	1.75-1	2.07-1
Letdown Heat Exchanger Room - B	1.60-4	900	2.20-3	2.24-1	2.66-1
Radwaste & Blowdown Mixed Bed Demineralizer Valve Aisle	1.58-3	2500	2.95-4	1.59-1	1.89-3
Blowdown Mixed Bed Demineralizer Cubicle	2.00-4	1500	2.98-4	1.59-1	1.91-3

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 383'-0"					
Radwaste Mixed Bed Demineralizer Cubicle	2.00-4	1000	3.89-4	2.11-1	2.49-3
Seal Water Heat Exchanger Room	1.10-4	800	1.70-3	1.73-1	2.05-1
El. 391'-6"					
Filter Cubicles	5.99-5	260	1.42-3	7.65-1	8.83-3
Filter Cubicles	5.99-5	160	2.55-3	1.37+0	1.63-2
Filter Cubicles	5.99-5	250	1.44-3	7.77-1	8.97-3
El. 394'-6"					
Auxiliary Steam Pipe Tunnel	1.20-4	2000	-	8.68-2	8.56-1
El. 394'-6"					
Pipe Tunnel - Unit 1	1.36-3	8000	2.84-3	6.26-1	2.59-1
Pipe Tunnel - Unit 2	1.36-3	10900	2.72-3	8.03-1	2.29-1
El. 401'-0"					
Boric Acid Tank Room	2.60-4	5800	5.55-4	5.65-2	6.70-2
Main Demineralizer Valve Aisle	1.25-3	8000	3.65-4	1.97-1	2.34-3
Main Demineralizer Cubicles	9.99-5	900	6.24-4	3.37-1	3.99-3

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 401'-0" (Cont'd)					
Main Demineralizer Cubicles	9.99-5	600	7.53-4	4.07-1	4.83-3
Main Demineralizer Cubicles	9.99-5	500	8.31-4	4.50-1	5.33-3
Main Demineralizer Pipe Tunnel	7.99-5	8000	7.09-4	3.83-1	5.75-3
Laundry Drain Tank Room	2.20-4	800	6.43-10	3.47-8	5.14-7
Spent Resin & Concentrates Pump Room	5.99-4	2000	-	1.21-3	8.56-1
Surface Condenser Room - A	1.81-3	5100	-	7.10-2	2.95-2
Surface Condenser Room - B	1.71-3	5700	-	6.00-2	2.48-2
Surface Condenser Room - C	1.56-3	4600	-	6.78-2	2.81-2
El. 414'-0"					
Radwaste Evaporator Room - A	8.99-4	5100	1.46-4	1.09-1	1.02+0
Radwaste Evaporator Room - B	1.41-3	5700	2.02-4	1.13-1	1.40+0
Radwaste Evaporator Room - C	1.41-3	4600	2.50-4	1.33-1	1.73+0

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
<u>El. 417'-0"</u>					
Concentrates Holding Tank Room	2.00-5	3050	8.12-7	7.93-5	2.31-2
Spent Resin Storage Tank	2.00-5	2450	-	-	6.96-2
<u>El. 426'-0</u>					
Laundry Room	3.70-4	4250	2.03-10	1.10-7	1.63-7
Volume Control Tank Valve Aisle	7.40-4	2900	1.96-3	1.01+0	1.17-1
Volume Control Tank Room	1.90-4	2900	2.43-3	1.29+0	1.45-1
Waste Gas Analyzer Rack Room(3)	1.40-4	200	1.17-4	5.97+0	2.15-1
Waste Gas Cabinet Aisle (3)	1.80-4	2500	1.26-5	2.15-1	2.29-2
Purge Room	6.20-4	2550	3.01-3	3.07-1	3.63-1
Waste Gas Compressor Room (3)	4.49-4	1000	1.97-4	2.49+0	3.44-1

(1) The leak rates given in the table are based on leakages of $5x10^{-3}$ lb/hr per valve or flange; $2x10^{-2}$ lb/hr per pump seal for liquid and twice the equivalent liquid volume for gas or vapor. Such large amounts of leakage are expected to be rare, therefore, the actual function of MPC is expected to be a small fraction of the values given.

(2) Following partition factors are used:

Tritium	0.53 for hot liquid, 0.1 for cold liquid
Noble	1.0 of all
Iodine	0.1 for hot liquid (120°F), 0.001 for cold liquid (120°F)

(3) Annual average values are reported here. When the gas analyzer is processing gas from recycle evaporator vent condenser, they could exceed the given values temporarily.

TABLE 12.2-46 (Cont'd)

AREA	LEAK RATE (1) (gpm)	EXHAUST AIR FLOW RATE (cfm)	NUMBER OF MPC (2) TRITIUM	NOBLE	IODINE
El. 417'-0"					
Low Activity Spent Resin Tank	1.3-4	2750	-	-	-
Spent Resin Storage Tank	2.00-5	2450	-	-	6.96-2
El. 426'-0					
Laundry Room	3.70-4	3800	2.27-10	1.23-7	1.82-7
Volume Control Tank Valve Aisle	7.40-4	2900	1.96-3	1.01+0	1.17-1
Volume Control Tank Room	1.90-4	2900	2.43-3	1.29+0	1.45-1
Waste Gas Analyzer Rack Room(3)	1.40-4	200	1.17-4	5.97+0	2.15-1
Waste Gas Cabinet Aisle(3)	1.80-4	2500	1.26-5	2.15-1	2.29-2
Purge Room	6.20-4	2550	3.01-3	3.07-1	3.63-1
Waste Gas Compressor Room(3)	4.49-4	1000	1.97-4	2.49+0	3.44-1

(1) The leak rates given in the table are based on leakages of $5x10^{-3}$ lb/hr per valve or flange; $2x10^{-2}$ lb/hr per pump seal for liquid and twice the equivalent liquid volume for gas or vapor. Such large amounts of leakage are expected to be rare, therefore, the actual function of MPC is expected to be a small fraction of the values given.

(2) Following partition factors are used:

Tritium	0.53 for hot liquid, 0.1 for cold liquid
Noble	1.0 of all
Iodine	0.1 for hot liquid (120°F), 0.001 for cold liquid (120°F)

(3) Annual average values are reported here. When the gas analyzer is processing gas from recycle evaporator vent condenser, they could exceed the given values temporarily.

TABLE 12.2-47

CALCULATED AIRBORNE ACTIVITIES FOR DESIGN-BASIS

LEAK RATE IN CONTAINMENT BUILDING

AREA	TOTAL PRIMARY COOLANT LEAKAGE*	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC** IODINES	NOBLES	H-3
Containment free volume	50 lb/day	3000	1.61-1	1.30+0	1.72-2

* Estimate of reactor coolant leakage into the containment atmosphere from valve and pump seals as given in WCAP-8253.

** The partition factor for iodines in a hot liquid (>120°F) is 1 x 10^{-1}, H3 partition factor is 0.53 for primary coolant. Use of MPC is consistent with regulations that were in effect at the time of analysis.

TABLE 12.2-48

CALCULATED AIRBORNE ACTIVITIES FOR DESIGN-BASIS

LEAK RATE IN RADWASTE BUILDING

AREA	MAXIMUM LEAKAGE (gpm)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC*		
			IODINES	NOBLES	H-3
Radwaste Building general		8630	1.80-3 (Estimated)	1.50-1 (Estimated)	2.80-4 (Estimated)

* The partition factor for iodines in a hot liquid (>120°F) is 1 x 10^{-10}, in a cold liquid (<120°F) the partition factor is 1 x 10^{-3}; H3 partition factor is 0.53 for hot liquid and 1 x 10^{-1} for cold liquid. Use of MPC is consistent with regulations that were in effect at the time of analysis.

TABLE 12.2-49

TABLES 12.2-49 THROUGH 12.2-52 AND FIGURE 12.2-1 FOR THE VOLUME REDUCTION SYSTEM HAVE BEEN INTENTIONALLY DELETED.

TABLE 12.2-50

TABLES 12.2-49 THROUGH 12.2-52 AND FIGURE 12.2-1 FOR THE VOLUME REDUCTION SYSTEM HAVE BEEN INTENTIONALLY DELETED.

TABLE 12.2-51

TABLES 12.2-49 THROUGH 12.2-52 AND FIGURE 12.2-1 FOR THE VOLUME REDUCTION SYSTEM HAVE BEEN INTENTIONALLY DELETED.

TABLE 12.2-52

TABLES 12.2-49 THROUGH 12.2-52 AND FIGURE 12.2-1 FOR THE VOLUME REDUCTION SYSTEM HAVE BEEN INTENTIONALLY DELETED.

TABLE 12.2-53

ASSUMED DEMINERALIZER RESIN
INVENTORY IN LOW ACTIVITY SPENT RESIN TANK
FOR SHIELDING SOURCES CALCULATION

QUANTITY	COMPONENT NAME	VOLUME OF RESIN IN EACH (ft^3)	FRACTIONAL CONTRIBUTION TO LASRT INVENTORY
4	Blowdown Mixed Bed Demineralizers	113	0.84
3	Radwaste Mixed Bed Demineralizers	29	0.16

TABLE 12.2-54

Low Activity Spent Resin Tank Shielding Design-Basis

Radionuclide Content (in Curies)

ISOTOPE	ACTIVITY (CURIES)	ACTIVITY (CURIES)*
Cr-51	8.50-04	8.94-05
Mn-54	8.18-04	6.70-04
Mn-56	3.52-04	--
Co-58	2.57-02	1.06-02
Fe-59	1.04-03	2.57-04
Co-60	8.02-04	7.76-04
Br-84	1.12-03	--
Rb-88	7.72-03	--
Rb-89	2.25-04	--
Sr-89	3.21-02	9.35-03
Sr-90	1.76-03	1.75-03
Y-90	1.31-03	--
Sr-91	8.33-04	--
Sr-92	9.28-05	--
Y-92	9.28-05	--
Zr-95	6.90-04	2.60-04
Nb-95	7.22-04	1.22-04
Mo-99	1.11-02	--
I-131	4.06-01	1.73-04
Xe-131m	3.46-02	1.78-04
Te-132	8.65-01	--
I-132	9.62-01	--
I-133	3.73+00	--
Xe-133m	9.08-02	--
Xe-133	3.09+00	2.10-05
Te-134	9.78-04	--
I-134	2.60-02	--
Cs-134	6.28-04	5.78-04
I-135	6.80-01	--
Xe-135m	1.95-01	--
Xe-135	6.80-01	--
Cs-136	3.22-04	2.81-06
Cs-137	3.22-03	3.20-03
Cs-138	2.09-03	--
Ba-140	3.21-02	2.54-04
La-140	3.05-02	--
Ce-144	3.53-04	--
Pr-144	3.53-04	--

* neglected below 10-06 activity

TABLE 12.2-55

OLD STEAM GENERATOR STORAGE FACILITY SURVEYED DOSE RATES

	BYRON	BRAIDWOOD
Inside channel head (middle of tubesheet)	10 R/hr	11 R/hr
Inside tube region	NA	5 R/hr
Outside tube region	NA	40 mR/hr
Outside steam dome	1 mR/hr	2 mR/hr

Notes:

1. The dose rates represent the maximum surveyed dose rates inside and outside the steam generator regions with the steam generator drained.

2. Waste samples at Byron and Braidwood indicate that Co-58 and Co-60 are the dominant gamma-emitting isotopes.

12.3 RADIATION PROTECTION DESIGN FEATURES

Radiation protection design features are provided to reduce direct radiation, control airborne radioactivity, identify radiation areas, decontaminate personnel and equipment, calibrate radiation monitors, and maintain personnel radiation exposure as low as is reasonably achievable (ALARA). Illustrative examples of the application of various radiation protection design features, including several types of shielding to specific components, are provided in Attachment 12.3A.

12.3.1 Description of Facility Design Considerations

12.3.1.1 Equipment Selection, Layout, and Segregation

In selecting and shielding equipment and components containing radioactive materials, prime consideration is given to protecting the operating and maintenance personnel from radiation, and to maintain personnel exposures ALARA.

Equipment containing radioactive materials is located in separate rooms or cubicles, where practicable, to protect operating and maintenance personnel from radiation associated with other equipment. Components are remotely operated and/or remotely serviced whenever practicable.

Items which require frequent maintenance and which are radioactive or potentially radioactive, such as pumps, valves, and instrumentation are to the extent practicable, separated from passive radioactive components such as tanks, filters, demineralizers, etc.

Areas containing more than one piece of radioactive equipment are, where practicable, designed and provided with shielding such that maintenance of one item is not restricted by radiation from other pieces of equipment. Where it is not practicable to provide permanent shielding, provisions (discussed in Subsection 12.3.2) for temporary shielding to minimize maintenance doses are provided.

Components which are not radioactive or potentially radioactive are physically separated, to the extent practicable, from components which are radioactive or potentially radioactive.

Radiation detector probe access holes are provided in most shield walls (e.g., shield hatches) for isolated equipment cubicles where access is only by means of removable shield walls.

Partially shielded configurations are reviewed for radiation scattering.

12.3.1.2 Cubicle Access

Access to radioactive or potentially radioactive cubicles or compartments is through entrances designed, where practicable, to permit access to an area of the room which has the lowest or relatively lowest radiation level. Entrances are designed to prevent source radiation from passing directly through entrance openings and into occupied areas. This is done, where practicable, by providing labyrinthine entrances to radioactive and potentially radioactive cubicles.

Typical labyrinthine entrances are shown in Figures 12.3-1 and 12.3-2. Radiation traveling through such labyrinthine entrances collides with the shield walls and consequently can be attenuated to some small fraction of the incident quantity.

Cubicle access for Byron/Braidwood Stations is either through a labyrinthine entrance with an overlap of 1 to 1-1/2 times the passageway width as seen in Figure 12.3-1 or through a double labyrinth arrangement as seen in Figure 12.3-2.

Not all entrances to radioactive areas are designed with labyrinthine entrances. Where labyrinthine entrances are not feasible, other alternatives include:

a. shield doors installed at personnel entrances,

b. removable concrete block walls, and

c. wall and floor removable shield hatches and plugs (such as for the radwaste filter and demineralizer compartments).

The following considerations govern the design of labyrinths:

a. A labyrinth is located and sized to cause unscattered radiation to be attenuated by the required amount of shielding, as shown in Figure 12.3-1. Normally, the labyrinth overlap is designed so that (with worst-case sources) the streaming leaving the labyrinthine entrance due to scattered radiation gives a dose rate which is less than three to five times the design dose rate of the surrounding area. Where strong sources of low energy gamma radiation are encountered, a double labyrinthine entrance such as depicted in Figure 12.3-2 is used in order to meet this criterion.

b. When the design of the labyrinth is determined by other design considerations, a shield door, isolation of the entrance (e.g., rope off area), extended labyrinth overlap, or a removable labyrinth is also specified.

c. If the labyrinth height is shorter than the ceiling height, as is often the case, a roof is provided above the labyrinth section.

d. Galleries and other elevated occupied areas are protected from radiation passing through the roof of the labyrinth. The roofs have a thickness which maintain the design dose rate of these elevated areas.

e. Labyrinths inside source cubicles require roofs if any part of a source is higher than the top of the labyrinth. The roof thickness is dependent upon the location of the source, and the thickness is calculated on a cubicle-by-cubicle basis.

12.3.1.3 Draining and Flushing Capability of Equipment

Consideration is given in the radiation protection design to identifying the need for adequate draining and flushing capability of equipment designed for radioactive or potentially radioactive service.

The potentially high activity radwaste storage tanks were selected and their designs reviewed to assure adequate draining capability to minimize activity buildup and excessive radiation levels over the plant lifetime. Tanks containing radioactive material have sloped bottoms wherever practicable so that sludge accumulation is minimized and ease of drainage is enhanced.

Where practicable, equipment is selected and the design reviewed to assure that there are no obvious ledges or pockets where radioactivity may be trapped or accumulated.

To the extent practicable, drain piping is of welded construction and is welded in a manner, e.g., using consumable inserts, to minimize crevices which might collect radioactive material. (Use of backing rings in the welds or use of socket welds may be acceptable if the weld is embedded in concrete.)

The design of the spent resin storage and exchange systems is reviewed to assure that the layout and components are such as to prevent the retention of resin beads or fragments in connections, bends, horizontal sections, reducers, etc.

All equipment drains which are considered to be radioactive are directed to appropriate liquid radwaste storage tanks. Sumps are used as intermediate collection points. Such sumps and tanks are appropriately shielded or appropriately located within radiation areas.

The design of the radwaste filters was checked to assure that the filters can be drained and flushed prior to filter element replacement.

Flushing capability of radioactive service equipment is important to assure a minimum of radioactive crud or sludge retention in the equipment prior to maintenance or removal of the equipment.

All potentially high activity source storage vessels were selected and their designs checked to assure adequate draining capability. These tanks include the volume control tank, the spent resin storage tank, the concentrates holding tank, the chemical/regeneration waste drain tank, the auxiliary building floor and equipment drain tanks, and the recycle holdup tanks.

Draining capability is assured:

a. to minimize personnel exposure during testing, surveillance, and maintenance activities and

b. to minimize activity (crud) buildup and avoid excessive radiation levels to accessible areas during plant lifetime.

Adequate draining capability is assured wherever practicable by selecting tanks which have sloped bottoms and which have, or can be provided with, drain lines connected to the lowest level of the tanks. Drainage of the above listed high activity source storage tanks is via remotely operated valves or by valves which are located remotely from the tank cubicle in lower radiation areas. (For location of valves with respect to shielded areas, refer to Subsection 12.3.1.8.)

Flushing of radwaste tanks is accomplished by washing down the tank interiors with demineralized water and/or cleaning agents. Where practicable, provisions are made to remove crud sedimentation by remote mechanical means with hoses.

Where practicable, flushing of radwaste tank interior is accomplished by an installed sparger (where justified) or by providing a recirculation line for the pump servicing the tank to the bottom of the tank so that a spraying effect can be utilized to get settled deposits in suspension, so that they may be pumped or drained out of the tank. For manual flushing, adequate capability is provided in the form of water connections located near the tank cubicles.

Flushing is required when major maintenance and/or removal of the tank is necessary and also when necessary to reduce radiation levels in adjacent areas due to sources within the tank. Flushed water is directed to tanks having sufficient capacity and shielding necessary to contain and shield the flushed water.

When practicable, the above applies to other high activity source items such as pumps. Where adequate draining and flushing capability is not practicable, shielding is designed to account for worst-case radioactive crud buildup.

12.3.1.4 Floor and Sink Drains

Adequate floor drainage is provided for each room or cubicle housing components which contain, or may contain, radioactive liquids. Floors are properly sloped to the floor drain to facilitate floor drainage and prevent water puddles.

All floor drains which are considered to be radioactive are directed to appropriate liquid radwaste storage tanks. Sumps are used as intermediate collection points. Such sumps and tanks are appropriately shielded or appropriately located within radiation areas. Shielding of radwaste drain piping is discussed in Subsection 12.3.1.6.

To the extent practicable, greater potential radiation area floor drains are segregated from lesser potential radiation area floor drains to protect against backflow of radioactive liquids into lower potential radiation areas, if drainage is blocked or if a large spill occurs. Air circulation through the floor drain system is prevented by the use of water-filled seals (loop seals) or by sealing individual floor drains. The use of such seals also prevents backflow of radioactive gases into the room from the floor drain system.

Sink drains which are expected to contain radioactive fluids are reviewed for appropriate shielding and routing requirements.

Loop seals are present on sink drain lines which may handle radioactive fluids.

All floor drains in the auxiliary, containment, fuel handling, and radwaste/service buildings, except for those areas listed below, are considered to be radioactive and shall discharge to either the auxiliary building floor drain tanks or the chemical drain tank through various sump pumps. Exceptions to this requirement are:

a. diesel-generator oil storage tank rooms,

b. auxiliary feedwater tunnel,

c. main steam/steam generator feedwater tunnel,

d. tendon tunnel,

e. tendon tunnel access area,

f. diesel-generator rooms,

g. cable spreading rooms,

h. switchgear rooms,

i. office areas in service building,

j. storage rooms in service building,

k. auxiliary electrical equipment room,

l. battery rooms,

m. auxiliary building HVAC equipment area (elevation 451 feet), and

n. essential service water pump rooms.

o. auxiliary building HVAC chilled water coil areas (Byron only) on elevation 451 feet.

p. auxiliary building chiller "A" area on elevation 463 feet.

12.3.1.4.1 Design of Drain System

a. Equipment drains in the turbine building discharge to the two turbine building equipment drain sumps, one per unit, from which they are piped to the turbine building equipment drain tank. At Byron, the drains are treated by the wastewater treatment system and discharged to the circulating water system (CW) flume or to the release tank 0WX26T. At Braidwood, the drains are treated by the wastewater treatment system and discharged to the cooling pond.

b. Equipment drains in the auxiliary, containment, and fuel handling buildings discharge to the two auxiliary building equipment drain collection tanks. Pumps are provided to pump the drains to the auxiliary building equipment drain tanks.

c. Floor drains that are expected to handle chemical waste solutions from potentially radioactive areas are kept separate from other floor drains and are routed to the chemical drain tank, unless otherwise specified.

d. Leak detection sumps are provided for various areas in the auxiliary building that contain safety-related equipment required for long-term operation.

e. A storm drain system, complete with oil separators, is provided to remove all roof and storm drainage.

f. Borated equipment drains are recycled to the recycle holdup tanks.

g. High radiation area floor drains are routed separately from low radiation area floor drains to prevent backflow of high contamination into low radiation areas.

h. The top elevation of floor drains are set below nominal elevations of the floor area to be drained.

i. Floors are sloped to the drain to facilitate floor drainage and prevent water puddles.

j. Slotted cover plates are used to prevent solids from entering floor drain sumps. These cover plates are removable to provide full access to the sump.

k. The arrangement of drains from cubicles containing radioactive equipment is such that air from a zone of high airborne radioactivity potential does not circulate through the drain system to normally accessible areas. The prevention of air circulation is done through the use of loop seals.

l. Drain piping of equipment and systems which carry caustics or acids is the same material as the equipment or system they are draining.

m. Drain lines are sloped 1/8-inch per foot to assure complete drainage of piping. An exception to this is in containment where drain lines may not be sloped 1/8-inch per foot. This does not adversely impact operation of the containment floor drain system, which will continue to function as designed.

n. Drain piping is of welded construction and is welded in a manner to avoid crevices (except where embedded in concrete), which might collect radioactive solids. All potentially high radioactive drain piping from the equipment to the loop seal is welded using a consumable insert.

o. Equipment drains which interconnect pieces of equipment are designed so as not to inadvertently transfer fluid from one piece of equipment to another.

p. Shielding of radwaste drain piping is provided as necessary. Radwaste drain piping not specifically shielded is routed so that it is not exposed to normally high access areas and general access routes. Vertical runs of radwaste drain piping not specifically shielded is run against walls and sufficiently isolated so as to facilitate the installation of compensatory shielding, if required.

q. All floor drain piping to the sumps (unless otherwise noted) is carbon steel unless required to be otherwise by design due to flow of corrosive liquids.

r. Primary sample drains are routed to the chemical drain tank and from there processed in the radwaste evaporators (Braidwood only).

12.3.1.5 Venting of Equipment

Where practicable, all radioactive or potentially radioactive equipment (such as filters, demineralizers, and radwaste tanks) is vented to a filtered vent header to minimize the possibility of airborne radioactivity in occupied areas or equipment cubicles due to equipment venting.

Radwaste sumps (i.e., sumps designed to handle drains from radioactive service equipment or from floor areas of potentially radioactive components) are normally either vented to a high radiation area, such as to within the cubicle the sump is located, if it is high radiation cubicle, or to a filtered vent header. Venting of radwaste sumps is important to control the concentrations of radioactive contaminants normally released to the air from potentially contaminated water held in the sumps. Subsection 12.3.1.5.1 discusses the sumps with venting. For sumps which are in shielded cubicles and which vent to the cubicle, cubicle ventilation rates are such as to assure adequate control over expected airborne concentrations of radioiodine. If venting to other areas is required, the sump covers have air inleakage and have no special provisions for sealing since the sump can maintain a slightly negative pressure with respect to the area in which the sump is located. A small amount of air inleakage to the sump is desirable to maintain air flow through the vent line.

12.3.1.5.1 Sumps Requiring Venting

Venting is provided for the auxiliary building equipment drain collection sumps.

Venting of these sumps minimizes the possibility of potential airborne radioactivity in the sump areas. Venting is via a small vent line connected to the sump cover plates. This line is routed to a filtered vent header. Slightly negative pressure is maintained in the vent line with respect to the area in which the sump is located.

12.3.1.6 Routing and Shielding of Lines and Ventilation Ducts

12.3.1.6.1 Routing and Shielding of Lines

All potentially radioactive process lines are evaluated to determine proper routing and shielding requirements, based on minimizing radiation exposures to station operating and maintenance personnel.

Radioactive process piping is routed in shielding pipe tunnels, trenches, or chases, or in areas where the radiation field due to the pipe is consistent with the radiation zone for that area.

To aid in preventing crud buildup in process piping, sharp bends, dead ends, and other obvious crud traps are minimized. In general, socket welds and welds employing backing rings are avoided to the extent practicable; these welds contribute to radioactive crud accumulation which results in increased radiation fields near the weld. Where practicable, welds employing consumable inserts are used instead of socket welds or welds using backing rings because the consumable insert weld makes the inside-of-pipe surface smoother and minimizes crevices which may trap crud at the weld. Socket welds and welds employing backing rings are used, however, if the weld is to be embedded in concrete (such as in concrete floor slabs); for these cases, radiation fields due to radioactive crud accumulation are attenuated by the concrete around the weld.

Shielding of radwaste drain piping (including floor and sink drain piping) is provided as necessary. Radwaste drain piping not specifically shielded is routed so that it is not exposed to normally occupied areas and general access routes. Vertical runs of radwaste drain piping not specifically shielded are run against walls and sufficiently isolated so as to facilitate compensatory shielding, if required.

To the extent practicable, radioactive or potentially radioactive sample lines used for grab samples are routed so that grab samples can be taken in low radiation areas.

Radioactive lines are process system piping, drain lines, sample lines, and other lines which normally do, or may contain, radioactive fluids. Special attention is given to the routing and shielding of radioactive lines.

The following guidelines are followed for routing and shielding of radioactive lines:

a. Routing of radioactive lines in low radiation zones is avoided to the extent practicable.

b. Lines that require shielding are routed in shielded pipe tunnels or in radiation areas to the extent practicable.

c. Penetrations through shielded pipe tunnels are not made by lines which do not, themselves, run through the pipe tunnels.

d. Lines that carry radwaste demineralizer resins, filter backwash, filter/demineralizer sludges, or other particulates have large radius bends and are continuously sloped. On radwaste demineralizer resin lines, welded piping is used but the use of socket welds or welds employing backing rings is avoided to the extent practicable; also, the use of loop seals on these lines is avoided to the extent practicable.

e. Slightly radioactive lines are routed in a manner which minimizes radiation exposure to plant operating and maintenance personnel. Slightly radioactive lines in low radiation zones are, to the extent practicable, routed at a minimum elevation above the finished floor of 10 feet 0 inch, or as high above the floor as is practicable. To the extent practicable, slightly radioactive lines are not routed near normally traveled passageways, nor near galleries or other elevated work areas.

f. For field routing of 2-inch and under nonseismic radioactive piping, the guidelines listed below are followed.

1. Piping is installed at as high an elevation as is practicable but, in no case, below 10 feet 0 inch from the finished floor level in general access areas, nonsource cubicles, and hallways.

2. Piping is routed as close as possible to existing walls or structures to take advantage of their shielding effect.

3. Radioactive piping is not routed near groups of nonradioactive piping thereby not limiting accessibility to nonradioactive system components.

4. Radioactive piping is not routed near an area radiation monitor thereby causing abnormally high radiation readings which are nonrepresentative of the general area in which the radiation monitor is located.

5. To aid in preventing radioactive crud buildup in the piping, sharp bends, dead ends, and other obvious crud traps are avoided to the extent practicable. The use of socket welds or welds employing backing rings on the piping is avoided to the extent practicable.

12.3.1.6.2 Routing and Shielding of Ventilation Ducts

HVAC duct routing was reviewed to assure that air flow is from areas of lower potential radiation contamination to areas of higher potential radiation contamination.

Ventilation duct penetrations of shield walls, floors, and ceilings are evaluated to determine if parapet and labyrinthine shielding around the ducts is necessary. Penetrations in shield walls for HVAC ducts is discussed further in Subsection 12.3.2.3.

12.3.1.7 Waste Filters and Demineralizers

The waste filters and demineralizers which accumulate radioactivity and which, if unshielded, could cause the area design dose rate to be exceeded, are located, to the extent practicable, in separately shielded cubicles. Shielding is provided between such adjacent filters and demineralizers to minimize personnel exposure during removal or maintenance operations.

A radiation detector probe access hole is provided in most of the filter and demineralizer removable shield hatches so that radiation levels of the contained equipment may be measured without removing the shield hatches. Figure 12.3-4 shows a typical probe access hole.

The waste filters are designed where practicable to permit removal by a remote handling device. Draining and flushing of radwaste filters is discussed in Subsection 12.3.1.3.

Waste filters also include HVAC filters which may accumulate airborne radioactive materials. These filters are located in areas of the station where access is controlled. Shielding is provided as necessary around HVAC filters (e.g., charcoal filters) to ensure that resultant dose rates from the filter areas are less than the design dose rates for the areas, and to minimize radiation exposure to maintenance personnel during filter removal or maintenance.

HVAC filters are designed for easy removal and sized to allow proper disposal as per Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for ESF Atmosphere Cleanup System Air Filtration and Adsorption Units of LWRs," Revision 2.

For charcoal air filters, charcoal filtration capacities are such as to assure that radioiodine loadings meet criteria for ESF atmospheric cleanup system air filtration and adsorption units.

12.3.1.8 Valves and Instruments

Where practicable, valves are located and shielded from adjacent radiation sources so that they can be operated or serviced without causing excessive exposure to operating or maintenance personnel.

Shielded valve aisles are provided where necessary to allow greater accessibility to frequently operated or maintained valves. The valves are installed in the valve aisle shielded from the equipment they serve. Whether the valves are remotely operated or hand operated, the valves and associated piping are shielded from the valve operating area.

12.3.1.8.1 Valves

a. To extent practicable, all valves servicing radioactive or potentially radioactive equipment are located in shielded valve aisles, apart from the (adjacent) equipment being serviced. Walk-in valve aisles are used where practicable (see Figure 12.3-3). Locating valves in pipe tunnels cannot be avoided entirely, however.

b. All radioactive or potentially radioactive manually operated valves (and associated piping) are shielded from the valve operating area, to the extent practicable. Where practicable, use is made of remote manual valve operators (valve extensions or reach rods) connected to the manual operated handwheels or geared handwheels and passing through the shielding to allow valve operation in the valve operating area (see Figure 12.3-3). This protects valve operating personnel from radiation due to radioactivity in the valves and associated fluid piping in the valve aisle.

c. Radioactive pipe runs to and from valves located in valve aisles are minimized to reduce the amount of radioactive material in valve aisles. This is done by maximizing the amount of radioactive runs behind shielding (e.g., running as much of the radioactive pipe behind the shield wall which separates the valve aisle from the [adjacent] equipment compartment of the component which the valve services).

d. To the extent practicable, all motor-operated valves and pneumatic operated valves (air-operated valves) which are in radioactive or potentially radioactive service are located in areas which are shielded from the (adjacent) component or item of equipment which the valves service. Locating these valves (which are typically higher maintenance items than manual operated valves) in shielded areas minimizes potential personnel radiation exposures due to other nearby radiation sources during valve maintenance and inservice inspection.

e. Valves servicing radioactive or potentially radioactive equipment are installed and positioned with respect to other valves so that (1) service or maintenance time is minimized, and (2) compensatory shielding (e.g., lead blankets) is used, where practicable, to protect workers from adjacent radioactive valves and piping.

f. For valve maintenance, provision is made for draining or flushing the valve and associated connecting lines of radioactive fluids so that radiation exposures are minimized.

Figure 12.3-3 shows a typical walk-in valve aisle arrangement for Byron/Braidwood Stations.

12.3.1.8.2 <u>Instruments</u>

a. Output devices such as instrument readouts, pressure switches, electrical bistable devices, electric converters, control devices, etc., are located and positioned in areas (e.g., at valve operating stations) which result in the lowest personnel exposures, consistent with other requirements such as instrument accuracy and precision. Use of transducers is maximized in high radiation areas.

b. The following is considered in the location and positioning of the instrument readout devices to assure ALARA exposures.

1. Locate in readily accessible areas.

2. Position at convenient elevation for observation and application of parallax corrective devices.

3. Face readout toward direction convenient for reading.

4. Provide easily readable numbers and easily observable pointers and needles.

5. Preclude or minimize application of scale multipliers on readout.

6. Locate to take advantage of amount of lighting available.

7. Locate instruments and instrument readouts away from local hot spots caused by streaming radiation or from the accumulation of radioactivity in lines, ducts, filters, and equipment.

c. Wherever practicable, radiation monitoring equipment with remote readout is located in areas to which personnel normally have access.

12.3.1.9 Contamination Control and Decontamination

In addition to the safety design features discussed above, the following safety design features specifically relating to decontamination and contamination control are incorporated into the radiation protection design of the station.

a. Curbs

Where practicable and where failure of radioactive storage tanks, vessels, or associated piping is postulated, either the floor of the cubicle is situated at an elevation lower than the entrance to the cubicle or curb walls are provided to restrict radioactive material to the cubicle.

Curbs are provided for equipment decontamination pads to restrict washdown water to the pad and avoid contamination of adjacent areas.

b. Protective Surface Coatings

Wherever there exists a potential for leakage or spillage of radioactive material onto concrete surfaces (e.g., shield walls, floors, or ceilings), such surfaces are coated with a nonporous coating to enhance decontamination.

The following guidelines and criteria are used for the application of coating systems to potentially contaminated concrete surfaces in the station to enable them to be effectively decontaminated.

The function of the protective coating system is to facilitate decontamination of surfaces by providing a clean, smooth, and hard finish that is minimally free of cracks, is nonabsorbent, and is water-repellent. Surface contaminants can then be removed by means of washing, sweeping, scrubbing, or wiping in one or more applications.

a. The coating systems are capable of performing their surface protective functions throughout the 40-year plant lifetime (including reasonable maintenance and touch-up activities) and under the variable radiation source and environmental conditions anticipated for the plant.

b. The coating systems applied to floors, curbs, dado, and wainscot are capable of maintaining their integrity in protecting these surfaces under conditions of water immersion. The coating systems used on floors, curbs, and dado is therefore solvent-based. The wainscot can be either solvent or water-based.

c. To enable the coating systems to perform their intended function, a surface preparation system appropriate to the surface as well as to each coating system, is first applied. The surface preparation system includes surface cleaning, the filling of holes and the application of primer coating.

d. The coating systems used on floors and ramps are capable of maintaining their integrity in protecting these surfaces under the traffic patterns (people, lift trucks, etc.) anticipated in the various areas. The thickest of the field coating systems should be specified for such areas that involve continuous use to avoid deteriorating and thereby compromising the coating.

Protective coating systems are applied to concrete surfaces on the following basis:

a. Where no other requirements are necessitated, all walls are coated to 1 foot-0 inch dado height to protect this lowest section during sweeping and washing of the floor.

b. Walls that require only partial height coverage are coated to one of several standard wainscot heights (usually 5 feet-0 inch or 8 feet-0 inch). General examples include the walls of potentially radioactive heat exchangers, and certain access area locations.

c. Cubicles containing radioactive equipment that reach above the highest wainscot level are coated to full height and in most cases, the ceiling. The coating of rooms utilizing monorail or crane systems for handling radioactive materials are based on the elevated height of the materials.

d. Cubicles containing radioactive processing equipment such as pumps or pressurized pipe with valving are coated to full height, where necessary. Potentially radioactive contaminated water can come from the room (or area) on the floor above, through penetrations in the ceiling.

e. Walls are coated to full height if the potential exists for leakage of radioactive contaminated water from the room or area on the floor above, through penetrations in the ceiling.

f. Cubicles, rooms, and areas that require complete wall coverage have their ceiling fully coated as well. This includes the underside of removable shield hatches and plugs as well as fixed ceilings.

g. Pipe tunnels that are accessible and contain radioactive pipes are fully coated.

Areas that require partial wall coverage (although complete wall coverage may be dictated by equipment size) include the following:

a. areas around sampling stations or panels receiving radioactive process streams for monitoring;

b. areas through which heavy traffic patterns are expected; and

c. clothing change areas, personnel monitoring points, and counting room.

12.3.1.9.1 <u>Equipment Decontamination Facilities</u>

Equipment decontamination facilities are provided in the station as required for the decontamination of contaminated equipment, tools, etc. The design of these facilities includes adequate shielding, and ventilation and filtration of the room air.

A separate area, the equipment decontamination facility, is provided on elevation 346 feet 0 inch in the auxiliary building for cleaning, and decontaminating tools and small pieces of equipment.

12.3.1.9.2 Personnel Decontamination Facilities

A personnel decontamination facility is supplied on elevation 426 feet 0 inch in the auxiliary building to provide for prompt decontamination of plant personnel, if the need should arise.

12.3.1.9.3 Station Decontamination

Radiation decontamination of the station is currently expected to be required at least once during the life of the station. The radiation protection safety design features discussed above assure less complicated station radiation decontamination when it is required.

12.3.1.10 Traffic Patterns and Access Control Points

Traffic patterns are established to maintain occupational radiation exposures ALARA. Anticipated traffic patterns have been used to determine design dose rates in the various areas, and thus have affected the determination of radiation zones.

The majority of normal personnel traffic occurs between the service building and the auxiliary building. The remainder of the traffic occurs in operating areas (where panels and motor-control centers are located), hallways, elevators, and stairwells.

Access control points (i.e., check points for personnel) are flexible and are determined on a day-to-day basis, depending on contamination levels and maintenance activities.

12.3.1.11 Radiation Zones

Radiation zones have been defined as a means of classifying the occupancy restrictions on various areas within the plant boundary. The design criteria for each zone are described in the subsections which follow and are tabulated in Table 12.3-1. The radiation zones assigned to the areas of the plant, and upon which the shielding has been designed, are shown in the radiation zone maps in Figures 12.3-27 through 12.3-71.

Selection of appropriate design dose rates for particular plant areas is one method of maintaining individual doses within regulatory limits. Maintaining total collective exposure ALARA has been considered in plant layout and zoning designations. The plant design has an abundance of general access areas. These areas are designed so that 100% occupancy in these areas results in a total annual dose which is far below the regulatory limits. The general access areas are an integral part of the ALARA concept of exposure control. These areas are used to travel from one part of the station to almost every other part of the station. If it becomes necessary, a limited amount of maintenance can be performed in some sections of the general access areas, but such sections are used only when reductions in exposures result.

The zone designations are only a tool to aid administrative controls. The zones given in Table 12.3-1 are based on design-basis radiation sources. The actual maximum dose rates for the zones that are less than or equal to 100 mrem/hr are expected to be a small fraction of the given dose rates. A more precise zoning is obtained by using the data from periodic radiation surveys. The dose rates were determined using limits that were in effect at the time when the zones were designated.

Zoning decisions for radiation areas during operation are determined by the residual radioactivity of the equipment (due to plateout and crud buildup) and the activity of material which may be present in the equipment. The shutdown condition for the same areas has a much lower level of radiation because only the residual activity is present. The majority of the occupational radiation exposure (~80%) is accumulated while performing surveys, inspections, and maintenance during operation and/or shutdown conditions.

The total person-rem exposure during surveys and inspection is kept ALARA administratively. The administrative controls are described in Subsection 12.5.3.1. The station design aids the administrative controls by segregating equipment with shielding, which allows high maintenance items to be located in ALARA radiation environments. Thus, the background radiation due to one type of equipment on a second type is kept to a small fraction of the residual radiation of the second type. Since two or more of the same type of equipment can be in the same area, administrative controls will determine the maintenance procedure necessary to keep radiation exposure ALARA.

<u>Zone I-A</u>

Zone I-A has no restriction on occupancy. A I-A area would represent, for example, plant site where radiation due to occupancy on a 40 hr/week, 50 week/yr basis, will not exceed the whole body dose of 0.5 rem/yr. The environs around the plant such as the pump house, electrical switchyards, and turbine hall, are examples of a Zone I-A area.

It is expected that nonplant personnel or visitors to the site will receive considerably less than 0.5 rem/yr because of the relatively small time interval during which they are on the site.

Zones I-B, I-C, and I-D

Zones I-B, I-C, and I-D are areas which individuals can occupy on a 40 hr/week, 50 week/yr basis, and not exceed a whole body dose of 1.25 rems per calendar quarter. The design dose rates are from 0.5 to 2 mrem/hr in these zones. The area will remain accessible. Corridors in the auxiliary building and areas outside radioactive enclosures where personnel can walk freely are included in this zone.

Zone II-A

Zone II-A is a radiation area that plant personnel can occupy periodically. This zone has a design dose rate of 4 mrem/hr. The radiation level in a Zone II-A area will be posted, but the area will remain accessible to the plant personnel.

Zone II-B, II-C, and II-D

Zones II-B, II-C, and II-D are areas where dose rates are from greater than 4 mR/hr to 100 mR/hr. Occupancy is limited. The time a worker with a permit can stay in this room is determined by four factors:

a. the actual radiation level in the room;

b. the nature of the radioactivity (airborne, gamma, etc.);

c. the past radiation history of the worker; and

d. nature of the required job.

The "nature of the required job" means that the necessity of the job being done to ensure the safe operation of the station will be considered when work in these radiation areas is being planned.

Auxiliary equipment which requires manual operation or inspection or maintenance during unit operation will not be located in these zones.

All equipment in areas designated as Zone I-A, I-B, I-C, I-D, or II-A will not contain radioactive materials, or if it does, the activity will be such that the dose rate outside the equipment is consistent with the design dose rate in the area. Such equipment could include fluid system, monitor tanks, and monitor pumps.

Zone III

Zone III represents areas where design dose rates are in excess of 100 mR/hr and occupancy periods are limited.

Zone IV

This zone is not assigned at Byron/Braidwood.

Zone V

Zone V is the main control room area. Zone V normal dose rate will be less than or equal to 0.2 mR/hr during normal operations. During an accident, the integrated whole body dose will not exceed 5 rem.

12.3.1.12 Laboratory Complex

The station laboratory complex is located in a controlled access area on the mezzanine floor of the auxiliary building. The facilities located in this complex are: a high level laboratory, a low level laboratory, a counting room, mask cleaning room (Byron only), instrument storage room (Braidwood only), personnel decontamination room, chemistry offices, and supervisor offices. This complex serves as a center for the chemistry activities at the station.

12.3.1.12.1 High Level Laboratory

The high level laboratory is designed to provide for the safe and efficient processing and analysis of radioactive and potentially radioactive samples. Such samples may be expected for such systems as the: primary coolant loop, chemical and volume control, fuel handling and storage, steam generator blowdown, and radwaste.

The major facilities provided in this laboratory are: fume hoods (with HEPA filtered exhausts), sinks (with drains routed to the liquid radwaste system), sufficient workbench space to allow frequently used equipment to be left in place, sufficient built-in storage space to assure a safe, uncluttered work environment, computer grade regulated electrical circuits, and a close tolerance HVAC system (temperature and humidity) to assure optimal performance of sensitive laboratory equipment.

To minimize the accumulation and spread of surface contamination; floor coatings, surface coatings, workbench surfaces, fume hood interiors, and sink and drain pipe materials are chosen to minimize the adherence and ease the removal of contamination. To minimize the spread of airborne radioactivity, fume hoods are provided for the storage and processing of volatile radioactive samples, and the high level laboratory is kept at a negative pressure with respect to the adjacent low level laboratory, counting room and laboratory corridor. All air exhaust from this laboratory is filtered prior to its release to the environment via the station vent stack.

12.3.1.12.2 Low Level Laboratory

The low level laboratory is located adjacent to the high level laboratory on the mezzanine floor of the auxiliary building. It is designed to provide a radiation and contamination free environment for the chemical preparation and analysis of nonradioactive samples (i.e., those samples which could not pose a radiological danger to the laboratory workers). The major equipment provided in the low level laboratory includes: fume hoods, sinks, workbenches, and storage facilities.

12.3.1.12.3 Counting Room

The counting room is located near the high and the low level laboratories on the mezzanine floor of the auxiliary building. This room is provided with computer grade regulated electric circuits and nonfluorescent lighting to assure the optimal performance of the counting equipment. The desired radiation level in the counting room should be below background. To assure that the counting room will not be affected by any in-plant airborne radioactivity the room is maintained at a positive pressure with respect to all surrounding areas and is ventilated with fresh filtered and conditioned air. The room HVAC is designed to maintain the temperature and humidity tolerances required by the detectors and their associated electronics and computer equipment. The use of thick concrete walls for shielding the counting room was precluded due to considerations of natural radiation emanating from the concrete itself.

The original equipment provided in the counting room includes:

a. gamma-ray spectrometer subsystem,

b. multichannel analyzer subsystem,

c. data analysis subsystem,

d. standard alpha, beta counting subsystem,

e. low-background alpha, beta counting subsystem,

f. automatic sample changer (for d and e), and

g. liquid scintillation counting system.

Localized radiation shielding is provided for the counting equipment when needed.

12.3.1.12.4 Chemistry Storage

A chemistry storage room is located in the general area of the low level chemistry room and the counting room on the mezzanine

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floor of the auxiliary building. It provides storage space for the chemistry supplies used within the laboratory complex.

12.3.1.12.5 Mask Cleaning Room

The laboratory complex includes a mask cleaning room. The room includes space and equipment for collecting, cleaning, inspecting, and storing respiratory protective equipment.

12.3.1.12.6 Personnel Decontamination Room

The decontamination room associated with the complex is designed to facilitate the decontamination of station personnel. A shower and sink are provided.

12.3.1.12.7 Office Space

The chemistry office and the supervisor offices in the laboratory complex are provided to assure adequate, local office space for the laboratory complex workers.

12.3.1.13 Laundry Facility

The station laundry facility is located on the mezzanine floor of the auxiliary building. It is designed to receive, store, and distribute the radiological protective clothing used in-plant. The floor and surface coatings in the laundry have been chosen to minimize the buildup and ease the removal of surface contamination. The laundry room is kept at negative pressure with respect to all surrounding areas to minimize the spread of airborne contamination originating from the handling of contaminated equipment.

floor of the auxiliary building. It provides storage space for the chemistry supplies used within the laboratory complex.

12.3.1.12.5 Instrument Storage Room

An instrument storage room is located within the laboratory complex.

12.3.1.12.6 Personnel Decontamination Room

The decontamination room associated with the complex is designed to facilitate the decontamination of station personnel. A shower and sink are provided.

12.3.1.12.7 Office Space

The chemistry office and the supervisor offices in the laboratory complex are provided to assure adequate, local office space for the laboratory complex workers.

12.3.1.13 Laundry Facility

The station laundry room is located on the mezzanine floor of the auxiliary building. It is designed for storage of Radiation Protection equipment and supplies, sorting of low level radioactive trash, and occasional laundering of contaminated personal clothing. The floor and surface coatings in the laundry room have been chosen to minimize the buildup and ease the removal of surface contamination. The laundry room is kept at negative pressure with respect to all surrounding areas to minimize the spread of airborne contamination originating from the handling of contaminated equipment.

The radiation zone map in Figure 12.3-31 shows the laundry room to be in a zone of less than or equal to 4 mrem/hr. This is under the most extreme conditions. The laundry room dose rates should average less than 1 mrem/hr during normal operation.

12.3.1.14 Survey Instrument Calibration Room

The instrument calibration room located on the Unit 1 side of the auxiliary building on the 401-foot level is designed to provide a location where radiation protection instrumentation can be calibrated, stored, serviced, and decontaminated when necessary.

12.3.1.15 Locker Room Facilities

Change areas are provided in the plant as necessary for individuals to don protective clothing for work in contaminated areas. Storage of personnel clothing is provided at these locations or other designated areas.

12.3.1.16 Design Features to Assist Decommissioning

The radiation protection design features established for station operation will aid in maintaining occupational radiation exposure ALARA during decommissioning. The shielding design allows for efficient mothballing and entombment. Decommissioning by removal of all contaminated and activated equipment will be aided by remote handling of equipment, equipment layout (Subsection 12.3.1.1), and administrative planning, which includes the health physics program (Section 12.5).

Specifications and limitations on cobalt and nickel content in equipment components will serve to limit radiation doses from the buildup, transport, and deposition of activated corrosion products in reactor coolant and auxiliary systems during both operation and subsequent decommissioning. A summary of the features in Westinghouse PWRs that reduce occupational exposure are given in Reference 9. Information on the steps taken to minimize Co-58 and Co-60 is given in Chapter 6 of Reference 9.

Radiocobalt and crud buildup in the primary coolant above 250°F are controlled below specification limits by continuous monitoring and controlling of the oxygen concentration. Hydrazine additions to the primary coolant and a hydrogen or nitrogen blanket in the volume control tank are the means of oxygen control. Control of pH in the primary coolant is accomplished by lithium hydroxide addition and is maintained between a pH of 4.5

and 10.5 depending on the disassociation of the boric acid present in the primary coolant.

The National Environmental Studies Project of the Atomic Industrial Forum has analyzed the decommissioning alternatives for LWRs in Reference 10. The majority of the estimated PWR occupational radiation exposure due to removal/dismantling comes from decontaminating the primary and radwaste systems. Experience gained through decontamination of Exelon Generation Company stations will be applied to decommissioning, which should produce additional dose saving procedures. Estimated occupational radiation exposures from the study are given in Table 12.3-7. The dominant radioactive isotopes that are expected to be found during decommissioning are given in Table 12.3-8.

12.3.1.17 Old Steam Generator Storage Facility Features

Four old Unit 1 steam generators are stored in the old steam generator storage facility (OSGSF). The OSGSF has an 18-inch concrete roof and 30-inch concrete walls. A vestibule, which contains a lockable, personnel-access door, is designed to minimize radiation streaming beyond the outside surface of the OSGSF. The OSGSF has a water collection sump. The sump access and monitoring port are located within the vestibule and are designed to allow monitoring of the collection sump without entry into the facility (entry only into the vestibule is required) and to allow radiological survey access. The sump is checked for water content in accordance with station radiation protection procedures, as well as sampled and discharged in accordance with applicable station procedures. The general arrangement of the OSGSF is given in Drawing M-24-23.

The OSGSF has been designed such that the dose rates at the exterior of the facility (walls and roof) are within the dose limits of 10 CFR 20. The area exterior to the OSGSF is a Zone 1-A area. The radiation zones assigned to the OSGSF are shown in Figure 12.3-71.

12.3.2 Shielding

The design of the station shielding is based on the design dose rates and the established design criteria. Using the sources given in Section 12.2 and the shielding design criteria, the shielding design is determined.

The original licensed power level was 3411 MWt. The original source term and shielding analyses were performed at a power level of 3565 MWt. Byron and Braidwood Nuclear Stations have uprated the core power level twice. First to a core power level of 3586.6 MWt, then to the Measurement Uncertainty Recapture uprate core power level of 3645 MWt. Accounting for core power level uncertainty, the analyzed core power level is 3658.3 MWt. This represents an increase of 2.6% from the original design basis. The core fission source given in Table 12.3-5 will increase by 2.6% after the uprates.

As stated in Section 12.2.4, the plant design basis radiation source terms will either remain valid for uprate or will increase by a maximum of 0.6%. As noted in that section, this small percentage increase is well within the conservative margin that was maintained in calculating the original source terms and modeling the shielding configurations to develop the design dose rates. Consequently, the radiation protection design features described in this section remain valid for power uprate.

Note that during plant operations, the plant ALARA program confirms adequacy of shielding and maintains the radiation levels in the plant within the design limits of the normal operation plant radiation zones.

12.3.2.1 General Shielding Design Criteria

Every component that handles radioactive fluids may require shielding; the thickness of which is based on the operational cycle of the component, the design dose rate, and the shielding material.

12.3.2.1.1 Regulatory Requirements

The shielding design dose rates for Byron/Braidwood meet 10 CFR 20 and 10 CFR 50, which are concerned with allowable radiation to individuals in restricted and unrestricted areas. The only shielding required to be safety-related is the control room and the primary containment shielding; this shielding satisfies the requirements stated in Criterion 19 of 10 CFR 50, Appendix A, and 10 CFR 20.

12.3.2.1.2 Shielding Requirements

Radiation protection of personnel, equipment, and materials is largely dependent upon the adequacy of the design of the station shielding system. Radiation shielding has the passive protection function of radiation attenuation and consists of material placed between radiation sources and personnel and/or equipment and materials needing protection from radiation.

The shielding system is designed and constructed to assure that the station can be operated and maintained such that the resultant radiation level and doses are within the limitations of applicable regulations and are as low as is reasonably achievable (ALARA). Specific design dose rate limits recommended to achieve this objective are discussed in Subsection 12.3.1 and listed in Table 12.3-2.

Shielding must be capable of performing its protective function throughout the plant lifetime and under the variable source and environmental conditions associated with all normal, anticipated abnormal operational, and design-basis accident conditions identified in the safety analysis reports and as noted in this section.

a. Normal Operating Conditions

For the purposes of shielding design, normal station operating conditions are considered to include conditions generally known as anticipated abnormal operational occurrences. Two modes of normal station operation are:

1. normal power operation of the reactor, including anticipated operational occurrences, and

2. normal shutdown of the reactor.

Shielding is designed to provide the required protective function under such conditions.

b. Accident Conditions

Station shielding provides protection to plant operating personnel and the general public under postulated design-basis accident conditions as defined in the Chapter 15.0.

1. Control Room Habitability

The main control room and associated areas are shielded such that, after a postulated design basis accident, the dose in the control room for the duration of the accident will not exceed 5 rem TEDE, including ingress and egress, as per requirements 10 CFR 50, Appendix A, Criterion 19. Subsection 6.4.2.5 describes control room shielding.

The radiation shielding protecting the main control room (and associated areas) is designed based on the anticipated radiation environment resulting from a postulated design basis accident. Figure 6.4-2 shows an isometric view of the main control room shielding.

2. Direct Offsite Doses

   All sources in the plant are adequately shielded to assure that radiation levels at the restricted area boundary are in compliance with 10 CFR 20 limits. Adequate station shielding is provided to limit site boundary doses, due to direct and scattered radiation from contained sources within the plant, practicing ALARA during normal operation in conformation with 10 CFR 20 and to within the limits specified in 10 CFR 100 for accidents analyzed using TID-14844 or 10 CFR 50.67 for accidents analyzed using alternative source term methodology.

3. Seismic and Safety Classification

   Structural walls of the station are designed, as required, to meet Seismic Category I requirements. Walls which are shielding walls may be designed Seismic Category I, depending upon the particular design requirements other than radiation protection requirements (e.g., structural integrity, load bearing capacity, etc.) that the walls must meet.

   The primary shield, the shield walls for the main control room, and the shield walls for the spent fuel pool are examples of shield walls which are designed Seismic Category I.

c. Protection of Equipment

   Appropriate shielding is provided, where needed:

   1. to limit radiation heating of building structural concrete,

   2. to reduce neutron activation of equipment, and

   3. to limit radiation to equipment and materials.

   Protection from neutrons and from neutron-induced gamma rays is important around neutron sources such as the nuclear reactor core. The primary shield around the reactor vessel is an example of station shielding designed to protect personnel and equipment against neutron radiation and neutron-induced gamma rays.

d. Additional Requirements

   In addition to the radiation protection functions discussed above, the shielding systems have other functional requirements. These generally depend on the location of the shield and the access requirements to equipment or areas beyond the shield. Thus, access to an area may be through the shield itself; e.g., through removable shield walls. Removable shield walls, portable shields, and compensatory shielding are discussed in Subsection 12.3.2.1.6.

12.3.2.1.3 Design Requirements

The station shielding system must be capable of performing its protective functions throughout the plant lifetime and under the variable source and environmental conditions which are anticipated and/or postulated for the plant.

The radiation attenuating materials which comprise the station shielding system are selected to assure no significant loss in radiation attenuation characteristics for at least 40 years of plant operation.

12.3.2.1.4 General Description and Design Parameters

The shielding system includes all concrete walls and associated radiation attenuating materials (e.g., lead, steel, and water) which are used to protect the public, plant personnel, equipment, and materials from radiation emitted from radioactive sources contained or generated within the plant. The radiation exposure of individuals, equipment, and materials is a function of the following basic parameters, which are given due consideration in the shielding design:

a. source strength (type, intensity, energy);

b. number of sources, source geometry, and self absorption factors;

c. shielding material, geometry, and mass between source(s) and receptor;

d. distance between source(s) and receptor;

e. time that receptor is exposed; and

f. allowed dose rate or dose.

Where radioactive crud buildup sources are known, the source strength parameter is appropriately adjusted and the shielding designed to accommodate the effects of crud buildup for at least 10 years of reactor operation. Where radioactive crud buildup sources are not known, but expected, the shielding design reflects appropriate conservatism to accommodate the expected effects of crud buildup for at least 10 years of reactor operation, and/or protective measures are used, where practicable, e.g., those discussed in Subsection 12.3.1.

12.3.2.1.5 Shielding Materials and Construction Methods

Bulk shielding structures such as cubicle shielding walls, floors, and ceilings are mainly designed of ordinary concrete, either of (solid) block or poured-in-place construction. Where space limitations are encountered, a special high density concrete (e.g., Hematite concrete) is employed to assure adequate radiation protection. Concrete is a mixture of materials, the exact proportions of which may differ from application to application. Concrete for radiation shielding is classified as ordinary or high density according to the unit weight of the aggregate. The design of concrete mixtures and forms, the construction of concrete radiation shielding structures, and the quality assurance provisions needed to verify that the desired quality of construction has been achieved is in accordance with accepted design criteria for concrete radiation shields.

Poured-in-place concrete construction is normally used for shielding structures which are load-bearing structural walls.

Concrete block walls are provided where necessary to accommodate equipment installation, removal, and construction. Concrete block wall installation is controlled to assure as-built radiation attenuation characteristics similar to those expected from equivalent poured concrete.

In the case of the primary shield around the reactor vessel, nuclear heating is not severe enough to warrant special designs (e.g., water cooling coils) for cooling the primary shield.

The reactor vessel nozzle inspection cavity hatches are made of stainless steel with no special neutron shielding material. They do not exhibit neutron shielding qualities.

Where a potential of leakage or spillage of radioactive material exists, effective features are provided in the design of the shielding to prevent the spread of contamination by seepage through walls. As discussed in Subsection 12.3.1.9, wall surfaces are coated with a nonporous coating to permit effective decontamination.

12.3.2.1.6 Removable Shield Walls, Portable Shielding, and Compensatory Shielding

Shielding is designed to be removable, where required, to provide personnel access for inspection, servicing, maintenance, or replacement of plant equipment.

Removable shield panels are provided in shield walls, floors, or ceilings as necessary where frequent access for maintenance or removal of equipment is required and if radiation levels in the area can cause excessive exposure. Such shielding is designed to minimize exposure to operating and maintenance personnel.

Compensatory, portable, or temporary shielding is considered in station design only as required where other more permanent shielding is not practicable. Where compensatory shielding is necessary, provisions are made to accommodate such shielding in terms of space, structural loading, clearances, and equipment accessibility.

The station shielding system uses three types of removable shield walls: stacked unmortared block, shield hatches and plugs, and shield doors. The primary functions of a removable wall are equipment installation, inspection, maintenance, and removal.

The following are guidelines for the design of removable shield walls. Note: the term major maintenance requires the removal of a removable shield wall in addition to repairing and maintaining equipment.

12.3.2.1.6.1 Stacked (Unmortared) Block

Removable stacked block walls that are provided to accommodate removal of equipment are constructed such that the top of the removable unmortared block sections are offset and provided with a lintel arrangement. The blocks are held in place by special metal frames to resist lateral pressure and seismic loads. Use of stacked unmortared block avoids unnecessary exposure associated with disassembly or mortared blocks.

Removable stacked block shield walls are used in the shield design when a room contains equipment that seldom requires replacement or major maintenance. Seldom is defined in the section as once a year. The type of shielded equipment which fits into this category are heat exchangers, pumps, and radwaste tanks.

12.3.2.1.6.2 Removable Shield Hatches and Plugs

Removable shield hatches (or removable floor slabs) and plugs are used in the shield design when a room contains equipment which often requires replacement or maintenance. Often is defined in this section to mean more frequent than once a year.

In addition to equipment that requires frequent maintenance, shield hatches or plugs are used, whenever practicable, for access to equipment and piping which have, or are in radiation areas that have, a dose rate greater than 3 R/hr.

The use of removable shield hatches or plugs minimizes the maintenance exposure to station personnel; shield hatch and plug design and construction shall be in accordance with ANSI N 101.6-1972.

A radiation detector probe access hole is provided in most of the filter and demineralizer removable shield hatches so that radiation levels of the contained equipment may be measured without removing the shield hatches. This is provided by boring a vertical stepped hole in the top of the shield hatch for insertion of a radiation detector. The arrangement is pictured in Figure 12.3-4.

The types of equipment that require removable shield hatches are demineralizers, filters, and pumps and motors, which are radioactive or are in radioactive areas.

12.3.2.1.6.3 <u>Shield Doors</u>

Shield doors are used when access requirements, maintenance requirements, or design consideration make it undesirable to adequately employ the removable shield walls mentioned previously. Shield doors can also be used with labyrinthine entrances where the dose rate at the entrance due to scattered radiation is greater than the design dose rate.

12.3.2.1.7 <u>Inspection (Inservice) and Maintenance Requirements</u>

Shielding is designed to permit access for required inspections, testing, and maintenance of plant systems and components which require these functions.

During construction, shield walls are visually inspected for cracks and separations that might compromise the shield. There are initial preoperational radiation surveys taken as well as periodic routine radiation surveys during power operation. These surveys serve as a check on the radiation buildup within auxiliary equipment and the adequacy of shielding design. Installed radiation monitoring systems survey continuously the radiation condition at certain areas of the plant and also serve as a check on the adequacy of shield wall design and construction.

As discussed in Subsection 12.3.1, biological protection of personnel during anticipated inspection and maintenance activities are considered in shielding design in the effort to maintain exposures ALARA.

12.3.2.1.8 <u>Shield Thicknesses</u>

Shield thicknesses are designed to reduce the average area dose rate to or below the assigned area dose rate level for worst-case conditions of normal plant operation or, where applicable, for accident conditions. Worst-case conditions include source terms appropriate to maximum power level and 1% failed fuel fraction as discussed in Section 11.1.

Shielding thickness are designed with consideration given to all sources in the area including localized hot spots or penetrations. Design parameters are listed in Subsection 12.3.2.1.4. Byron/Braidwood's shielding design is pictured in Drawings M-24-1 through M-24-23. Computer codes used in shielding design account for energy spectra and source strengths for each nuclide (including daughter products), material cross sections or attenuation coefficients for each material or element comprising the shield, dose buildup factors, and other relevant parameters.

12.3.2.1.9 Calculational Methods

In the design of the primary shield, the one-dimensional transport code ANISN (Reference 1) was used to calculate the transport of neutrons and gammas from the core. It also analyzed the subsequent production of capture-gamma rays in regions external to the core. The CASK code (Reference 2) coupled neutron-gamma ray library of cross sections was utilized with the ANISN code to enable all production and loss mechanisms for both neutrons and gamma rays to be handled in a single calculation. The parameters used in the ANISN calculation of the primary shield are given in Table 12.3-4. The fixed neutron source spectrum is given in Table 12.3-5.

Dose rates for siting and shielding design of the OSGSF were determined by calculating the direct dose rate using a point-kernel methodology and the skyshine dose rate using Monte Carlo transport methodology. The analyses used measured dose rates obtained at each steam generator region in conjunction with waste samples to identify the dominant gamma-emitting isotopes (see Table 12.2-55).

All other shields are designed for only gamma-ray attenuation by the standard point attenuation kernel (buildup factor, exponential attenuation, and geometry factor), numerically integrated over the volume of the source. The buildup factors and gamma-ray attenuation coefficients were obtained from published data (References 3 and 4). ISOSHLD-III (Reference 5) and QAD (Reference 6) are two point-kernel computer codes used in this design effort for Byron. For Braidwood design effort three point-kernel computer codes (References 5, 6 and 11) were used.

Tanks, demineralizers, filters and evaporators are generally mocked-up as cylinders with source and source densities homogenized and containing the maximum source volume capacity. Components containing radioactive water, including demineralizers, evaporators and filters are assumed to have a homogenized source density of 1.0 gm/cc. Tanks containing radioactive gases are assumed to contain their sources at the density of air (1.293 x 10^{-3} gm/cc). Dimensions are obtained from the vendor drawings of the component.

Spent fuel, charcoal filters, activated reactor internals and head, heat exchangers, radwaste drums, and other radioactive components have more complicated source geometries and source material compositions are more diverse. In all cases, the source is homogenized in order to fit into one of the simpler shielding geometry categories.

For example, the shielding of the radioactive drum storage area is mocked-up using the finite slab geometry option of ISOSHLD. The source composition is based on the composition of a single radwaste drum (Table 12.2-44) with a reduction factor used for the packing fraction encountered when cylindrical drums are placed adjacent to each other. Drum storage areas are assumed to be filled to maximum capacity. Sources used for the intermediate activity storage areas are the spent resin decayed for 90 days shown in Table 12.2-43 adjusted to a radionuclide content equivalent to 4.5 ft^3 of resin per drum. Sources for the low activity storage areas are based on radwaste evaporator concentrates (Table 12.2-39). Again an adjustment is made to 4.5 ft^3 of evaporator concentrate per drum. The resultant shielding of the intermediate and low activity drum storage areas is shown on Drawing M-24-19 and the design basis is given in Table 12.3-2.

Scattered radiation from labyrinths and penetrations is analyzed by the point-kernel single-scatter computer code GGG (Reference 7) or by the Monte Carlo code OGRE (Reference 8). As mentioned in Subsection 12.1.2.2.1, penetrations are located such that direct radiation from the source to the dose point is minimized, and the major contribution to the dose rate is from scattered radiation. If possible, wall penetrations are located above head height, and the use of wall and floor penetrations which run between radioactive areas and unlimited-access areas is minimal.

12.3.2.2 Specific Shielding Design Criteria

For purposes of design and operational control, it is necessary and convenient to classify areas (or zones) at the station according to expected personnel access and occupancy requirements. Areas of the station are assigned a design dose rate based on maintaining personnel exposures below 10 CFR 20 limits. Shielding is then designed in conjunction with appropriate radiological access control patterns to assure that area dose rates do not exceed area design dose rates.

Zone classification and dose rate categories for Byron/Braidwood are summarized in Table 12.3-1. Design dose rates for areas surrounding specific equipment and components are set forth in Table 12.3-2.

The shielding design-basis geometries of most major potentially radioactive components are given in Table 12.3-6. The calculations were performed using the computer codes, geometries, and compositions shown. But several considerations in the interpretation of the table need further explanation.

a. The dimensions shown are in many cases, approximate. However, they have been chosen such that the conservatism of the calculation is not compromised. The numbers shown are representative shielding design basis only and should not be used for any other purpose.

b. Source compositions are homogenized and tanks are assumed filled to the maximum level to represent the worst case. For some of the heat exchangers and steam generators the cooling coils were included for self-shielding. For others, cooling coils are ignored for added conservatism.

c. The shell thicknesses of components were in general included in the models for completeness. However, the effect of a fraction of an inch of iron on the gamma radiation considered was found to be insignificant and not included in the table.

d. Where the calculation of ceiling and/or floor shielding thicknesses was necessary, an axial as well as radial case was set up. The table shows representative dimensions for calculation in either direction.

e. The model shown in the table represents one piece of equipment. Where two or more components are in close proximity, dose rates are multiplied by a factor greater than 1 to account for multiple sources. This correction factor is generally equal to the number of components for components in the same cubicle.

f. Pumps are modeled as a pipe which is the same size as the largest pipe attached to the pump. The length of the pipe is determined by the length of the cubicle which houses it.

g. Pipes for pipe tunnel shielding are assumed to contain the same worst case sources as the outlet of the component they are connected to. Multiple pipes are assumed to be carrying their fluid simultaneously.

h. Dose rate detectors are placed on the outside surface of the wall and the dose rate is calculated for at least three thicknesses in each direction in order to obtain a graph of dose rate vs. thickness. This graph is used for choosing a final design shielding thickness using the design-basis dose rates of Table 12.3-2.

i. The source is assumed to be located near the inside surface of the shielding wall to account for associated piping which may run along the wall.

j. The resultant dose rate from all radioactive components in a particular area is considered in the choice of the shielding thickness to meet the design dose rate given on Table 12.3-2. For example, two components which are adjacent to a general access area (1 mr/hr) contribute less than 0.5 mr/hr each.

Expected peak external dose rates throughout the station, covering the two modes of normal plant operation described in

Subsection 12.3.2.1.2, are illustrated on the radiation mapping drawings, Figures 12.3-27 through 12.3-70. The dose rate categories used are given in Table 12.3-1, and each category is mapped on the drawings with a distinct graphic art screen.

The main control room and associated areas under accident conditions are included as a special region on the radiation mapping drawings.

12.3.2.3 Shield Wall Penetrations and Streaming Ratios

Penetrations in shield walls for pipes, HVAC ducts, and openings are located and designed to minimize radiation levels to personnel. Location and orientation of penetrations is selected to avoid streaming to areas most likely to be occupied by operating and maintenance personnel.

Compensatory shielding is used where necessary to reduce radiation streaming due to penetrations and localized shield deficiencies (expected hot spots).

Techniques which are used include increased wall thickness, provision for labyrinths or shadow shielding, provision for bends or directing the streaming path away from accessible areas, use of higher density materials such as lead, steel, or lead wool, etc.

Streaming along edges of access hatches, plugs, doors, etc., is minimized by the use of stepped off-sets.

Dose rates from radiation streaming are limited to a peak value at the penetration (i.e., as close as possible to the penetration on the low radiation side of the shield) of:

a. five times the design dose rate for uncontrolled access areas,

b. five times the design dose rate for penetrations located from 0 to 10 feet above the floors in controlled access areas which have design dose rates ≤ 10 mrem/hr, and

c. ten times the design dose rate for penetrations located more than 10 feet above the floor in controlled access areas.

For uncontrolled access areas having design dose rates of greater than 10 mrem/hr, specific streaming ratios for penetrations less than 10 feet above the floor are area dependent and may be more or less restrictive than those for controlled access areas of ≤ 10 mrem/hr.

The general dose rate which includes radiation streaming, averaged over accessible locations in the protected area, satisfies the design dose rate for the designated area.

Each penetration through a shield wall provides a streaming path for radiation which reduces the shielding effectiveness of the wall, except when the average density of a penetration with a small void content is greater than the average density of the shielding material being penetrated. The magnitude of the reduced effectiveness depends on geometry, material composition, and source characteristics.

In order to minimize the hazard of streaming and to maximum personnel protection, the guidelines listed below are followed in designing and locating shield wall penetrations.

a. Unnecessary penetrations are avoided. A service run or duct is not routed through a shielded cubicle unless that service is provided for equipment within the cubicle.

b. Penetrations are located as far away from radiation sources (e.g., the vessels or piping containing radioactive material) as is practicable.

c. Wherever it is practicable to do so, the penetration is located (1) near where two or three shield walls join, for example, near the upper corners of a room (so that the penetration is far away from radiation sources), and (2) near beams and columns which may serve as extra shielding to at least one side of the penetration (e.g., when beam is between source and penetration).

d. The penetration is located as high above the floor as is practicable and not less than 8 feet if possible.

e. The penetration penetrates through the thinnest of shield walls when a choice exists.

f. The diameter of the penetration is chosen as small as practicable. For electrical penetrations, use of sleeves or conduit having larger than 6-inch nominal diameter is avoided.

g. HVAC ducting avoids penetrating shield walls where practicable. HVAC ducts are routed through the labyrinthine entrances above the doorways of shielded cubicle were feasible. Cases exist, however, where shield wall penetration is necessary. In these cases the proper shielding option(s) to be taken are determined on an individual basis.

h. If electrical pipe or conduit is routed near the entrance to a radiation source cubicle, advantage is taken where practicable of the HVAC penetrations above the doorway and the conduits are run next to the HVAC control dampers and along the inside walls of the labyrinth and room. (In this case, no shield walls are penetrated.)

i. Where practicable, all pipe and conduit penetrations are grouted.

j. Offset penetrations are used when large lines or ducts penetrate shielding walls of cubicles which contain high levels or radiation, i.e., shield walls greater than or equal to 3-foot thick. HVAC ducts and openings are the most common penetrations that incorporate offsets, but in general, offsets are not used unless no other method will work.

12.3.3 Ventilation Requirements

The protective features for the ventilation systems are discussed in detail in Sections 9.4, 11.3, and 11.5.

Drawing M-24-3 (top center) depicts a typical physical layout of the filter systems utilized in the various plant ventilation systems.

Subsection 6.5.1 addresses the operation and design of the engineered safety feature filter systems.

Specific ventilation system designs are discussed in the following subsections:

a. control room HVAC system - Subsection 9.4.1;

b. radwaste building vent system - Subsection 9.4.3.1;

c. laboratory HVAC system - Subsection 9.4.3.2; and

d. auxiliary building HVAC system - Subsection 9.4.5.1.

12.3.3.1 Station Ventilation

The design of the station ventilation systems protects plant operating and maintenance personnel and the general public from exposure to radiation from airborne radioactive sources. This requirement applies to all operating conditions, including refueling, maintenance, and anticipated operational occurrences.

For areas other than the control room, the design philosophy is to prevent radioactive contamination of inlet air by preventing release of radioactive contamination to the outside air, instead of filtering inlet air. Exhaust air from all potentially contaminated areas shall be filtered to meet this philosophy.

Within the station, airflow is normally directed from lesser potential contamination areas to greater potential contamination areas. Areas of greater potential contamination are maintained at a more negative pressure than areas of lesser potential contamination (e.g., general access areas).

The design of the ventilation system for the control room complex is such that, following postulated design-basis accidents, radiation doses to main control room personnel for the duration of the accident will be within the limits set forth in 10 CFR 50, Appendix A, Criterion 19. Radiation protection for the control room consists of adequate air recirculation rates and systems for controlling iodine and particulates in addition to shielding. Shielding of the main control room is discussed in Subsection 6.4.2.5.

The station ventilation systems are designed so that exhaust air from potentially contaminated areas can be routed through appropriate filters prior to discharge through the ventilation stacks. Stack releases shall be within acceptable limits such that they do not cause offsite doses to exceed the limits set forth in 10 CFR 100 for accidents analyzed using TID-14844, or 10 CFR 50.67 for accidents analyzed using alternative source term methodology, or the limits set forth in 10 CFR 50 Appendix I for normal operating conditions.

Radiation protection considerations for waste filters (which include HVAC filters) are discussed in Subsection 12.3.1.7.

12.3.3.2 Design Criteria

To meet the design objectives, the following radiological safety design guidelines were utilized:

a. The system is designed to maintain air flows from clean areas to potentially contaminated areas and from areas of potentially lower level contamination to areas of potentially higher level contamination (prior to exhaust).

b. The system is designed to ensure that negative pressure differential with respect to surrounding areas is maintained inside potentially contaminated cubicles. Control dampers and seals are provided to assure the airflow patterns can be properly maintained.

c. Fume hoods are utilized in the laboratories to facilitate safe processing of radioactive samples by directing contaminants away from the breathing zone to the filtering and ventilation system.

d. Equipment decontamination facilities are ventilated to assure control of released contamination and prevent personnel exposure and the spread of contamination.

e. Exhaust air is routed through HEPA filters or a combination of HEPA and charcoal filters where necessary before release to the atmosphere to reduce onsite and offsite radioactivity levels.

f. Air is supplied to each principal building via separate supply intakes and duct systems.

g. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered and can be passed through charcoal adsorbers to prevent contamination of the control room by smoke or excessive radioactivity.

h. Transient airborne contamination may result due to maintenance. Special procedures, such as: portable air handling units, and the use of plastic tents is instituted to minimize the contamination on a case by case basis.

i. All exhaust ventilation systems designed to handle potentially contaminated air in the plant are of similar design. A typical filtration system is equipped with a demister and/or prefilter, a heater for humidity control, a set of prefilters, and a set of HEPA filters. Filter systems designed to remove radioiodine are equipped with a charcoal filter bank and an additional set of HEPA filters to collect charcoal fines emerging from the charcoal filters. Dampers are provided before and after the filter train to isolate the train during filter changes.

j. All filter systems in which radioactive materials could accumulate to produce significant radiation fields external to the ductwork are appropriately located and shielded to minimize exposure to personnel and equipment.

k. Filters in all systems are changed based upon the airflow and the pressure drop across the filter bank. In the case of the prefilters, a pressure drop of 1 inch of water equivalent across the bank is cause for changeout. HEPA filters are changed when the pressure drop across them reaches 2 inches of water equivalent. Charcoal adsorbers are changed based on the residual adsorption capacity of the bed as measured by test samples or canisters removed and analyzed at intervals.

l. While the majority of the activity in the filter train is removed by simply removing the contaminated filters, further decontamination of the internal structure is facilitated by the proximity of electrical outlets for operation of decontamination equipment, and water supply for washdown of the interior, if necessary. Drains are provided on the filter housing for removal of contaminated water.

These guides are incorporated and fully described in Section 9.4.

12.3.3.3 Cubicles Requiring Charcoal Air Filtration

Cubicles which contain the following systems or components shall have provisions to exhaust the ventilation air through charcoal filters.

a. post-LOCA recirculation systems;

b. waste filters and demineralizers (see Subsection 12.3.1.7);

c. evaporators for radwaste or recycle; and

d. items with significant concentration of I-131 (more than 0.1 times the I-131 concentration in the reactor coolant, or more than 0.25 μCi/cc, whichever is more limiting).

In general, cubicles containing static tanks and heat exchangers need not have ventilation air passed through charcoal filters since the leakage from such components on cubicle floor is not assumed to have a I-131 concentration exceeding 0.1 times the I-131 concentration in the primary reactor coolant. (The venting of radwaste tanks is through charcoal filters, however, as discussed in Subsection 12.3.1.5).

The following is a list of cubicles which have provisions to pass ventilation air through charcoal filters; leaking equipment in these cubicles could produce levels of airborne I-131 which are one-tenth the levels produced due to leaks in primary coolant equipment.

a. radwaste evaporator cubicles,

b. recycle evaporator cubicles,

c. demineralizer cubicles, and valve aisles

d. primary sample room (local filtration),

e. RHR heat exchanger cubicles,

f. letdown heat exchanger valve aisles,

g. centrifugal charging pump cubicles,

h. positive displacement charging pump cubicles,

i. safety injection pump cubicles,

j. auxiliary building equipment drain pump cubicle,

k. waste gas compressor cubicle,

l. gas analyzer cubicle,

m. recycle evaporator feed pump cubicles, and valve aisles

n. gas decay tank cubicles, valve aisles, and pipe tunnel,

o. RHR pump cubicles,

p. containment spray pump cubicles,

q. volume control tank valve aisles,

r. surface condenser rooms,

s. fuel handling building,

t. volume reduction equipment cubicles,

u. radwaste and blowdown mixed bed demineralizer valve aisle, operating area, and cubicles,

v. filter valve aisle, operating area, pipe tunnel, associated filter cubicles, and main area,

w. clothes change and shower room

x. collection drain sump rooms,

y. pipe tunnels,

z spray additive tank room and pipe penetration area,

aa. CASP room,

bb. recycle holdup tank pipe tunnel and tank room,

cc. floor drain sump rooms,

dd. auxiliary steam pipe tunnels,

ee. spent resin and concentrates pump room,

ff. surface condenser rooms,

gg. letdown reheat heat exchanger valve operating area, and

hh. HRSS lab area and tank and pump room.

12.3.3.3 Cubicles Requiring Charcoal Air Filtration

Cubicles which contain the following systems or components shall have provisions to exhaust the ventilation air through charcoal filters.

a. post-LOCA recirculation systems;

b. waste filters and demineralizers (see Subsection 12.3.1.7);

c. evaporators for radwaste or recycle; and

d. items with significant concentration of I-131 (more than 0.1 times the I-131 concentration in the reactor coolant, or more than 0.25 μCi/cc, whichever is more limiting).

In general, cubicles containing static tanks and heat exchangers need not have ventilation air passed through charcoal filters since the leakage from such components on cubicle floor is not assumed to have a I-131 concentration exceeding 0.1 times the I-131 concentration in the primary reactor coolant. (The venting of radwaste tanks is through charcoal filters, however, as discussed in Subsection 12.3.1.5).

The following is a list of cubicles which have provisions to pass ventilation air through charcoal filters; leaking equipment in these cubicles could produce levels of airborne I-131 which are one-tenth the levels produced due to leaks in primary coolant equipment.

a. radwaste evaporator cubicles,

b. recycle evaporator cubicles,

c. demineralizer cubicles, and valve aisles

d. primary sample room (local filtration),

e. RHR heat exchanger cubicles,

f. letdown heat exchanger valve aisles,

g. centrifugal charging pump cubicles,

h. positive displacement charging pump cubicles,

i. safety injection pump cubicles,

j. auxiliary building equipment drain pump cubicle,

k. waste gas compressor cubicle,

l. gas analyzer cubicle,

m. recycle evaporator feed pump cubic
aisles

n. gas decay tank cubicles, valve aisl
tunnel,

o. RHR pump cubicles,

p. containment spray pump cubicles,

q. volume control tank valve aisles,

r. surface condenser rooms,

s. fuel handling building,

t. volume reduction equipment cubicles,

u. radwaste and blowdown mixed bed demineralizer valve aisle, operating area, and cubicles,

v. filter valve aisle, operating area, pipe tunnel, associated filter cubicles, and main area,

w. mask cleaning room

x. pipe tunnels,

y. spray additive tank room and pipe penetration area,

z. CASP room,

aa. recycle holdup tank pipe tunnel and tank room,

bb. floor drain sump rooms,

cc. auxiliary steam pipe tunnels,

dd. spent resin and concentrates pump room,

ee. surface condenser rooms,

ff. letdown reheat heat exchanger valve operating area,

gg. HRSS lab area and tank and pump room,

hh. Unit 2 collection drain sump room/hot machine shop, and

ii. Unit 1 collection drain sump room.

12.3.3.4 Ventilation Design Features

The ventilation system parameters for radiologically significant areas in the auxiliary building are provided in Table 12.2-45.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Two fixed systems are provided to monitor radiation/radioactivity levels within the plant. These are:

a. the area radiation monitoring system (ARMS), and

b. the continuous airborne monitoring system (CAMS).

Portable CAMS, grab sampling capability, and automatic samplers are also provided to supplement the fixed monitoring systems.

The fixed ARMS is provided to continuously measure, indicate, and trend the levels of radiation in general access and operational areas. Radiation alarms are activated when predetermined levels are exceeded. The objective is to keep operating personnel informed of the radiation levels in the selected areas and thus assist in avoiding unnecessary or inadvertent exposure.

The fixed CAMS is provided to measure, indicate, and trend the levels of airborne radioactivity in the air exhausted from cubicles or branch HVAC exhaust ducts. The objective is to warn operators that airborne activity may be present in the area or cubicle serviced by the monitored exhaust system, and thereby assist in avoiding unnecessary or inadvertent exposure. CAMS also provides a means for identifying trends in air concentration levels and the source of the activity. Each fixed CAM activates visual and audible control room alarms when predetermined levels are exceeded. The fixed monitors are also used to assist in the monitoring and control of effluents as described in Section 11.5. Portable CAMS are generally used to monitor the ambient air in normally occupied areas. They may be used in conjunction with the fixed CAMS to locate the source of the airborne activity.

12.3.4.1 Area Radiation Monitoring Instrumentation

The area radiation monitoring system (ARMS) is provided to fulfill the following specific radiological safety objectives:

a. To provide operating personnel in the main control room with an indication and record of radiation levels at selected locations within the various plant buildings (e.g., to warn of excessive radiation levels in areas where nuclear fuel is stored or handled).

b. To contribute radiation information to the control room so that correct decisions may be made with respect to deployment of personnel in the event of a radiation incident.

c. To assist in the detection of unauthorized or inadvertent movement of radioactive material in the plant including the radwaste area.

d. To supplement other systems in detecting abnormal migrations from and radioactive material in the process streams.

e. To provide local alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area. Area monitors in high noise areas feature visual as well as audible alarms.

f. To assist in maintaining in-plant personnel exposure as low as reasonably achievable.

To implement these objectives area radiation detectors are provided throughout the plant at locations indicated in Table 12.3-3 and shown on the radiation shielding design drawings in Section 12.3 (see Drawings M-24-1 through M-24-23).

ARM's are installed in the vicinity of the fuel pool and on the fuel handling building overhead crane and in containment to sense abnormal or accident conditions as indicated in Table 12.3-3.

The ranges and initial setpoints are also given in Table 12.3-3. The general requirements for the ARM's are as follows:

Energy Response

Gamma energy response of the detectors extends from 0.02 to 3 MeV. The energy dependence is within ±20%.

Channel Accuracy

The overall channel accuracy within the environmental limitations of the system is ±20% or better of reading (digital output).

Precision

The reproducibility of each channel for any given measurement over its stated range is ±10% or better at the 95% confidence level.

Power Supply

Area radiation monitors receive power from 120-Vac buses. The audio and visual alarms receive power from the same 120-Vac buses. Nuclear safety-related area radiation monitors receive power from ESF buses.

Calibration

Area radiation monitor calibration frequency is established based on safety significance of the application and equipment historical performance. Area radiation detectors have the capability of being calibrated for dose rate in the calibration facility by exposing the detectors to the radiation field from an isotope of known activity. Cabling is provided from the calibration facility to the control room permitting readout in the control room. The intensity of the calibration field can be varied, thereby allowing a multipoint calibration.

Location

Location of area radiation detectors is provided in Table 12.3-3. Each area radiation detector is connected to the main control room central processor by microprocessors (monitors) which control the detectors, process and store its data. The central processing console for each unit includes a video display unit at Byron and a CRT display and printer at Braidwood. Each monitor maintains trend files that can be accessed through the central processing console.

Dedicated readout modules and recorders are provided for those nuclear safety-related area radiation monitors only, whose application in the plant design requires a safety-related operator interface and/or data collection capability. All monitors are designed to fail in the safe (Alarm) mode.

Conformance to Applicable Regulations

The ARMS conforms to Sections 4.2 and 5.3.4.1 of ANSI N13.2-1969. Qualified personnel have been used in the engineering phase and will be used during operation to assure that radiation exposures to plant personnel will be ALARA. Regulatory guidance concerning effluents and ANSI N13.1-1969, do not directly apply to the ARMS.

12.3.4.2 Continuous Airborne Monitoring Instrumentation

The continuous airborne monitoring system (CAMS) is provided for monitoring in-plant airborne radioactivity levels. The specific radiological safety objectives are the same as Subsection 12.3.4.1. Continuous air monitors (CAMs) are discussed in Section

11.5 and identified in Table 11.5-1, including location, range, sensitivity, and alarm setpoints. Monitor locations are shown and identified in Drawings M-24-1 through M-24-22. Probe locations are shown on the HVAC system drawings in Section 9.4.

The fixed continuous airborne monitors (CAMs) are provided to monitor for airborne radioactivity in compartments which may be occupied and may contain airborne radioactivity. Since there are too many rooms or cubicles to monitor independently, a limited number of CAMs are provided to continuously monitor the air from selected branch exhaust ducts of the HVAC system.

The exhaust from a single room may be diluted by the exhaust from other rooms before the air gets to the monitoring point. Therefore, the monitor must be sensitive enough to respond to the diluted activity. The maximum possible dilution factor for any cubicle is:

$$DF = \frac{F\ \text{cubicle (cfm)}}{F\ \text{duct (cfm)}}$$

Using the detectability factor for MPC_a ($DMPC_a$) given in Table 12.3-9, an expression can be written for the time, T, it takes to detect the presence of MPC_a levels in the exhaust ducts (The term MPC as used in this section refers to a 10CFR20 limit in effect prior to January 1, 1994):

$$T = 1/(DMPC_a * DF),\ \text{(hrs)}$$

where,

T	= time to detect particulate and iodine MPC_a, (hr),
$DMPC_a$	= detectability factor for MPC_a (see Table 12.3-9),
DF	= dilution factor for 10 MPC-HR detectability,
$F_{cubicle}$	= flow in exhaust from cubicle, (cfm), and
F_{duct}	= flow in branch duct where monitor is located, (cfm).

Table 12.3-9 shows the sensitivity of the particulate, iodine, and noble gas channels for the isotopes of greatest interest. These sensitivities were compared to maximum permissible concentrations in air (MPC_a) of the most restrictive particulate and iodine radionuclides in the areas and cubicles of lowest ventilation flow rate. The criterion used was that airborne radioactivity from the areas described above and having an activity concentration of one MPC_a would be detected within 10 hours. Exhaust flow rates from cubicles and in branch ducts were examined to determine dilution factors for this assessment. The exhaust flow rates for the monitored branch ducts and the individual room exhaust flow rates are given in the drawings cited in Section 9.4. The location of the radiation monitors are also shown on these drawings. An investigation using the above data indicates that the system is capable of detecting 10 MPC_a-hrs of airborne particulate and iodine radioactivity in the rooms, cubicles, and areas discussed above which may be occupied and may contain airborne radioactivity.

The general requirements for the CAMS are as indicated in the following.

Energy Response of Channels

Gamma energy response of the detector channels used for gamma monitoring extends from 0.08 to 3 MeV. The energy dependence is within ±20%. Beta detector channels are capable of detecting minimum of 0.07 MeV beta (e.g., 7 mg/cm^2 aluminum window).

Channel Accuracy

The accuracy of each channel is within ±20% or better of the reading.

Precision

The precision is ±10% or better at the 95% confidence level.

Particulate Filters

Filters have an efficiency of 99% or better for 0.3 micron particles.

Iodine Collector

Iodine collectors consist of activated, impregnated charcoal cartridges in metal canisters. Prefilters are installed upstream of cartridges to remove particulates.

Representative Sampling Design

Sampling systems are designed to assure representative sampling for off-line CAM. Isokinetic sampling nozzles are used for extraction of gaseous samples from gaseous streams. The in-duct isokinetic probes comply to the standard set forth in ANSI N13.1-1969. Sample piping is designed to avoid sharp bends and stagnant zones. Off-line detector assemblies are designed with temperature, pressure, and flow regulators as required for instrumentation. All off-line monitors are capable of being purged with air.

Power Supply

Electric power is provided to CAMs from permanent supplies.

Alarms

The CAMs are provided with two adjustable alarm setpoints (alert and high alarms). There is also an instrument failure alarm. Each of the above indicates in the control room and has a relay contact output at the microprocessor cabinet.

Periodic Testing

All CAMs are capable of being checked, tested, and calibrated periodically to verify proper operation. Check sources, test signals, and calibration sources are provided as applicable.

It is possible to periodically test those CAM's, which are related to nuclear safety in accordance with criteria for periodic testing of protection system actuation functions and IEEE 338-1971. Such testability means the ability to duplicate required functions as closely as possible (e.g., during reactor operation) without impairing plant operation. The air sample calibration programs comply with the guidance contained in ANSI guide IEEE N232C Section 4.5.

All trip circuits are capable of convenient operational verification by means of test signals or through the use of portable sources.

Radionuclide standards of two or more different source strengths are provided. Gaseous detectors requiring in-place radiogas calibration are provided with necessary isolation valves. Recirculation design is employed to minimize gas usage.

The shield assembly is designed to allow quick and simple purging, decontamination and removal of sample canister, and replacement with standard canister.

12.3.5 References

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2. RSIC Data Library, "DIC-23/CASK 40 - Group Coupled Neutron and Gamma-Ray Cross Section Data," Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, December 1972.

3. J. H. Hubbel, "Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 keV to 10 GeV," NSRDS-NBS29, August 1969.

4. S. Buscagline and R. Manzini, "Buildup Factors: Coefficients of the J. J. Taylor Equation," ORNL-tw-80, February 1964.

5. R. L. Angle, J. Greenborg, and M. M. Hendrickson, "ISOSHLD - A Computer Code for General-Purpose Isotope Shielding Analysis," BNWL-236, Pacific Northwest Laboratory, Richland, Washington, June 1966, Supplement 1, March 1977, Supplement 2, April 1969.

6. R. E. Malenfant, "QAD: A Series of Point-Kernel General-Purpose Shielding Programs," LA-3573, Los Alamos Scientific Laboratory, April 5, 1967.

7. R. E. Malenfant, "G^3: A General-Purpose Gamma-Ray Scattering Program," LA-5176, Los Alamos Scientific Laboratory, June 1973.

8. D. K. Trubey and M. B. Emmett, "OGRE-G, A General-Purpose Monte Carlo Gamma-Ray Transport Code," ORNL-TM-1212, Oak Ridge National Laboratory, 1966.

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

10. W. J. Manion and T. S. LaGuardia, "An Engineering Evaluation of Nuclear Power Reactor Decomissioning Alternatives," Atomic Industrial Forum, Inc., Document No. AIF/NESP-009, November 1976.

11. Braidwood only – MicroShield 8.01, Grove Software RadiationSoftware.com, 2008, Incorporated in URS qualified software library, April 2009.

TABLE 12.3-1

CLASSIFICATION OF RADIATION ZONES FOR SHIELD DESIGN AND RADIOLOGICAL ACCESS CONTROL

ZONE DESIGNATION	DESIGN DOSE RATE* (mrem/hr)	TYPICAL REGIONS	RADIOLOGICAL ACCESS CONTROL	GENERALIZED NRC POSTING REQUIRED
I-A	0.2	Plant grounds outside security fencing and office areas	Per station procedures	None
I-B	0.5	Most plant grounds within security fencing (also OSGSF roof)	Per station procedures	None
I-C	1	Most operating areas and passageways	Per station procedures	None
I-D	2	Assigned as required in design	Per station procedures	None
II-A	4	Assigned as required in design	Per station procedures	None
II-B	10	Assigned as required in design	Per station procedures	Radiation area

*For a given operating mode, the design dose rate is the maximum dose rate expected after 10 years of plant operation in a given region outside highly localized radiation streaming paths.

TABLE 12.3-1 (Cont'd)

ZONE DESIGNATION	DESIGN DOSE RATE* (mrem/hr)	TYPICAL REGIONS	RADIOLOGICAL ACCESS CONTROL	GENERALIZED NRC POSTING REQUIRED
II-C	20	Assigned as required in design	Per station procedures	Radiation area
II-D	100	Assigned as required in design	Per station procedures	Radiation area
III	>100	Generally a source region	Per station procedures	High radiation area
IV	Not Assigned			
V	0.2 (Normally)	Main control room	Per station procedures	None
	Postaccident**	Main control room	Per station procedures	As required

*For a given operating mode, the design dose rate is the maximum dose rate expected after 10 years of plant operation in a given region outside highly localized radiation streaming paths.

**For the initial, 30-day, postaccident period, the design doses to personnel during access and occupancy of the control room are limited to a maximum of 5 rem to the whole body, 30 rem to the thyroid, 30 rem to the bone, and 15 rem to the lung for accidents analyzed using TID-14844. For accidents analyzed using alternative source term methodology, radiation exposure limits are provided in 10 CFR 50.67.

TABLE 12.3-2

SPECIFIC SHIELDING DESIGN CRITERIA

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
I.	Reactor containment building							
I.C.1	Reactor containment building	17Z6	Normal operation	Area outside building	1Z1	0.5 mrem/hr	I-B	1A9-1 2A9-1
I.C.2	Reactor containment and equipment El. 426 ft - 0 in.	18Z10	Normal operation	(1) outside per-sonnel air lock and equipment hatch	5Z2 5Z12	4 mrem/hr	II-A	1A6-1 2A6-1
	El. 401 ft - 0 in.	17Z6	Normal operation	(2) emergency personnel air lock	17Z8	1 mrem/hr	I-C	1C5-10 2C5-10

TABLE DEFINITIONS

*	- This is the design dose rate for the protected area and does not include the contribution of any radioactive components that might be in the area.
**	- Hot spot criteria
x	- Design dose rate is proportional to RBP given in parentheses.
xx	- Same as xxx except that the protected area is a radiation area.
xxx	- This protected area is a high radiation area due to the presence of radioactive valves and piping. These sources hinder detection of radiation from the shielded source and would cause high personnel exposure. Therefore a RBP during startup testing would only achieve unnecessary personnel exposure.
N/A	- Not applicable to start up testing; the zone designation in brackets is the expected level following shutdown or during refueling.
HOT SPOT	- The dose rate near penetration in shield walls are permitted to be five times the dose rate specified for the shield wall. For additional information on the hot spot criteria, see Subsection 12.3.2.3.
TML	- Too many zones to list.

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
1.C.3	Reactor containment and equipment	18Z10	Normal operation purge penetration	Area outside containment in auxiliary bldg. on El. 451'-0"	3Z12 3Z14	2 mrem/hr 10 mrem/hr	I-D **	1A7-9 2A7-9
I.P.	Primary shield							
I.P.1	Reactor core and reactor pressure vessel	15Z1	Normal operation	Outside center-plane of core outside primary shield	15Z2			
			(1) neutrons plus gammas			5 mrem/hr	III	xxx
			(2) thermal neutrons (<1.12eV)			1 x 10^5 neutrons/cm^2-sec	III	xxx
			(3) epithermal neutrons (1.12 eV<E≤3.35 keV)			6.5 x 10^3 neutrons/cm^2-sec	III	xxx
			(4) fast - neutrons (E>3.35 keV)			7.5 x 10^3 neutrons/cm^2-sec	III	xxx
I.P.2	Reactor core and reactor pressure vessel (shutdown)	15Z1	(1) 8 hours after shutdown	Outside primary shield	15Z2	25 mrem/hr	[II-D]	N/A
			(2) 1 day after shutdown	Outside primary shield		10 mrem/hr	[II-B]	N/A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
I.M.	Missile wall (secondary shield)							
I.M.1	Reactor coolant loop, steam generators, and containment sump	15Z2 16Z1 17Z5	(1) normal operation	Outside missile wall and fan cooler penetrations,	15Z3 16Z3 17Z6	20 mrem/hr	II-D	1,2C3-3 1,2C4-2 1,2C5-2 1,2C6-5
			(2) 1 day after shutdown	Outside missile wall		2 mrem/hr	[I-D]	N/A
I.RP.	Refueling cavity							
I.RP.1	Upper internals	17Z1	100 hours after shutdown	Outside storage area	17Z5	5 mrem/hr	[III]	N/A
I.RP.2	Lower internals	17Z1	1 week after shutdown	Outside storage area	17Z5	5 mrem/hr	[III]	N/A
I.RP.3	Spent fuel assembly and RCC elements, reactor and reactor cavity pool water	17Z1	100 hours after shutdown	(1) outside refueling canal, inside missile wall	16Z1	100 mrem/hr	[III]	N/A
				(2) 2 feet above water level	18Z10	2 mrem/hr	[II-A]	N/A
				(3) outside fuel transfer canal, outside missile wall	16Z3	1 mrem/hr	[I-C]	N/A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
I.B.	Reactor containment building general							
I.B.1	Reactor coolant loop	15Z2	Normal operation	Area outside missile wall	15Z3	20 mrem/hr	II-D	1C3-2 2C3-2
I.B.2	Reactor coolant drain, tanks and pumps	15Z2	(1) normal operation	Area outside missile wall	15Z3	20 mrem/hr	II-D	1C3-4 2C3-4
			(2) during refueling	Area outside missile wall		2 mrem/hr	[II-D]	N/A
I.B.2	Containment sump pumps	15Z2	Shutdown	Area outside missile wall	15Z3	2 mrem/hr	[II-D]	N/A
I.B.3	Incore instrument shaft and storage area	17Z2 15Z2	(1) normal operation	Area outside missile wall	17Z6 15Z3	20 mrem/hr	II-D	XXX
			(2) during refueling	Area outside missile wall		2 mrem/hr	[I-D]	N/A
I.B.4	Regenerative heat exchangers and excess letdown	17Z3 17Z4	(1) normal operation	Area outside missile wall	17Z6	20 mrem/hr	II-D	1C5-4
	heat exchangers	N/A	(2) shutdown	Area outside heat exchanger cubicle		2 mrem/hr	[I-D]	N/A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
I.B.5	Seal table and core detector storage	17Z2	(1) normal operation	Area outside missile wall	17Z6	20 mrem/hr 100 mrem/hr	II-D **	1C5-3 2C5-3
			(2) shutdown	Area outside cubicle		2 mrem/hr	[I-D]	N/A
I.B.6	Containment char-coal filter system	18Z10	(1) after cleanup mode	Area outside filter housing	18Z10	5 mrem/hr	II-D	xx
			(2) during refueling	Area outside filter housing		2 mrem/hr	[II-D]	N/A
I.B.7	Main steam pipe chases	16Z5 16Z6	Normal operation	Area outside pipe tunnel	17Z6	20 mrem/hr	II-D	1C5-6 2C5-6
II.	Fuel handling building							
II.SF.1	Spent fuel pit	14Z2	Containing 1-2/3 core with 1 week decay	(1) outside spent fuel pool wall near heat exchangers	14Z4	15 mrem/hr	[III]	N/A
				(2) pipe tunnel on sides of spent fuel pool	14Z1 14Z9	15 mrem/hr	[III]	N/A
				(3) inside fuel transfer canal	14Z10	50 mrem/hr	[III]	N/A
				(4) 2 feet above water level	13Z8	4 mrem/hr	[II-A]	N/A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
II.SF.2	Spent fuel transfer canal	14Z10	Spent fuel assembly with 100 hours decay	(1) outside of canal	12Z5 12Z10	15 mrem/hr	[III]	N/A
				(2) 2 feet above water level	13Z9	2 mrem/hr	[I-D]	N/A
II.SF.3	Spent fuel pit heat exchangers	14Z4	Normal operation	Area outside heat exchanger cubicle	14Z6	2 mrem/hr	I-D	F5-2 F5-3
II.SF.4	Spent fuel pit pumps and sump	14Z4	Normal operation	Area outside pump cubicle	14Z6	2 mrem/hr	I-D	F5-3
II.SF.5	Spent fuel pit skimmer pump	14Z4	Normal operation	Area outside pump cubicle	14Z6	2 mrem/hr	I-D	Ft-3
II.SF.6	Fuel handling building	14Z6	Normal operation	Area outside building	1Z1	0.5 mrem/hr	I-B	F5-5A F5-5B
III.	Auxiliary building							
III.1	Elevation 330 ft. - 0 in.							
III.1.1	Auxiliary building sumps	10Z1 10Z5	Filled with contaminated water	Area outside cubicles	10Z4 10Z8	2 mrem/hr	I-D	1A1-1 2A1-1
III.1.2	Auxiliary building equip. drain pumps	10Z3 10Z7	Pumping contaminated water	Outside pump rooms	10Z4 10Z8	2 mrem/hr	I-D	1A1-2 2A1-2

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.2	Elevation 346 ft - 0 in.							
III.2.1	Recycle holdup tanks	9Z14 9Z13	Filled with contaminated water	Area outside cubicle	9Z1	2 mrem/hr	I-D	OA2-14A OA2-14C
III.2.2A	Gas decay tanks	9Z6	Filled with fission product gases	(1) area outside tank cubicles	9Z1	2 mrem/hr	I-D	OA2-17
				(2) valve aisle	9Z2	15 mrem/hr	III	xxx
III.2.2B	Waste gas valve aisle	9Z2	Normal operation	(1) valve operating area	9Z1	2 mrem/hr 10 mrem/hr	I-D **	OP2-34 thru OP2-38
				(2) area outside cubicle	9Z1	2 mrem/hr	I-D	OA2-15 OA2-16
III.2.3	Auxiliary building collection sump pumps	9Z18 9Z35	Filled with contaminated water	Area outside pump and sump area	9Z1	2 mrem/hr	I-D	1A2-12 2A2-12
III.2.4	Auxiliary building equipment drain tanks	9Z19 9Z20	Filled with contaminated water	Area outside cubicles	9Z1	2 mrem/hr	1-D	1A2-10 2A2-10
III.2.5	Containment spray pumps	9Z7 9Z8	(1) Normal operation	Area outside pump cubicle	9Z27 9Z28	15 mrem/hr	III	xxx
			(2) 12 hours after LOCA	Area outside pump cubicle		100 mrem/hr	-	-

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.2.6	Moderating heat ex-changers, letdown chiller and letdown reheat heat ex-changers	9Z21 9Z25 9Z33 9Z22 9Z26 9Z34	Normal operation	(1) area out-side cubicle	9Z1	2 mrem/hr	I-D	1A2-3,4,5,6,9 2A2-3,4,5,6,9
				(2) valve opera-ting area	9Z37 9Z36	4 mrem/hr	II-A	1A2-7,8 2A2-7,8
				(3) valve aisle	9Z23 9Z24	15 mrem/hr	III	xxx
III.2.7	Recycle evaporator feed pumps	9Z15 9Z16	Pumping con-taminated water	(1) outside pump cubicle	9Z1	2 mrem/hr	I-D	OA2-14A OA2-14C
				(2) valve opera-ting area	9Z1	2 mrem/hr	I-D	OA2-14B
				(3) valve aisle	9Z17	15 mrem/hr	II-D	xx
III.2.8	Pipe tunnel for recycle evaporator	11Z7	Contaminated water, normal operation	Area outside pipe tunnel	9Z1	2 mrem/hr	I-D	OA2-13C
III.2.9	Recycle evaporator packages	9Z9 9Z11	Operation of evaporators	Outside evaporator package cubicles	9Z1	2 mrem/hr 10 mrem	I-D **	OA2-13A&B OP2-31&32
III.2.10	Waste gas pipe tunnel	11Z7	(1) Pipes con-taining contami-nated gas	Area outside pipe tunnel	8Z1	2 mrem/hr	I-D	OA3-18A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.2.11	RHR pumps	9Z27 9Z28	(1) operation at 4 hours after shutdown	Entrance to pump cubicle	9Z7 9Z8	15 mrem/hr	[III]	N/A
			(2) gap release accident, any 8 hour period	Area outside pump cubicle	9Z7 9Z8	0.5 rem after a gap release accident	-	-
			(3) 12 hours after LOCA	Area outside pump cubicle	9Z7 9Z8	100 mrem/hr	-	-
III.3	Elevation 364 ft 0 in.							
III.3.1	Recycle holdup tanks	8Z11 8Z12	Filled with contaminated water	(1) outside tank cubicle	8Z1	2 mrem/hr	I-D	OA3-12A OA3-12B
				(2) valve operating area	8Z1	2 mrem/hr	I-D	OA3-12C
III.3.2	RHR heat exchangers	8Z18 8Z28 8Z6 8Z21	(1) operation 4 hours after shutdown	Outside heat exchanger cubicle	8Z1	2 mrem/hr	[I-D]	N/A
			(2) 12 hours after LOCA	Outside heat exchanger cubicle		100 mrem/hr	-	-

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
			(3) Operation at 2 hours after a gap release accident	Outside heat exchanger cubicle		5 mrem/hr	-	-
III.3.3	Safety injection	12Z6 12Z9	Pumping sump water, 12 hours after LOCA	Outside pump cubicle	12Z5 12Z10	100 mrem/hr	-	-
III.3.4	Blowdown condensers	8Z6 8Z10	Primary system leakage of 1 gpm with a total blowdown flow for each unit at 135 gpm	Outside condenser cubicle	8Z1	2 mrem/hr	I-D	OA3-14A OA3-14B
III.3.5	Chemical drain tank	8Z2	Filled with contaminated water	Outside tank cubicle	8Z1	2 mrem/hr	I-D	OA3-17
III.3.6	Chemical drain pumps	8Z3	Pumping contaminated water	Outside pump cubicle	8Z1	2 mrem/hr	I-D	OA3-16
III.3.7	Auxiliary building floor drain tanks	8Z7 8Z8	Filled with contaminated water	Outside tank cubicle	8Z1	2 mrem/hr 10 mrem/hr	I-D **	OA3-11B OP3-29
III.3.8	Auxiliary building floor drain pumps	8Z9	Pumping contaminated water	Outside pump cubicle	8Z1	2 mrem/hr	I-D	OA3-13
III.3.9	Vertical pipe tunnel	8Z16	Pipes containing waste gas	Area outside pipe tunnel	8Z1	2 mrem/hr	I-D	x(OA4-16B)

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.3.10	Charging pumps	8Z20 8Z25	Normal Operation	Outside pump cubicle	8Z1	2 mrem/hr	I-D	IA3-5 2A3-5
			Pumping sump water 12 hours after LOCA	Outside pump cubicle	8Z1	100 mrem/hr	-	-
III.3.11	Chemical/Regeneration waste drain tank	8Z4	Demineralizer regenerants	Outside cubicle	8Z1	2 mrem/hr (10 mrem/hr)	I-D **	OA3-16 OP3-35
III.3.11.A	Chemical/Regeneration waste drain tank removable slab	9Z5	Demineralizer regenerate	Area above cubicle	8Z1	2 mrem/hr	I-D	OA3-18B
III.3.12	Chemical/Regeneration waste drain pumps	8Z5	Pump demineralizer regenerants	Outside cubicle	8Z1	2 mrem/hr	I-D	OA3-15
III.3.13	Pipe tunnel El. 375 ft - 6 in.	11Z4	(1) radioactive water, normal operation	Area outside pipe tunnel	8Z1	2 mrem/hr	I-D	OA4-28
			(2) contaminated piping at 12 hours after LOCA	Area outside pipe tunnel	8Z1	100 mrem/hr	-	-
III.3.14	Pipe tunnels El. 374 ft - 6 in.	11Z6	Normal operation	Area outside tunnel	8Z1	2 mrem/hr	I-D	OA3-11A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.4	Elevation 383 ft - 0 in.							
III.4.1	Anion and cation demineralizers	7Z10	Contaminated demineralizer	Outside deminera-lizer cubicle	7Z1	2 mrem/hr 10 mrem/hr	I-D **	OP4-25
III.4.2	Blowdown mixed bed demineralizers	7Z10	Contaminated demineralizer	(1) valve aisle	7Z12	4 mrem/hr	III	xxx
				(2) valve opera-ting area	7Z20	4 mrem/hr	II-A	OA4-5
III.4.3	Radwaste mixed bed demineralizer	7Z11	Contaminated demineralizer	(1) valve aisle	7Z12	4 mrem/hr	III	xxx
				(2) valve opera-ting area	7Z20	4 mrem/hr	II-A	x(OA4-5)
III.4.4	Anion filters	10Z42	Contaminated filter	(1) area outside filter cubicle	7Z1	2 mrem/hr	I-D	x(OA4-5)
				(2) valve opera-ting area	7Z20	4 mrem/hr	II-A	x(OA4-5)
				(3) pipe tunnel	7Z13	15 mrem/hr	III	xxx
III.4.5	Cation filters	10Z41	Contaminated filter	(1) pipe tunnel	7Z13	15 mrem/hr	III	xxx
				(2) valve opera-ting area	7Z20	4 mrem/hr	II-A	x(OA4-5)

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.4.6	Blowdown mixed bed demineralizer after filters, and rad-waste filters	10Z38	Contaminated filter	(1) pipe tunnel	7Z13	15 mrem/hr	III	xxx
				(2) valve operating area	7Z20	4 mrem/hr	II-A	x(OA4-5)
III.4.7	Recycle evaporator condensate filter	10Z43	Contaminated filter	(1) pipe tunnel	7Z15	15 mrem/hr	III	xxx
				(2) valve operating area	10Z36	4 mrem/hr	II-A	x(OA4-7)
III.4.8	Recycle evaporator filter valve aisles	7Z14	Contaminated water in pipes and valves	(1) area outside valve aisles	7Z1	2 mrem/hr	I-D	OA4-6 OA4-8 OA4-11B
				(2) Valve operating aisle	7Z20 10Z36	4 mrem/hr	II-A	x(OA4-5) x(OA4-7) OA4-11A
III.4.9	Recycle evaporator feed filters	10Z43	Contaminated filter	(1) pipe tunnel	7Z15	15 mrem/hr	III	xxx
				(2) valve operating area	10Z36	4 mrem/hr	II-A	x(OA4-7)
III.4.10	Seal water return filters	10Z43	Contaminated filter	(1) pipe tunnel	7Z15	15 mrem/hr	III	xxx
				(2) valve operating area	10Z36	4 mrem/hr	II-A	OA4-7

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.4.11	Seal water injection filters	10Z43	Contaminated filter	(1) pipe tunnel	7Z16	15 mrem/hr	III	xxx
				(2) valve operating area	10Z36	4 mrem/hr	II-A	x(OA4-7)
III.4.12	Reactor coolant filters	10Z43	Contaminated filter	(1) pipe tunnel	7Z15	15 mrem/hr	III	xxx
				(2) valve operating area	10Z36	4 mrem/hr	II-A	x(OA4-7)
III.4.13	Vertical HVAC Pipe Tunnel	7Z40	Contaminated water in piping and airborne in duct	Area outside tunnel	7Z1	2 mrem/hr	I-D	OA4-29
III.4.14	Spent fuel pit and skimmer filters	10Z43	Contaminated filter	(1) pipe tunnel	7Z16	15 mrem/hr	III	xxx
				(2) Outside top of filter cubicles	6Z1	2 mrem/hr	I-D	OA5-23
III.4.15	Blowdown prefilters	10Z40	Contaminated filter	(1) area outside filter cubicle	7Z1	2 mrem/hr	I-D	OA4-8
				(2) valve operating area	7Z20	4 mrem/hr	II-A	x(OA4-5)
				(3) pipe tunnel	7Z15	15 mrem/hr	III	xxx

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.4.16	Auxiliary building floor drain filter	10Z47	Contaminated filter	(1) pipe tunnel	7Z13	15 mrem/hr	III	xxx
				(2) valve operating area	7Z20	4 mrem/hr	II-A	x(OA4-5)
III.4.17	Auxiliary building equipment drain filter	10Z46	Contaminated filter	(1) pipe tunnel	7Z13	15 mrem/hr	III	xxx
				(2) valve operating area	10Z36	4 mrem/hr	II-A	x(OA4-7)
III.4.18	Regeneration waste drain filter	10Z45	Contaminated filter	(1) area outside filter cubicle	7Z1	2 mrem/hr	I-D	x(OA4-10)
				(2) pipe tunnel	7Z16	15 mrem/hr	III	xxx
III.4.19	Chemical drain filter	10Z45	Contaminated filter	(1) area outside filter cubicle	7Z1	2 mrem/hr	I-D	OA4-10
				(2) pipe tunnel	7Z16	15 mrem/hr	III	xxx
III.4.20	Drumming stations and pipe tunnel	7Z5 7Z7 11Z14	Operation of drum processing system	(1) outside drumming station	7Z1	2 mrem/hr	I-D	OA4-14
				(2) maintenance aisle	7Z3	4 mrem/hr	II-A	OA4-13

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.4.21	Drumming station conveyor tunnel	7Z2	(1) Transporting of drums filled with contaminated material	Outside tunnel at 24-wall and N-wall	7Z43	1 mrem/hr 5 mrem/hr	I-C **	OP4-27
			(2) Drumming station operation	Radwaste and shutdown control rooms	7Z43	1 mrem/hr	I-C	OA4-15A OA4-15B
III.4.22	Letdown heat exchangers and seal water heat exchangers	7Z27 7Z30 7Z34 7Z37 7Z31 7Z38	Reactor coolant in tube side of heat exchangers	(1) outside heat exchanger cubicles	7Z1	2 mrem/hr	I-D	IA4-1 2A4-1 1A4-4 2A4-4
				(2) valve operating area	7Z29 7Z36	4 mrem/hr	II-A	1A4-3 2A4-3
				(3) valve aisle	7Z28 7Z35	15 mrem/hr	III	xxx
III.4.23	RHR heat exchangers	7Z26 7Z33 7Z32 7Z39	(1) 4 hours after shutdown	Outside cubicle	7Z1	2 mrem/hr	[I-D]	N/A
			(2) 12 hours after LOCA	Outside heat exchangers cubicle	7Z1	100 mrem/hr	-	-

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
			(3) 2 hours after a gap release accident	Outside cubicle	7Z1	5 mrem/hr	-	-
III.4.24	Pipe tunnels	7Z13 7Z15 7Z16	Pipes containing radioactive water, normal operation	(1) area outside pipe tunnels	7Z1	2 mrem/hr	I-D	x(OA4-9) x(OA4-10)f
				(2) valve aisles	7Z12 7Z14 7Z17	4 mrem/hr	III	xxx
			12 hours after LOCA	Area outside tunnel	7Z1	100 mrem/hr	-	-
III.4.25	Vertical waste gas tunnel	7Z41	Normal operation	Area outside tunnel	7Z43	1 mrem/hr	I-C	OA4-16B
III.4.26	Pipe tunnel El. 394 ft - 6 in.	11Z3	(1) radioactive water pipes, normal operation	Area outside pipe tunnel	7Z1	2 mrem/hr	I-D	ORE-AR008 Area Rad Monitor
			(2) 12 hours after LOCA	Area outside pipe tunnel	7Z1	100 mrem/hr	-	-
III.4.27	Pipe tunnel El. 394 ft - 0 in.	11Z5	Contaminated water and sludge	Area outside tunnel	7Z1	2 mrem/hr 10 mrem/hr	I-D **	OP4-27
III.5	Elevation 401 ft - 0 in.							
III.5.1	Primary sample room	6Z14	Radioactive samples	Outside room	6Z1	2 mrem/hr	I-D	x(OA5-10A)

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.5.2	Sample heat exchangers	6Z18	Cooling radio-active samples	(1) primary sample room	6Z14	4 mrem/hr	II-A	1A5-11
				(2) outside sample room	6Z1	2 mrem/hr	I-D	OA5-10A
				(3) entrance to sample cooler	6Z14	4 mrem/hr	II-A	OA5-10B
III.5.3	Dumbwaiter	TML	Radioactive samples	Outside shaft	6Z14	4 mrem/hr	II-A	2A5-11
III.5.4	Thermal regenera-tion demineralizers	10Z31	Contaminated demineralizer	Pipe tunnel	6Z26 6Z27	15 mrem/hr	III	xxx
III.5.5	Recycle evaporators condensate demineralizer	10Z35	Contaminated demineralizer	(1) area outside	6Z1	2 mrem/hr	I-D	1,2A5-5
				(2) pipe tunnels	6Z26 6Z27	15 mrem/hr	III	xxx
III.5.6	Recycle evaporator feed demineralizers	10Z35	Contaminated demineralizer	Pipe tunnel	6Z26 6Z27	15 mrem/hr	III	xxx
III.5.7	Cation bed demineralizers	10Z32	Contaminated demineralizer	Pipe tunnels	6Z26 6Z27	15 mrem/hr	III	xxx
III.5.8	Mixed bed demineralizers	10Z33	Contaminated demineralizer	Pipe tunnels	6Z26 6Z27	15 mrem/hr	III	xxx

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.5.9	Spent fuel pit demineralizer	10Z34	Contaminated demineralizer	Pipe tunnels	6Z26 6Z27	15 mrem/hr	III	xxx
III.5.10	Radwaste evaporator surface condensers and feed pumps	6Z5 6Z6 6Z7	Operation of evaporators	Area outside surface condenser cubicles	6Z1	2 mrem/hr	I-D	OA5-15A OA5-15B OA5-15C
III.5.11	Boric acid tanks	6Z13	Contaminated water	Area outside tank cubicle	6Z1	2 mrem/hr	I-D	OA5-12 OA5-13
III.5.12	Vertical pipe	6Z36	(1) pipes containing waste gas sources	Area outside pipe tunnel	6Z2	2 mrem/hr	I-D	x(OA4-16B)
		6Z33 6Z34	(2) HVAC and radioactive pipes	Area outside pipe tunnel	7Z1	2 mrem/hr	I-D	x(OA4-29)
III.5.13	Laundry drain and laundry drain tank filter	6Z20 6Z21	Normal operation	Area outside tank and filter	6Z1	2 mrem/hr	I-D	OA5-8
III.5.14	Pipe tunnel	6Z26 6Z27	Radioactive water normal operation	(1) area outside pipe tunnel	6Z1	2 mrem/hr	I-D	1A5-5 2A5-5
				(2) valve aisle	6Z25 6Z28	15 mrem/hr	III	xxx
			12 hours after LOCA	Area outside pipe tunnel	6Z1	100 mrem/hr	-	-

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.5.15	Pipe penetration areas	6Z23 6Z32	Normal operation radioactive water in pipes	Area outside penetration area	6Z1	2 mrem/hr	I-D	1A5-1 2A5-1
			12 hours after LOCA	Area outside	6Z1	100 mrem/hr	-	-
III.5.16	Spent resin and concentrates pumps	6Z3	(1) spent resins and evaporator concentrates	(1) outside pump cubicle	6Z1	2 mrem/hr 10 mrem/hr	I-D **	OP5-37
				(2) valve aisle	6Z4	15 mrem/hr	III	xxx
			(2) radioactive pipes and valve	Valve operating area	6Z1	2 mrem/hr	I-D	OA5-18
III.5.17	Calibration rooms	6Z30	Calibration of instruments	(1) outside Room shielded	6Z1	2 mrem/hr	I-D	OA5-20 OA5-21 OA5-22
				(2) containment roof stairs	6Z35	2 mrem/hr	I-D	x(OA5-20)
				(3) interior entrance hall	6Z38	4 mrem/hr	II-A	OA5-6
III.6.	Elevation 426 ft - 0 in.							
III.6.1	Laundry room	5Z10	Operation of laundry	Area outside laundry room	5Z1	2 mrem/hr	I-D	OA6-9 OA6-10

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
		5Z10	Normal operation (includes air-borne)	Area inside laundry room	5Z10	4 mrem/hr	II-A	OA6-8
		5Z11	Heavily contami-nated laundry	Hamper storage entrance	5Z10	4 mrem/hr	II-A	OA6-7
III.6.2	Hot lab	5Z13	Processing radioactive samples	Area outside room	5Z1	2 mrem/hr	I-D	OA6-12 OA6-19
III.6.3	Radwaste evaporators	5Z24 5Z26 5Z27	Operation of evaporators	Area outside evaporator cubicles	5Z1	2 mrem/hr	I-D	OA6-24A OA6-24B OA6-24C
III.6.4	Waste gas compressor packages	5Z21 5Z22	Fission product gases	Outside com-pressor cubicle	5Z1	2 mrem/hr	I-D	OA6-25A
III.6.5	Automatic gas analyzer	5Z23	Analyzing fission product gases	(1) outside analyzer cubicle	5Z1	2 mrem/hr	I-D	OA6-25B
		5Z28		(2) outside valve room	5Z1	2 mrem/hr	I-D	OA6-39
III.6.6	Concentrates holding tank	5Z20	Storage of evaporator concentrates	Area outside cubicle	5Z1	2 mrem/hr	I-D	OA6-26B
III.6.7	Spent resin storage tank	5Z19	Radioactive resins	(1) area outside tank cubicle	5Z1	2 mrem/hr	I-D	OA6-26A

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.6.8	Volume control tank	5Z6 5Z7	Radioactive water	(1) area outside tank cubicle	5Z1	2 mrem/hr	I-D	1A6-5 2A6-5
				(2) valve aisle	5Z8 5Z9	15 mrem/hr	III	xxx
		5Z8 5Z9	Pipe with radioactive water	Valve operating area	5Z1	2 mrem/hr	I-D	1A6-6 2A6-6
III.6.9	Mask cleaning room (Byron)	5Z16	Decontamination and storage	Area outside room	5Z1	2 mrem/hr	I-D	OA6-15A
III.6.9A	Mask cleaning room (Braidwood)	5Z32	Decontamination and storage	Area outside room	5Z1	2 mrem/hr	I-D	OA6-13
III.6.10	Decontamination Facility	5Z31	Equipment decon-tamination and storage	Area outside room	5Z1	2 mrem/hr	I-D	OA6-4
III.7	Elevation 451 ft - 0 in.							
III.7.1	Control room area	3Z7 3Z12 3Z14	(1) normal operation	Inside control room area	3Z1	0.2 mrem/hr	V	OA7-2 OA7-5
		TML	(2) LOCA direct plus immersion dose	Inside control room area	3Z1	<5 rem during the 30 days post-LOCA	-	-

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
III.7.2	Purge room area	18Z10	Normal plant operation hot spot applies to VQ penetrations	Inside purge room area between el. 451 ft. 0 in. and 476 ft. - 6 in.	3Z12 3Z14	2 mrem/hr 10 mrem/hr	I-D **	1A7-9 2A7-9
III.7.3	Auxiliary building HVAC charcoal filter area	3Z13	Contaminated VA charcoal filters	Corridor outside charcoal filter banks	3Z12 3Z14	2 mrem/hr	I-D	OA7-8
III.8	Areas separating main portion of auxiliary building from containment buildings							
III.8.1	Elevation 346 ft-0 in. Elevation 364 ft-0 in. Elevation 383 ft-0 in.	TML	(1) normal and shutdown operation	Outside separation area in main portion of auxiliary building	9Z1 8Z1	2 mrem/hr	I-D	1,2A2-12 1,2A3-10 OA4-12
			(2) radioactive pipes 12 hours after LOCA	Same as above	9Z1 8Z1 7Z1	100 mrem/hr	-	-
III.8.2	Elevation 401 ft-0 in.	See III.5.16						
III.8.3	Elevation 426 ft-0 in.	See I.C.3						
III.8.4	Elevation 439 ft-0 in.	See I.C.3.						
III.9	Auxiliary building	TML	Normal operation	Auxiliary building roof	1Z1	0.5 mrem/hr	I-B	1,2A9-1 1,2A9-2 OA9-3

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
IV.	Radwaste Building							
IV.1	Elevations 397'-0" and 401'-0"							
IV.1.1	Fluid bed dryer	19Z6	VR system processing waste	Area outside room entrance	19Z2 10Z22	2 mrem/hr 10 mrem/hr	I-D **	R5-10A R5-6
IV.1.2	Incinerator	19Z7	Incinerator processing dry active waste	Area outside room	19Z12	2 mrem/hr 10 mrem/hr	I-D **	R5-12
IV.1.3	Scrubber and feed pumps	19Z4 19Z13	VR system processing waste	Area outside cubicles	19Z12	2 mrem/hr	I-D	R5-13
IV.1.4	Feed tank recirculation pumps	19Z10	Radioactive sludge and water	Area outside pump cubicle	19Z2	2 mrem/hr	I-D	R5-10A
IV.1.5	Radwaste drumming station	19Z9	Waste filled drum	Area outside drumming cubicle	19Z22	2 mrem/hr	I-D	R5-6
IV.1.6	Drum swipe and labeling station	19Z23	Waste filled drum	Area in front of station	19Z14	2 mrem/hr 10 mrem/hr	I-D **	R5-4
IV.1.7	Drum storage areas	19Z27 19Z26	Waste filled drums	(1) truck bay	19Z22	2 mrem/hr	I-D	R5-7 R5-9
				(2) loading platform	19Z14	2 mrem/hr	I-D	R5-5

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
IV.1.8	Truck bay	19Z22	Loading Waste filled drums	(1) Radwaste bldg entrance	19Z14	2 mrem/hr	I-D	R5-1
				(2) RB control room	19Z16	1 mrem/hr	I-C	R5-2
IV.2	Elevation 410 ft - 0 in.							
IV.2.1	Fluid bed dryer	19Z6	VR system processing waste	Area outside room	20Z1	0.2 mrem/hr	I-A	S6-3
IV.2.2	Drumming station	19Z9	Waste filled drum	Area above drumming unit	20Z1	0.2 mrem/hr	I-A	S6-5
IV.2.3	Feed tanks	19Z1	Radioactive sludge and water	Pump room entrance	19Z10	15 mrem/hr	III	***
IV.2.4	Gas/Solid separator	19Z5	VR system in operation	Operator area	19Z3	4 mrem/hr	II-A	R6-3
IV.2.5	VR system charcoal filter	19Z8	VR system in operation	Entrance to incinerator room	19Z3	4 mrem/hr	II-A	R6-4
IV.2.6	Recirculation pump skid	19Z10	Normal operation	Shredder area	19Z2	2 mrem/hr	I-D	R5-10B
IV.2.7	Transfer product hopper cubicle	19Z19	Normal operation	Area outside cubicle	19Z12	2 mrem/hr	I-D	R5-14
IV.2.8	Dryer feed tunnel	TML	Normal operation	Area outside tunnel	19Z12	2 mrem/hr	I-D	R5-15

TABLE 12.3-2 (Cont'd)

REFERENCE NUMBER	LOCATION OR SOURCE	SOURCE ZONE NO.	DESIGN CONDITION	PROTECTED AREA	PROTECTED ZONE NO.	DESIGN DOSE RATE*	ZONE DESIGNATION	RADIATION BASE POINT (RBP)
V.	Turbine building							
V.1	Turbine building	TML	Normal operation	Inside building	TML	1 mrem/hr	I-C	1,2T5-3
V.2	Safety valve enclosure to containment	16Z5 16Z6	Normal operation	Inside enclosure	16Z7 16Z8	4 mrem/hr	II-A	-
V.3	Condensate polishing area	21Z1	When polishers are used	Turbine bldg. doorway	TML	1 mrem/hr	I-C	OT5-7

Notes to Table

1. The zone designations are discussed in Table 12.3-1.
2. The "Zone Numbers" are shown on Figures 12.3-5 through 12.3-26 Drawings M-24-1 through M-24-23.
3. The design dose rate values given in this table are based on the design criteria for a solid shielding unit and does not reflect the impact of penetrations and voids except for a few protected areas that have ** indicated in the "Zone Designation" column. The exceptions indicate a few select hot spots, but this criteria can be applied to every protected area listed above. A detailed explanation of the hot spot criteria can be found in Subsection 12.3.2.3.
4. Verification of shielding walls and slabs that separate two radiation areas is not practical because the radiation fields(s) coming through the wall during normal operation will be masked by the radiation field in the protected area. Therefore, it is not practical to have a radiation base point for the area. The design dose rate is specified to protect maintenance operations. The "xx" and "xxx" in the radiation base point column identify these special types of protected areas.

TABLE 12.3-3

AREA RADIATION MONITORS

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
ORE-AR001	Aux. Bldg. El. 346	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR002	Aux. Bldg. El. 346	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP approved procedures	
ORE-AR003	Aux. Bldg. El. 346	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR004	Aux. Bldg. El. 364	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR005	Aux. Bldg. El. 364	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR006	Aux. Bldg. El. 364	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR007	Aux. Bldg. El. 383	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR008	Aux. Bldg. El. 383	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR009	Aux. Bldg. El. 383	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR010	Aux. Bldg. El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR011	Aux. Bldg. El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR012	Aux. Bldg. El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR013	Aux. Bldg. El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR014	Aux. Bldg. El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR015	Aux. Bldg. El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
ORE-AR016	Aux. Bldg. El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR017	Aux. Bldg. El. 451	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
ORE-AR031	Primary Sample Room	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR032	High Level Lab El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR035	Drumming Station El. 383	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR037	Fuel Handling Bldg. El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR038	Fuel Handling Bldg. El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR039	Fuel Handling Bldg. Crane Trolley El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	Interlock Crane Raise Circuit
ORE-AR041	Radwaste Bldg. Low Level Storage El. 410	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR042	Radwaste Bldg. El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR043	Radwaste Bldg. Truck Bay El. 397	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR044	Radwaste Bldg. Low Level Storage El. 401	1-100,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
ORE-AR045	Radwaste Bldg. High Level Storage El. 401	1-100,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	High back-ground area
ORE-AR046	Volume Reduction Area El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR047	Volume Reduction Area El. 401	1-100,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR048	Volume Reduction Area El. 401	1-100,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR049	Volume Reduction Area El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR050	Volume Reduction Area El. 401	1-100,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
1RE-AR001	Containment El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
2RE-AR001	Containment El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
1RE-AR002	Containment El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
2RE-AR002	Containment El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
1RE-AR003	Incore Seal Table El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
2RE-AR003	Incore Seal Table El. 401	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
1RE-AR010	Main Control Room El. 451	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
2RE-AR010	Main Control Room El. 451	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures	
1RE-AR011	Containment Fuel Handling Incident El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	≤ 2 x background in the Containment Building at RTP	Redundant with 1RE-AR012
2RE-AR011	Containment Fuel Handling Incident El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	≤ 2 x background in the Containment Building at RTP	Redundant with 2RE-AR012
1RE-AR012	Containment Fuel Handling Incident El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	≤ 2 x background in the Containment Building at RTP	
2RE-AR012	Containment Fuel Handling Incident El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	≤ 2 x background in the Containment Building at RTP	
0RE-AR055	Fuel Building Fuel Handling Incident El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	≤ 5 mR/hr	Redundant with 0RE-AR056
0RE-AR056	Fuel Building Fuel Handling Incident El. 426	0.1-10,000 mR/hr	GM	0.08-3 MeV	≤ 5 mR/hr	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
1RE-AR013	Volume Control Tank Cubicle El. 426	0.1-10,000 mR/hr	IC	0.08-3 MeV	per RP-approved procedures*	High back-ground cubicle
2RE-AR013	Volume Control Tank Cubicle El. 426	0.1-10,000 mR/hr	IC	0.08-3 MeV	per RP-approved procedures*	High back-ground cubicle
1RE-AR020	High Range Containment El. 514'-8" (Actual detector El.)	10^0-10^8 R/hr	IC		Per E-Plan EALs	
2RE-AR020	High Range Containment El. 514'-8" (Actual detector El.)	10^0-10^8 R/hr	IC		Per E-Plan EALs	
1RE-AR021	High Range Containment El. 514'-8" (Actual detector El.)	10^0-10^8 R/hr	IC		Per E-Plan EALs	
2RE-AR021	High Range Containment El. 514'-8" (Actual detector El.)	10^0-10^8 R/hr	IC		Per E-Plan EALs	
ORE-AR073	TSC Monitor Room El. 435	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
ORE-AR074	TSC Health Physics Office El. 451	0.1-10,000 mR/hr	GM	0.08-3 MeV	per RP-approved procedures*	
1RE-AR022A	Main Steamline 1A	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 1RE-AR023A

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
1RE-AR022B	Main Steamline 1B	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 1RE-AR023B
1RE-AR022C	Main Steamline 1C	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 1RE-AR023C
1RE-AR022D	Main Steamline 1D	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 1RE-AR023D
2RE-AR022A	Main Steamline 2A	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 2RE-AR023A
2RE-AR022B	Main Steamline 2B	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 2RE-AR023B
2RE-AR022C	Main Steamline 2C	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 2RE-AR023C
2RE-AR022D	Main Steamline 2D	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	Redundant with 2RE-AR023D
1RE-AR023A	Main Steamline 1A	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
1RE-AR023B	Main Steamline 1B	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
1RE-AR023C	Main Steamline 1C	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
1RE-AR023D	Main Steamline 1D	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
2RE-AR023A	Main Steamline 2A	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
2RE-AR023B	Main Steamline 2B	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
2RE-AR023C	Main Steamline 2C	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
2RE-AR023D	Main Steamline 2D	0.1-10,000 mR/hr	GM	0.02-3 MeV	≤ 3X background	
1RE-AR024A	Main Steamline 1A & 1D Pen. R13	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
1RE-AR024B	Main Steamline 1B & 1C Pen. R20	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR024A	Main Steamline 2A & 2D Pen. R41	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR024B	Main Steamline 2B & 2C Pen. R34	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
1RE-AR025A	Piping Penetration El. 364' - R5	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
1RE-AR025B	Piping Penetration El. 364' - R7	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR025A	Piping Penetration El. 364' - R28	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR025B	Piping Penetration El. 364' - R26	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
1RE-AR026A	Piping Penetration El. 383' - R5	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
1RE-AR026B	Piping Penetration El. 383' - R7	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR026A	Piping Penetration El. 383' - R28	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR026B	Piping Penetration El. 383' - R26	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
1RE-AR027A	Piping Penetration El. 401' - R5	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
1RE-AR027B	Piping Penetration El. 401' - R7	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	

TABLE 12.3-3 (Cont'd)

RADIATION DETECTOR NO	SERVICE	RANGE	TYPE OF DETECTOR	ENERGY RANGE	SETPOINT	REMARKS
2RE-AR027A	Piping Penetration El. 401' - R28	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	
2RE-AR027B	Piping Penetration El. 401' - R26	0.1-10,000 R/hr	IC	0.1-3 MeV	per RP-approved procedures	

*Local indication and alarm provided.

TABLE 12.3-4

PARAMETERS USED IN THE CALCULATION

OF THE PRIMARY SHIELD THICKNESS

CORE POWER RATING	
Total Core Thermal (Mw)	3565
Power Density (watts/cc)	109.2
CORE EFFECTIVE DIMENSIONS (cm)	
Height	365.76
Diameter	337.09
CORE VOLUME FRACTIONS	
UO_2	.3052
Zirconium	.0943
Stainless Steel	.0053
Inconel	.0043
Water	.5909

REACTOR DIMENSIONS

REGION	MATERIAL	OUTSIDE RADIUS (cm)	THICKNESS (cm)
Core	(see above)	168.545	168.545
Baffle	SS	172.402	2.858
Water	H_2O	187.960	16.558
Barrel	SS	193.675	5.715
Shield Panel	Void*	200.667	6.985
Water	H_2O	219.71	19.05
Pressure Vessel	CS	241.618	101.7
Void + Neutron	-	-	-
Detector Cavity	Air	343.318	-
Primary Shield	Ordinary concrete	517.208	173.89

* The worst case radial traverse was chosen for the ANISN model which travels in-between the intermittent shield panels and goes through a neutron detector cavity.

TABLE 12.3-4 (Cont'd)

CORE RADIAL SOURCE DISTRIBUTION

FRACTION OF CORE RADIUS	RADIAL POWER DISTRIBUTION
.1	1.0
.2	0.999
.3	0.998
.4	0.996
.5	0.992
.6	0.975
.7	0.942
.8	0.858
.9	0.67
1.0	0.563

TABLE 12.3-5

CORE FISSION SOURCE FOR

PRIMARY SHIELD CALCULATION

GROUP	UPPER ENERGY (MeV)	CORE TOTAL NEUTRON SOURCE (n/cc-sec)
1	15	1.32×10^{9}
2	12.2	7.56×10^{9}
3	10.0	2.94×10^{10}
4	8.18	1.25×10^{11}
5	6.36	2.86×10^{11}
6	4.96	4.09×10^{11}
7	4.06	9.08×10^{11}
8	3.01	7.53×10^{11}
9	2.46	1.98×10^{11}
10	2.35	1.02×10^{12}
11	1.83	1.85×10^{12}
12	1.11	1.68×10^{12}
13	0.55	1.15×10^{12}
14	0.111	1.31×10^{11}
15	0.003	0.0
TOTAL		8.53×10^{12}

TABLE 12.3-6

SHIELDING DESIGN-BASIS GEOMETRY FOR SHIELDING THICKNESS CALCULATIONS

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Reactor	fission spectrum	A	UO_2, Zr, SS Inconel, H_2O	4.4	C	R = 5.5 (core)
Steam Generator	Table 12.2-1, 12.2-2, 12.2-3	I	H_2O, Fe	.756	C	R = 11.3 H = 28.3
Pressurizer	Table 12.2-4, 12.2-5, 12.2-6	I	H_2O	.68	C	R = 3.5 H = 38 (normal water level)
Reactor Coolant Pumps and Piping	Table 12.2-1, 12.2-2	I	H_2O	.68	C	R = 1.25 H = as required
Reactor Coolant Drain Tank	Table 12.2-2	I	H_2O	1.0	C	R = 1.5 H = 7.4
Regenerative Heat Exchanger	Table 12.2-1	I	H_2O	1.0	C	R = .83 H = 18
Excess Letdown Heat Exchanger	Table 12.2-1	I	H_2O	1.0	C	R = .75 H = 14
Incore Detectors and Drive Wires	Table 12.2-26, 12.2-27	I	Fe	7.87	L	H = 15

TABLE 12.3-6 (Cont'd)

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Fuel Assembly in Refueling Cavity	Table 12.2-23 adjusted for one fuel assembly (4-day decay)	I	UO_2, Zr, SS Inconel, H_2O	4.4	S	W = .7 L = .7 H = 13.25
Volume Control Tank	Table 12.2-8	I	H_2O	liquid: 1.0 vapor: 0.001293	C	R = 3.25, H = 3.6 R = 3.25, H = 5.4
Recycle Holdup Tank	Table 12.2-9, 12.2-10	I	H_2O	liquid: 1.0 vapor: 0.001293		R = 14, H = 12.5 R = 14, H = 14.5
Recycle Evaporator	Table 12.2-11	I	H_2O	1.0	C	R = 1.8 H = 9.9
Recycle Evaporator Vent Condenser	Table 12.2-11	I	H_2O	1.293E-3	C	R = .33 H = .8
RHR Heat Exchanger	Table 12.2-12	I	H_2O	1.0	C	R = 3.6 H = 28
RHR Pump and Piping	Table 12.2-12	I	H_2O	1.0	C	R = .58 H = 17
Mixed Bed Demineralizer	Table 12.2-13	I/M*	H_2O	1.0	C	R = 1.083 H = 8
Cation Bed Demineralizer	Table 12.2-14	I	H_2O	1.0	C	R = 1.33 H = 3.6
Thermal Regeneration Demineralizer	Table 12.2-15	I	H_2O	1.0	C	R = 1.0 H = 5.6

*For Braidwood

TABLE 12.3-6 (Cont'd)

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Recycle Evaporator Feed Demineralizer	Table 12.2-16	I	H_2O	1.0	C	R = 1.083 H = 8
Recycle Evaporator Condensate Demineralizer	Table 12.2-17	I	H_2O	1.0	C	R = 1.083 H = 8
Spent Fuel Pit Demineralizer	Table 12.2-18	I	H_2O	1.0	C	R = 1.083 H = 8
Reactor Coolant Filter	Table 12.2-19	I	H_2O	.38	C	R = .28 H = 1.6
Seal Water Return Filter	Table 12.2-20	I	H_2O	.38	C	R = .28 H = 1.6
Recycle Evaporator Feed Filter	Table 12.2-20	I	H_2O	.38	C	R = .28 H = 1.6
Spent Fuel Pit Filter	Table 12.2-20	I	H_2O	.38	C	R = .28 H = 1.6
Spent Fuel Pit Skimmer Filter	Table 12.2-20	I	H_2O	.38	C	R = .28 H = 1.6
Seal Water Injection Filter	Table 12.2-20	I	H_2O	.38	C	R = .11 H = 1.7
Recycle Evaporator Concentrates Filter	Table 12.2-21	I	H_2O	.38	C	R = .11 H = 1.7

TABLE 12.3-6 (Cont'd)

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Recycle Evaporator Condensate Filter	Table 12.2-22	I	H_2O	.38	C	R = .1 H = 1.7
Waste Gas Decay Tanks	Table 12.2-22	I	H_2O	0.001293	cylindrical	R = 4.25 H = 10.6
Spent Fuel Storage Area	5/3 core fission products	I	UO_2, Zr, SS Inconel, H_2O	2.75	S	W = 11.88 L = 16.25 H = 62
(Transfer of) One Spent Fuel Assembly	Table 12.2-23 adjusted for one fuel assembly (4-day decay)	Q	UO_2, Zr, SS Inconel, H_2O	4.4	S	W = .7 L = .7 H = 13.25
Laundry Drain Tank	Table 12.2-33	I	H_2O	1.0	C	R = 3.25 H = 16.5
Blowdown Mixed Bed Demineralizer	Table 12.2-35	I	H_2O	1.0	C	R = 2 H = 5.16
Radwaste Mixed Bed Demineralizer	Table 12.2-35	I	H_2O	1.0	C	R = 2 H = 5.16
Concentrates Holding Tank	Table 12.2-36	I	H_2O	1.0	C	R = 5 H = 8.5
Blowdown Prefilter†	Table 12.2-37	I	H_2O	1.0	C	R = H =

TABLE 12.3-6 (Cont'd)

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Blowdown Afterfilter	Table 12.2-37	I	H_2O	1.0	C	R = .25 H =
Radwaste Afterfilter	Table 12.2-37	I	H_2O	1.0	C	R = .25 H =
Turbine Building Equipment Drain Filter	Table 12.2-37	I	H_2O	1.0	C	R = .25 H =
Turbine Building Floor Drain Filter	Table 12.2-37	I	H_2O	1.0	C	R = .25 H = 1.6
Auxiliary Building Equipment Drain Filter	Table 12.2-38	I	H_2O	1.0	C	R = .25 H = 1.6
Auxiliary Building Floor Drain Filter	Table 12.2-38	I	H_2O	1.0	C	R = .25 H = 1.6
Regeneration Waste Drain Filter	Table 12.2-38	I	H_2O	1.0	C	R = .25 H = 1.6
Chemical Drain Filter	Table 12.2-38	I	H_2O	1.0	C	R = .25 H = 1.6
Laundry Drain Filter	Table 12.2-38	I	H_2O	1.0	C	R = .25 H = 1.6
Radwaste Evaporator	Table 12.2-39	I	H_2O	1.0	C	R = 3 H = 15

TABLE 12.3-6 (Cont'd)

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Radwaste Evaporator Surface Condenser		I	H_2O	1.0	C	R = 1.5 H = 11
30,000 gal. Release Tank	Table 12.2-40	I	H_2O	1.0	C	R = 17 H = 8.6
Permeate Sample Tank	Table 12.2-40					
Blowdown Monitor Tank	Table 12.2-41	I	H_2O	1.0	C	R = 8 H = 15.83
Radwaste Evaporator Monitor Tank	Table 12.2-41	I	H_2O	1.0	C	R = 8 H = 15.83
Spent Resin Tank	Table 12.2-43, col. 1	I	H_2O	1.2	C	R = 4.5 H = 10.5
Radwaste Drum Storage	Table 12.2-43, col. 2	I	Table 12.2-44	1.33	S	W = 14.66 L = 18.33 H = 15
Refueling Water Storage Tanks	Table 12.2-25	I	H_2O Concrete	1.0 2.242	C	R = 25 H = 30
Condensate Storage Tank	10^{-3} μCi/cc @1.3 MeV	I	H_2O	1.0	C	R = 22 H = 44

TABLE 12.3-6 (Cont'd)

NAME OF COMPONENT	SOURCE	COMPUTER CODE USED*	SOURCE COMPOSITION	HOMOGENIZED SOURCE DENSITY (gm/cc)	GEOMETRY**	SOURCE DIMENSIONS*** (ft)
Auxiliary Building Charcoal Filters	Table 12.2-45	I	C	.45	S	W = 2.4 L = 16.75 H = 7.5
Steam Jet Air Ejector Vent Filter System	Table 12.2-45	I	C	.45	S	W = 2.4 L = 16.75 H = 7.5

* A = ANISN, I = ISOSHLD, Q = QAD, M=Microshield
** C = cylindrical, S = finite slab, L = line
*** R = radius, H = height, L = length, W = width
**** The permeate sample tank sources are less than or equal to the laundry drain tank sources. Thus, the same shielding requirement was recommended.
† Shielding was determined based on equipment 1/2WX02MA,B (housing-only prefilter vessels).

TABLE 12.3-7

ESTIMATED OCCUPATIONAL RADIATION EXPOSURE DURING DECOMMISSIONING

ALTERNATIVE	MAN-REM
Mothballing	150
Entombment	130
Prompt Dismantling	630
Mothballing with Delayed Dismantling*	150 + 310
Entombment with Delayed Dismantling*	130 + 310

* 104-year delay period before delayed dismantling.

The above information was assembled from Reference 10.

TABLE 12.3-8

DOMINANT RADIOACTIVE ISOTOPES FOR PROMPT DISMANTLING AND DELAYED DISMANTLING

SOURCE	PROMPT DISMANTLING 2 YEARS OF DECAY	DELAYED DISMANTLING 104 YEARS OF DECAY
Vessel and Internals	Fe55, Co60, Ni63	Ni63
Other systems	Co60	Sr90, Cs137

The above information was assembled from Reference 10.

TABLE 12.3-9

SENSITIVITY OF CONTINUOUS AIRBORNE MONITORING SYSTEM

	ISOTOPE	AVERAGE ENERGY (MEV)	GROSS SENSITIVITY (cpm/μCi)	SENSITIVITY (cpm/hr per μCi/cc)	MPC*** (μCi/cc)	DETECTABILITY* FACTOR FOR MPC_a
I.	Particulate Channel (Beta Scintillator)					
	CO-60	0.096	4.69×10^{5}	2.01×10^{12}	9×10^{-9}	200
	SR-90	0.200	1.20×10^{6}	5.10×10^{12}	1×10^{-9}	60
	TC-99	0.085	4.47×10^{5}	1.92×10^{12}	6×10^{-8}	1350
	CS-137	0.171	1.15×10^{6}	4.91×10^{12}	1×10^{-8}	575
II.	Iodine Channel (NaI Spectrometry windowed on I-131 peak)					
	I-131	0.364 (γ)	1.01×10^{5}	4.29×10^{11}	9×10^{-9}	850
III.	Noble Gases Channel (Beta Scintillator)					
	KR-85	0.100	-----	1.84×10^{7}**	1.0×10^{-5}	40
	XE-133	0.250	-----	3.6×10^{7}**	1.0×10^{-5}	80

* The Minimum detectable (activity) concentration is based on a signal count rate at a 95% confidence level as given by the formula in ANSI 13.10-1974 and modified for the GA system as follows:

MDC = 2 $(BCKG/20)^{1/2}$ ÷ Sensitivity, (for BCKG □ 100 cpm)

2 $(BCKG^2/2000)^{1/2}$ ÷ Sensitivity, (for 100 cpm < BCKG < 1×10^5 cpm)

Where BCKG is the total background counting rate (cpm). For the particulates 1905 cpm was used and for iodine and noble gases 100 cpm was used; this criterion will yield an answer that has a 95% statistical confidence level.

** cpm per μCi/cc

*** The term MPC refers to a 10CFR20 limit in effect prior to January 1, 1994.

ATTACHMENT 12.3A

EXAMPLES OF THE APPLICATION OF RADIATION PROTECTION DESIGN FEATURES TO SPECIFIC COMPONENTS

EXAMPLES OF THE APPLICATION OF RADIATION PROTECTION DESIGN FEATURES TO SPECIFIC COMPONENTS

The general principles and concepts of radiation protection design features including shielding to minimize occupational dose are described in the various subsections of 12.3.

The application of these features to the design of specific components is described below.

DEMINERALIZERS

The demineralizers are isolated from their valves, other equipment, and from general access areas. In addition to labyrinth entrances, some demineralizer rooms have removable ceiling hatches. At least one hatch contains a radiation probe hole which is utilized prior to removing the hatch. The metering device attached to the probe is properly calibrated so that operating personnel will have adequate radiation data prior to removing the hatch.

The valves for the demineralizers are located in a separate room. A typical arrangement is shown in Figure 12.3A-1. Valve operator stations located in general access areas are utilized wherever they are practical. Ventilation to the valve room is supplied from the general access area and is exhausted to the demineralizer room and/or the radwaste tunnel. The ventilation exhaust from the demineralizer room goes directly into an adjacent radwaste pipe tunnel.

SAMPLING STATION

Sampling stations can be located singly (inside labyrinth entrances when practical) or can be grouped together in sample panels. The sampling station is located as close to the sampling point as is practical, but not in direct view of a radioactive source. Shielding, drains, and flushing lines are used to reduce occupational radiation exposure whenever it is practicable to do so. A single sampling station and a sample panel are shown in Figure 12.3A-1.

HYDROGEN RECOMBINER

The hydrogen recombiner is a postaccident system. The containment hydrogen recombiners are located in a general access area at elevation 401 feet 0 inch adjacent to column rows 15U and 21U. This location was selected so that each recombiner is close to the containment yet shielded from it. The radiation shielding surrounding the recombiner is designed to protect the area directly adjacent to the recombiner from the postaccident radiation sources and to allow access to the recombiner (for maintenance, removability, and replacement) during the postaccident period as well as during normal station operation.

The recombiners are only to be operated during postaccident conditions and when they are being tested. Therefore, the recombiners will not become radioactive during normal station operation. A removal fence may be used to keep the recombiner removal path clear of traffic and equipment.

Start switches for the recombiners are located in the recombiner controls console. The Unit 1 recombiner control console is located away from the recombiners on elevation 401 feet 0 inch (column row 13/P). The Unit 2 recombiner control console is located away from the recombiners on elevation 439 feet 0 inch (column row 25/Q).

Area radiation monitors (ARMs) are located near the recombiner area so that station personnel will be alerted to high radiation levels. The Unit 1 ARM and recombiner area are shown on Figure 12.3A-2.

EVAPORATORS

The radioactive evaporator equipment is segregated from the remainder of the evaporator equipment. The radioactive equipment is located on an upper level which has only one access (a shielded staircase). Access to the upper level is through a closed door which is utilized in accordance with 10 CFR 20. The lower level contains the evaporator condensing equipment, the radiation monitor panel, and the control panel. This equipment is slightly radioactive (approximately 1 x 10^{-4} times the dose rate of the upper level) and needs to be separated from general access areas. Figure 12.3A-3 shows the layout of the evaporator equipment.

FUEL TRANSFER TUBE

The fuel transfer area is shown in Figures 12.3A-4, and 12.3A-5, and Drawing M-24 Sheet 14 and 16.

The shielding for the fuel transfer tube is based on a peak fuel assembly. This is an assembly that has 1.5 times the average 1000-day burnup. In order to obtain a dose rate of 5 mrads/hr in adjacent areas, 5 feet of ordinary concrete is required. The radiation streaming through the 2-inch expansion gap is reduced by attaching a 5-inch thick, 3-inch wide steel horseshoe shielding collar on the transfer tube sleeve.

The expected doses during fuel transfer are:

elevation 389	< 5 mrem/assembly,
elevation 399	< 5 mrem/assembly, and
tendon tunnel	< 2 mrem/assembly.

There will be zero access to the fuel transfer tube area during periods when fuel is being moved through the tube.

The fuel transfer tube will only be exposed for tube inspection. These inspections will only be scheduled for times when no fuel movement is scheduled. The tube inspection and the replacing of all shielding that was removed will be performed on a priority basis. The entire operation will be completed in the shortest time practical. Key operating personnel, especially the fuel transfer office are informed at the beginning and at the completion of the tube inspection.

12.4 DOSE ASSESSMENT

12.4.1 Estimated Occupancy of Plant Radiation Zones

It is difficult to predict the occupancy of any one zone on an average weekly basis, much less by function. An estimate based on an average yearly occupancy (even if such data existed) would give questionable results in dose calculations because the exposure is related to the operating history, which is highly episodic.

12.4.2 Estimates of Inhalation Doses

Small airborne radioactivity concentrations of radionuclides are expected within the various plant structures. Implementation of the health physics program (Section 12.5) mitigates against any significant ingestion doses to plant personnel.

Radionuclides are a potential hazard because they may be present and released in significant quantities from fluids. For this reason, maximum design-basis radioiodine concentrations have been calculated in the buildings most susceptible to airborne contamination. The assumptions used in calculating these airborne radioactivity concentrations are presented in Subsection 12.2.2.3. The resultant design-basis calculated concentrations are also tabulated in Tables 12.2-46, 12.2-47, and 12.2-48, and the thyroid dose acquired from their inhalation is tabulated in Table 12.4-4.

12.4.3 Objectives and Criteria for Design Dose Rates

The objectives and criteria for design dose rates in various areas are discussed in Section 12.3.

12.4.4 Estimated Annual Occupational Exposures

Many ALARA features have been incorporated into the design of Byron/Braidwood Stations - selection of materials to reduce crud levels, separation of radiation areas, etc. The actual dose rates experienced in most areas are expected to be lower than the respective design dose rates. So a calculation of occupational exposures based on design dose rates would overestimate these exposures in most cases. But either actual or design dose rate calculations would require a knowledge of occupancy factors.

Thus the most reliable prediction of occupational exposures must be based on data from the operating history of other PWRs. Table 12.4-1 shows the annual personnel exposure reported for various PWRs. Much of the reported data for older plants is due to backfitting work, while the multiple-unit plants are not strictly comparable to single-unit plants, due to shared facilities such as radwaste areas. Table 12.4-2 shows reported occupational exposure for six younger multiple-unit PWRs. Averages of total annual occupational exposure and average exposure per person differ in the two tables, as expected.

Table 12.4-2 also shows estimates of annual occupational exposure based on the average values of the total annual occupational exposure and average exposure per person. For older single-unit PWRs, the annual occupational exposure varies from year to year, but generally lies in a range between 60 and 1000 person-rem after the first year (Reference 5). Since Byron/Braidwood Stations are multiple-unit plants, it would be reasonable to assume that the annual occupational exposure is in the upper part of that range. Considering the experience at Zion during the first 4 years of operation, however, 800 to 1200 person-rem is probably an upper bound on the exposure. Thus, an estimate of 800 person-rem is given as the average annual occupational exposure.

Table 12.4-3 gives the reported occupational personnel exposure by work function for various operating plants. For the sake of comparison, data for three mature single-unit PWRs are shown. The estimated annual occupational exposures by work function, also given in Table 12.4-3, are based on the averages for the multiple-unit plants.

Table 12.4-5 compares the occupational exposure for single unit stations, two-unit stations, and Zion. The fifth year radiation dose estimate for Byron/Braidwood Stations is assumed to have the same exposure as the average two-unit station for routine operations and for special maintenance. Byron/Braidwood Stations' exposure will improve on Zion in the areas of routine maintenance and radwaste (see Section 12.3). Byron/Braidwood Stations' refueling exposure is assumed to be slightly above that of Zion because of greater fuel burnup.

12.4.5 Estimated Annual Dose at the Exclusion Area Boundary

The estimated annual dose at the exclusion area boundary (EAB) is given in Subsection 5.2.4 of the environmental report.

12.4.6 Deleted

12.4.7 Dose Reduction Program

Commonwealth Edison utilizes an alternate empirical feedback method of design refinements to reduce radiological exposure and maintain doses ALARA. The areas of greatest total exposure in person-rem are identified, then design feedback is made to ameliorate the conditions such that actual exposure rate in the newer plant will be decreased in similar areas.

An independent review was performed by Commonwealth Edison's corporate radiation protection group, which supports the generating stations. These production support specialists accumulated radiation histories from all of Commonwealth Edison's nuclear stations, and they also evaluated operations and procedures. Based on this data, the AE made the following design changes:

a. a partially mechanized solid radwaste system,

b. probe holes for radiation monitoring of filter cubicles,

c. lifting aids, and

d. improved drain systems.

Additional design features which were made to implement ALARA are described in Subsection 12.1.2.3.

The above process of dose improvement continues into the operation phase where the radiation evaluation program identifies areas where improvements are needed. Engineering support to the station is continuous, and these same feedback mechanisms continue to emphasize ALARA doses.

12.4.8 Radiological Environmental Monitoring Program

The radiological environmental monitoring program (REMP) conducted in the vicinity of the stations has as its objectives:

a. Provide data on measurable levels of radiation and radioactivity in the environment and relate these data to radioactive emissions;

b. Identify changes in the use of nearby offsite areas to ensure adequate surveillance and evaluation of doses to individuals from principal pathways of exposure;

c. Provide environmental surveillance in case of an unplanned release; and

d. Provide year-round monitoring of principal pathway exposure.

The REMP provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.b.2 of Appendix I to 10 CFR 50 and, thereby, supplements the radiological effluents monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of effluent measurements and the modeling of the environmental exposure pathways.

The site specific annex of the Offsite Dose Calculation Manual describes the current REMP and presents the required detection capabilities for environmental sample analyses tabulated in terms of the a priori minimum detectable concentration (MDC). The a priori MDC is a before-the-fact limit representing the capabilities of a measurement system and is not an after the fact limit for a particular measurement.

12.4.9 References

1. NUREG-0109, "Occupational Radiation Exposure at Light Water Cooled Power Reactors, 1969-1975," USNRC, August 1976.

2. "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum, September 1974.

3. Northern States Power Company, Prairie Island Nuclear Generating Plant, Units 1 and 2, Semiannual Operating Report Number 4, January to June 1975, Docket No. 50282-401, August 1975.

4. Northern States Power Company, Prairie Island Nuclear Generating Plant, Units 1 and 2, Semi-Annual Operating Report Number 5, July to December 1975, Docket No. 50282-493, March 1976.

5. Rochester Gas and Electric Corporation, Ginna Nuclear Power Plant Annual Operating Report Number 11, January 1, 1975 through December 31, 1975, Docket No. 50244-497, February 1976.

6. Connecticut Yankee Atomic Power Company, Haddam Neck Plant Annual Operating Report, 1975, Docket No. 50213-484.

7. Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Plant Semiannual Occupational Radiation Exposure Report, Errata, January to June 1975, Docket No. 50309-858, March 1976.

8. Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Plant Semiannual Operating Report, July to December 1975, Docket No. 50309, March 1976.

9. Virginia Electric and Power Company, Surry Power Station, Units 1 and 2, Semiannual Operating Report, July to December 1974, Docket No. 50280, August 1975.

10. Virginia Electric and Power Company, Surry Power Station, Units 1 and 2, Semiannual Operating Report, January to June 1975, Docket No. 50280, August 1975.

11. Virginia Electric and Power Company, Surry Power Station, Units 1 and 2, Semiannual Operating Report, July to December 1975, Docket No. 50280, March 1976.

12. Commonwealth Edison Company, Zion Station, Units 1 and 2, Semiannual Operating Report, July to December 1974, Docket No. 50295.

13. Commonwealth Edison Company, Zion Station, Units 1 and 2, Semiannual Report on Radioactive Waste, Environmental Monitoring and Occupational Personnel Exposure, July to December 1975. Docket No. 50295, February 1976.

14. The occupational exposure sections of the Annual Operating Reports as submitted to the NRC for the following stations: Oconee Units 1, 2, and 3; Point Beach Units 1 and 2; Prairie Island Units 1 and 2; Surry Units 1 and 2; Turkey Point Units 3 and 4; Zion Units 1 and 2.

15. NUREG-0463, "Occupational Radiation Exposure, Tenth Annual Report, 1977," USNRC, October 1978.

TABLE 12.4-1

PERSONNEL EXPOSURE DATA FOR VARIOUS OPERATING PWRs (1,15)*

STATION	YEAR	MEGAWATT-YEAR	MEASURABLY EXPOSED PERSONNEL	TOTAL ANNUAL MAN-REM	AVERAGE EXPOSURE (rem/person)	MAN-REM PER MEGA-WATT-YEAR
Arkansas-1	75	588	147	21	0.14	0.04
850 MWe	76	463	476	289	0.61	0.62
	77	610	601	256	0.43	0.42
Beaver Valley	77	328	331	87	0.26	0.27
852 MWe						
Calvert Cliffs-1	76	751	507	74	0.15	0.10
845 MWe	77	557	2265	547	0.24	0.98
D.C. Cook-1	76	805	395	116	0.29	0.14
1054 MWe	77	546	802	300	0.31	0.55
Fort Calhoun	75	252	469	294	0.63	1.17
457 MWe	76	265	516	313	0.61	1.18
	77	334	535	297	0.56	0.89
Ginna	70	268.5	170	207	1.21	0.77
490 MWe	71	327.8	340	430	1.26	1.31
	72	295.6	677	1032	1.52	3.49
	73	409.5	319	224	0.70	0.55
	74	253.7	884	1225	1.38	4.82
	75	365	685	538	0.78	1.47
	76	248	758	636	0.84	2.56
	77	346	530	401	0.76	1.16
Haddam Neck	69	397.6	138	106	0.77	0.27
575 MWe	70	424.7	734	689	0.94	1.62
	71	502	289	342	1.18	0.68

TABLE 12.4-1 (Cont'd)

STATION	YEAR	MEGAWATT-YEAR	MEASURABLY EXPOSED PERSONNEL	TOTAL ANNUAL MAN-REM	AVERAGE EXPOSURE (rem/person)	MAN-REM PER MEGA-WATT-YEAR
	72	515.6	355	325	0.91	0.63
	73	293	951	697	0.73	2.38
	74	519	550	201	0.37	0.39
	75	494	795	703	0.88	1.42
	76	482	644	449	0.70	0.93
	77	458	894	642	0.72	1.40
Kewaunee	75	401.9	104	28	0.27	0.07
535 MWe	76	405	381	270	0.71	0.67
	77	405	312	140	0.45	0.35
Maine Yankee	73	408.7	782	147	0.14	0.36
790 MWe	74	432.6	619	420	0.68	0.97
	75	542.9	440	319	0.73	0.59
	76	710	244	85	0.35	0.12
	77	587	508	245	0.48	0.42
Oconee 1, 2, 3	74	724	844	517	0.61	0.71
886 MWe x 3	75	1084	829	497	0.60	0.46
	76	1557	1215	1026	0.84	0.66
	77	1485	1595	1329	0.83	0.90
Point Beach 1, 2	73	693.7	501	588	1.17	0.85
497 MWe x 2	74	760	400	295	0.74	0.39
	75	801	339	459	1.35	0.57
	76	855	313	370	1.18	0.43
	77	834	417	430	1.03	0.52

TABLE 12.4-1 (Cont'd)

STATION	YEAR	MEGAWATT-YEAR	MEASURABLY EXPOSED PERSONNEL	TOTAL ANNUAL MAN-REM	AVERAGE EXPOSURE (rem/person)	MAN-REM PER MEGA-WATT-YEAR
Prairie Island 1, 2	74	181.9	150	18	0.12	0.10
530 MWe x 2	75	836	477	123	0.26	0.15
	76	723	818	447	0.55	0.62
	77	867	718	300	0.42	0.35
Robinson	71	295.3	283	364	1.28	1.23
700 MWe	72	580	245	215	0.87	0.37
	73	455	831	695	0.83	1.53
	74	577	853	672	0.78	1.16
	75	501.8	849	1142	1.35	2.28
	76	584	597	715	1.20	1.22
	77	493	634	455	0.72	0.92
San Onofre 1	69	289.8	123	42	0.34	0.14
436 MWe	70	365.9	251	155	0.61	0.42
	71	362	121	50	0.41	0.14
	72	372	326	256	0.78	0.69
	73	273.7	878	329	0.37	1.20
	74	377.8	219	71	0.32	0.19
	75	389	424	292	0.69	0.75
	76	297	1330	880	0.66	2.96
	77	266	985	847	0.86	3.18
Surry 1, 2	73	714	936	152	0.16	0.21
823 MWe x 2	74	718	1715	884	0.52	1.23
	75	1079	1948	1649	0.85	1.53
	76	928	2753	3165	1.15	3.41
	77	1082	1860	2307	1.24	2.13

TABLE 12.4-1 (Cont'd)

STATION	YEAR	MEGAWATT-YEAR	MEASURABLY EXPOSED PERSONNEL	TOTAL ANNUAL MAN-REM	AVERAGE EXPOSURE (rem/person)	MAN-REM PER MEGA-WATT-YEAR
Three Mile	75	675.9	~168	~83	~0.49	~0.1
Island 1	76	529	819	286	0.35	0.54
819 MWe	77	624	1122	360	0.32	0.58
Trojan 1	77	741	591	174	0.29	0.24
1130 MWe						
Turkey Point 3, 4	73	402	444	78	0.18	0.19
745 MWe x 2	74	953	794	454	0.57	0.48
	75	1003.7	1176	876	0.74	0.87
	76	972	1647	1184	0.72	1.22
	77	928	1319	1036	0.79	1.12
Zion 1, 2	74	424	306	56	0.18	0.13
1040 MWe x 2	75	1181	436	127	0.29	0.11
	76	1132	774	571	0.74	0.50
	77	1291	784	1004	1.28	0.78
	78	1578	1104	952	0.86	0.60
Average**					0.72	0.84

* The number of personnel includes station personnel, contractors, and temporary workers. Generally, only the number of individuals with exposures greater than 100 mrems is reported.

** Averages include values corresponding to the first year in which the power generated was 55% or more of the rated output and values corresponding to all subsequent years.

TABLE 12.4-2

PERSONNEL EXPOSURE DATA FOR MULTIPLE-UNIT OPERATING PWRs (1,15)

STATION	YEAR	TOTAL ANNUAL MAN-REM	AVERAGE EXPOSURE (REM/PERSON) TOTAL	CONTRACTOR	UTILITY
Oconee 1, 2, 3	74	517	0.61	0.57	0.63
886 MWe x 3	75	457	0.84	0.74	0.87
	76	987	1.07	0.84	1.15
Point Beach 1, 2	72	580			
497 MWe x 2	73	570	0.78		
	74	295	0.74		
	75	456	1.3		
	76	362	1.40	0.89	1.84
Prairie Island 1, 2	74	18	0.12	0.09	0.14
530 MWe x 2	75	123	0.26	0.22 (3,4)	0.23 (3,4)
	76	424	0.83		
Surry 1, 2	73	152	0.16		
823 MWe x 2	74	884	0.52		
	75	1549	1.91		
	76	3060	1.57	1.34	2.07
Turkey Point 3, 4	73	78	0.18		
745 MWe x 2	74	454	0.57		
	75	875	0.74		
	76	1408	1.21	1.43	0.86

TABLE 12.4-2 (Cont'd)

STATION	YEAR	TOTAL ANNUAL MAN-REM	AVERAGE EXPOSURE (REM/PERSON) TOTAL	CONTRACTOR	UTILITY
Zion 1, 2	74	33	0.18	0.15	0.20
1050 MWe x 2	75	118	0.08	0.05	0.15
	76[5]	525	0.31	0.23	0.46
Average (See Note)		543	0.78	--	--

Byron/Braidwood original Estimated Annual Man-rem 800

(1) Based on total annual man-rem average 543

(2) Based on 250 station employees) at 0.078 rem/person 800
775 contract workers)

NOTES:

Averages include values corresponding to the first year in which the power generated was 55% or more of the rated output, and values corresponding to all subsequent years. (See Table 12.4-1)

Data for Surry is not included in the averages, since the steam generator tube failures which resulted in high man-rem exposures at Surry are not expected to occur at Byron/Braidwood, which has a different steam generator design and all-volatile chemistry for feedwater conditioning.

The predicted occupational exposure at Zion for 1977 is 700-750 man-rem.

TABLE 12.4-3

REPORTED PERSONNEL EXPOSURE BY WORK FUNCTION FOR SEVERAL OPERATING PWRs (5-14)

STATION	YEAR	ROUTINE OPERATIONS AND SURVEILLANCE	ROUTINE MAINTENANCE AND INSERVICE INSPECTION	REFUELING	RADWASTE PROCESSING AND HANDLING	OTHER	SPECIAL MAINTENANCE
Ginna	1975	27	192	61	19	12	180
Haddam Neck	1975	30	185	64	7	1	381
Maine Yankee	1975	25	105	138	27	NR	0
Average (single-unit plants)		27	161	88	18	-	187
Oconee 1, 2, 3	1976	63	180	138	30	NR	575
Point Beach 1, 2	1976	56	148	125	26	NR	6
Prairie Island 1,2	1976	64	58	32	8	NR	262
Surry 1, 2	1974	46*	373	43	NR	NR	127
	1975	39	104	47	90	NR	160
	1976	429	1210	133	NR	NR	1287
Turkey Point 3,4	1976	111	977	9	24	NR	293
Zion 1,2	1975	16	190	3	8	NR	NR
	1976	59	162	13	14	NR	NR
	1977	42	334	18	26	NR	11
	1978	141	299	22	24	NR	75

TABLE 12.4-3 (Cont'd)

STATION	YEAR	ROUTINE OPERATIONS AND SURVEILLANCE	ROUTINE MAINTENANCE AND INSERVICE INSPECTION	REFUELING	RADWASTE PROCESSING AND HANDLING	OTHER	SPECIAL MAINTENANCE
Average (multiple-unit plants) (See Notes)		66	290	49	19	-	247

Byron/Braidwood Estimated Annual Man-rem

Routine Maintenance and Surveillance	65
Routine Maintenance and Inservice Inspection	300
Refueling	65
Radwaste Processing and Handling	20
Other	50
Special Maintenance	300
TOTAL	800

TABLE 12.4-3 (Cont'd)

NOTES:

Exposures given were reported as the sum of individual exposures greater than 500 mrem, except for Zion.

Where the breakdown in the original report was more detailed, categories have been condensed as necessary to obtain the categories given here.

The category "other" includes training, miscellaneous, security, consultants, etc.

Where data was incomplete for one-half of the year, the data was prorated from the other complete half of the year, except for refueling.

NR means "not reported," that is, no data for this or any similar category was reported.

*"Normal surveillance" only was reported.

Data for Surry is not included in the averages, since the steam generator tube failures which resulted in high man-rem exposures at Surry are not expected to occur at Byron/Braidwood, which has a different steam generator design and an all-volatile chemistry for feedwater conditioning.

Estimates are conservative to account for exposures less than 100 mrem which are not generally included in reports of occupational exposure and thus are not included in the averages.

TABLE 12.4-4

ANNUAL THYROID DOSES RESULTING FROM CALCULATED

DESIGN-BASIS AIRBORNE CONCENTRATIONS IN REMS/YR

ISOTOPE	AUXILIARY* BUILDING	CONTAINMENT** BUILDING	RADWASTE*** BUILDING
I-131	6.7×10^{-1}	1.67	1.8×10^{-3}
I-132	9.0×10^{-3}	1.65×10^{-2}	negligible
I-133	2.7×10^{-1}	5.0×10^{-1}	negligible
I-134	3.1×10^{-3}	8.2×10^{-4}	negligible
I-135	4.7×10^{-2}	3.3×10^{-2}	negligible

* The above dose rates are based on 13.3 hr/wk exposure of personnel in the auxiliary building, of which 50% is spent in clean areas, 35% in general areas with potential airborne, 10% in pump room and valve aisle, and 5% in radiation areas.

** Reactor building dose rates are based on 13.3 hr/wk of which 1% is spent in the containment.

*** Radwaste building dose rates are based on 13.3 hr/wk with 5% occupation time.

TABLE 12.4-5

ESTIMATED FIFTH YEAR RADIATION DOSE FOR B/B COMPARED WITH 1976 AND 1977 OPERATING DATA*

WORK FUNCTION	PER UNIT* 1976	PER UNIT* 1977	2 UNIT STATIONS* 1976	2 UNIT STATIONS* 1977	AVERAGE 1976-1978 ZION 1&2	B/B 1&2 (ESTIMATED)
Routine Operations and Surveillance	38	36	66	73	81	70
Routine Inspection and Maintenance	164	150	284	303	279	260
Refueling	29	22	51	46	19	20
Radwaste	19	20	32	41	21	10
Special Maintenance	123	110	210	230	420	230
TOTAL	373	338	643	693	820	590
Power Rating	(761 Mwe)		(1380 Mwe)		(2100 Mwe)	(2300 Mwe)

* PWR operating data taken from NUREG-0463, Table 7 and Appendix A minus the Surry 1&2 data. The units are man-rems unless designated otherwise.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 Organization

The administrative organization of the health physics program and personnel responsibilities are referenced in Subsections 12.1.1.1 and 12.1.1.2.

The experience and qualification of all station personnel are given in station procedures.

12.5.2 Equipment, Instrumentation, and Facilities

Table 12.5-1 lists the normal storage location of respiratory protective equipment, protective clothing, and portable and laboratory technical equipment and instrumentation.

Respiratory protective equipment is used to limit intakes of airborne radioactive material when engineering controls are not feasible and when consistent with the principle of minimizing total effective dose equivalent. The following types of respirators are among those available for use: air purifying full mask respirators, air line full mask respirators, air line airborne hood respirators, and positive pressure self-contained breathing apparatus (SCBA). At Byron, in addition to the equipment listed in Table 12.5-1, a reserve of emergency breathing air is maintained for control room personnel. At Braidwood, in addition to the equipment listed in Table 12.5-1, emergency breathing air for control room personnel is provided by additional SCBAs with bottled air available for backup.

Typical estimates for and the quantity, sensitivity, range, and frequency and methods of calibration for health physics instrumentation and technical equipment are specified in Table 12.5-2.

Table 12.5-2 shows portable radiation monitoring instrumentation capable of measuring exposure rates up to 10,000 R/hr. Such instrumentation would be used under accident conditions in areas where it is impractical to have installed stationary monitors. Since source calibration of high range instrumentation is impractical on the upper scales, only electronic calibrations will be performed for the upper scales/decades of high range exposure rate instrumentation.

Health physics and radiochemist facilities are described in Table 12.5-3.

12.5.3 Procedures

The health physics procedures have been developed to implement Exelon Generation Company's commitment to "As Low As Reasonably Achievable" (ALARA) as stated in Subsection 12.1.1.

12.5.3.1 Administrative Program

Strict administrative control of radiation exposure includes those methods described in Subsections 12.5.3.2, 12.5.3.3, and 12.5.3.5. Other administrative controls used include locked high radiation areas, radiation work permits, timekeeping of personnel in high radiation areas, and security measures including escorts for visitors within the plant security area.

12.5.3.2 Personnel External Exposure Program

The personnel external exposure program consists of multiple methods of reviewing external radiation levels and controls within the plant. These provide plant and personnel status information required to maintain an ALARA program (Subsection 12.1.1).

Area radiation monitors (ARMs) are located throughout the plant and provide general area indication of gamma radiation levels. These levels are continuously monitored and are alarmed in the control room. Some monitors also have local indication and alarm at certain in-plant locations. Besides surveillance by control room operators, these levels are periodically reviewed by a health physicist to note unusual trends. Process radiation monitors with control room indication and alarms also provide for immediate recognition of significant increases in in-plant dose rate levels.

Routine beta-gamma dose rate surveys are made of general access areas of the plant. This provides detailed dose rate information for normal in-plant exposure evaluation. The survey sheets are reviewed to note unusual trends and for determination of additional controls that may be required due to new or increased radiation dose rates.

Special beta-gamma dose rate surveys are made on an as needed basis for jobs that take place in normally inaccessible (i.e., high radiation) areas. These areas are not normally surveyed on a routine basis due to the required dose commitment being inconsistent with the ALARA program. Continuous or intermittent surveys are provided on an as needed basis as determined by radiation protection for radiation work permits (Subsection 12.5.3.1).

Personnel entering radiologically posted areas onsite are required to wear appropriate dosimetry. This dosimetry consists of a primary dosimeter of legal record and electronic self-reading dosimeters. Daily, the self-reading dosimeter readings (or equivalent) and timekeeping results (if applicable) are normally recorded, and are routinely reviewed by radiation protection management and by management in the individual's work group, if applicable. The primary dosimeters of legal record are changed routinely. Badge results are reviewed and are entered in the Exelon Generation Company computerized radiation exposure records system. These official and permanent records furnish the exposure data for the administrative control of radiation exposure. Required reports are made by radiation protection management through the use of this records system.

General area neutron dose rate measurements are made during startup after initial fuel loading and following refueling outages to verify neutron dose rates. Special neutron surveys and use of neutron dosimeters are provided when entrance is made into neutron areas when required by 10 CFR 20.

Radioactive materials and special nuclear materials are handled and stored under the direction of personnel as specified in Subsection 12.1.1.2.

12.5.3.3 Personnel Internal Exposure Program

The personnel internal exposure program consists of multiple methods of reviewing airborne radioactivity concentrations and controls within the plant. These provide plant and personnel status information required to maintain an ALARA program (Subsection 12.1.1).

The Station vent stack monitors (one for each of the two vent stacks) have detectors for air particulate, gas (low and high range), iodine, and background subtraction. In addition to surveillance by control room operators, monitor levels are periodically reviewed by radiation protection personnel to note unusual trends.

Continuous air monitors also monitor auxiliary building ventilation exhausts, containment purge systems, and the radwaste building ventilation exhausts. These are used to measure, indicate, and record levels of airborne radioactivity in air exhausted from plant areas and as trending devices by radiation protection personnel.

Portable grab samples are normally taken in accessible areas of the plant on a periodic basis. Special samples are taken as required by radiation protection personnel prior to issuing Radiation Work Permits and before other jobs as necessary. These air sample results are reviewed by radiation protection personnel and are used to determine respiratory protective equipment requirements in accordance with Station radiation protection procedures.

Whole body counts are performed for plant personnel with a frequency as specified in the station radiation protection procedures.

All personnel (permanent and temporary) are normally requested to have a whole body count or whole body screening before termination if they have worked in airborne radioactivity or with radioactive materials unless specifically exempted by radiation protection management. Other bioassay techniques may be substituted, such as urinalysis and fecal analysis. A personnel bioassay program is administered by a health physicist. Bioassay (in vivo measurement and/or measurement of radioactive material in excreta) are conducted as necessary to aid in determining the extent of an individual's internal exposure to concentration of radioactive material. The need for and frequency of bioassay are determined by the duration that a person works with radioactive materials or in an airborne radioactivity area. Specific frequencies are determined and controlled by procedures. All bioassay results are recorded as required.

The Byron/Braidwood bioassay program is implemented in compliance with Revision 1 of Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

12.5.3.4 Contamination Control Program

The contamination control program consists of multiple methods of controlling the spread of contamination to personnel and equipment within the plant. Routine smear surveys are periodically made of normally accessible areas of the plant and are recorded on survey sheets. These results are reviewed by a radiation protection supervisor or a health physicist. Special smear surveys are performed on an as needed basis for radiation work permits and for unconditional release of equipment, tools, and materials being removed from radiologically posted areas. Items which are contaminated are required to be decontaminated to within release limits or packaged and tagged in accordance with the station radiation protection procedures.

Workers in contaminated areas are required to be monitored for contamination prior to leaving the contamination control point for the areas. Additionally, portal-type monitors are utilized to monitor individuals leaving the radiologically posted area (RPA) via the main access area and again when leaving the site (in the security gatehouse). Actual instrumentation used for the contamination surveys is determined by station management.

12.5.3.5 Training Program

The radiation protection training programs are described in Section 13.2. This program covers the following:

a. general employee health physics,

b. general employee respiratory protection,

c. contractor health physics,

d. contractor respiratory protection,

e. general employee retraining,

f. Radiation Protection Technician training, and

g. Radiation Protection Technician retraining.

All personnel must understand how radiation protection relates to their jobs and have reasonable opportunities to discuss radiation protection safety with a member of the Radiation Protection department whenever the need arises. Plant personnel are made aware of Exelon Generation Company commitment to keep occupational radiation exposure as low as reasonably achievable (Subsection 12.1.1). A minimum goal of this program is that workers shall be sufficiently familiar with this commitment that they can explain what the management commitment is, what "As Low As Reasonably Achievable" means, why it is recommended, and how they have been advised to implement it on their jobs.

Qualifications of personnel, including training requirements for radiation protection personnel are described in Subsection 12.5.1.

TABLE 12.5-1

STORAGE LOCATION OF EQUIPMENT

EQUIPMENT	NORMAL STORAGE LOCATION
Self-Contained Breathing Apparatus (Pressure Demand)	Control Room, Technical Support Center, Operational Support Center
Full Face Masks (Air Purifying) Full Face Masks (Airline) Hoods (Airline)	Auxiliary Building - Mask Storage Area
Protective Clothing	Auxiliary Building
β-γ Air Ionization Chambers G-M Survey Instruments Neutron Detector	Auxiliary Building - Calibration Facility
Chemical Analysis Equipment	Hot Laboratory, Cold Laboratory

TABLE 12.5-2

HEALTH PHYSICS EQUIPMENT

TYPE DETECTOR/MONITOR*	ESTIMATED NUMBER	SENSITIVITY	RANGE	FREQUENCY	CALIBRATION METHOD
Gamma Ray Counting System	2	Variable	Variable	Per CY-AA approved procedure	Standard Reference Materials
Alpha/Beta Counting System	2	Variable	Variable	Per CY-AA approved procedure	Standard Reference Materials
Air Ion Chamber Exposure Rate Meter	30	Variable	Variable	Annual	Standard Reference Materials
GM Survey Count Rate Instrument	15	Variable	Variable	Annual	Standard Reference Materials
Alpha Scintillator Probe	2	Variable	0-100K cpm	Annual	Standard Reference Materials
High Range, Exposure Rate	5	Variable	0-10,000 R/hr	Annual	Standard Reference Materials
Neutron Detector	2	Variable	0-5 Rem/hr	Annual	Standard Reference Materials/Mini Pulser
Air Sampler	10	N/A	Variable	Annual	Air Flow Calibrator
Portable Continuous Air Monitor	5	Variable	0-50K cpm 0-10 cfm	Annual	Manometer, Standard Reference Materials

*The instrument/equipment list is intended to be typical of in-service instrumentation.

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TABLE 12.5-3

HEALTH PHYSICS AND RADIOCHEMICAL FACILITIES

NAME	LOCATION	PRIMARY FUNCTION
Calibration Facility	Auxiliary Building	Calibration of Gamma Dose Rate Instruments and Storage of Survey Instruments
Hot Laboratory	Auxiliary Building	Chemical Analysis and Radiochemical Separations
Cold Laboratory	Auxiliary Building	Chemical Analysis
Supply Room (Braidwood only)	Auxiliary Building	Storage of Chemicals, Glassware, and Laboratory Equipment
Counting Room	Auxiliary Building	Radioactivity and Radiological Determination of Samples
Laundry Room* (Byron)	Auxiliary Building	Storage of Protective Radiological Clothing
Laundry Room* (Braidwood)	Auxiliary Building	Store equipment and supplies, sort low level radioactive trash, and launder personal clothing
Mask Cleaning Facility	Auxiliary Building	Cleaning, Inspection, and Storage of Respiratory Equipment
Health Physics Offices	Turbine Building Area/Service Building	Administration/Offices

*An offsite vendor is utilized to clean potentially contaminated protective clothing.

Figure 12.2-1 has been deleted intentionally.

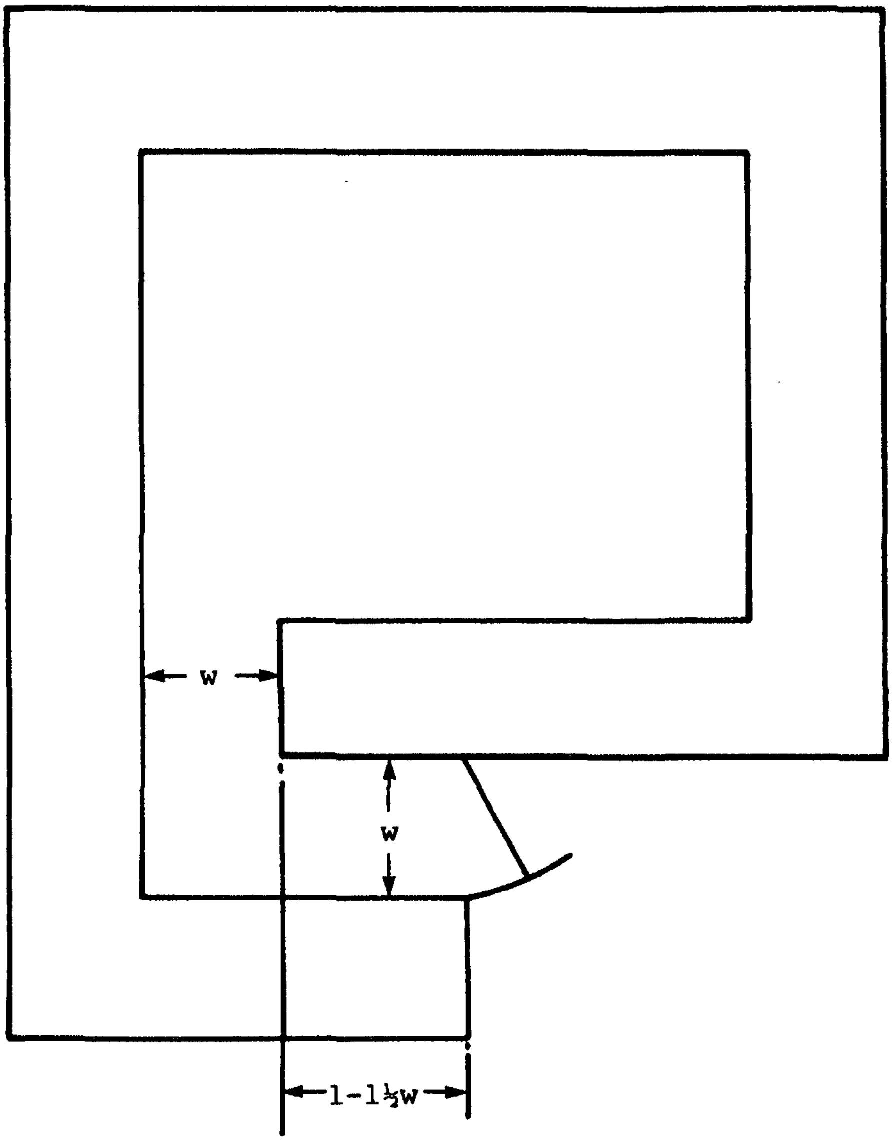

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FIGURE 12.3-1

SKETCH OF A SIMPLE
LABYRINTH ENTRANCE

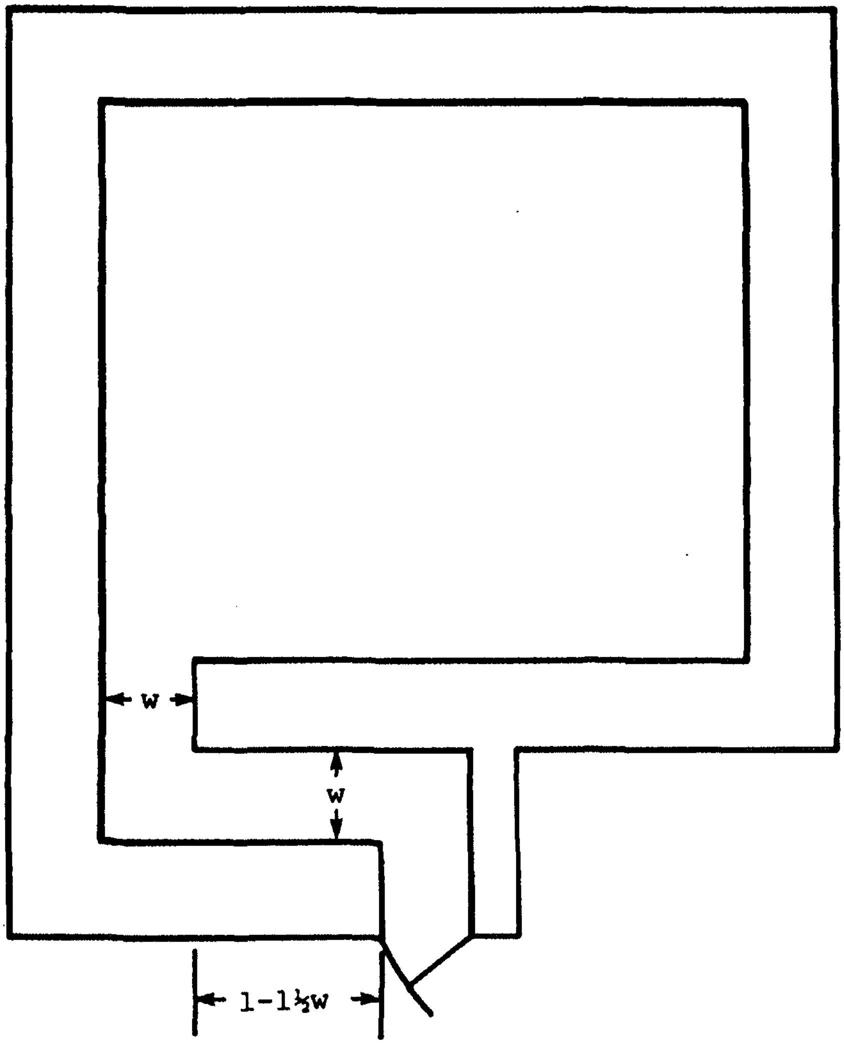

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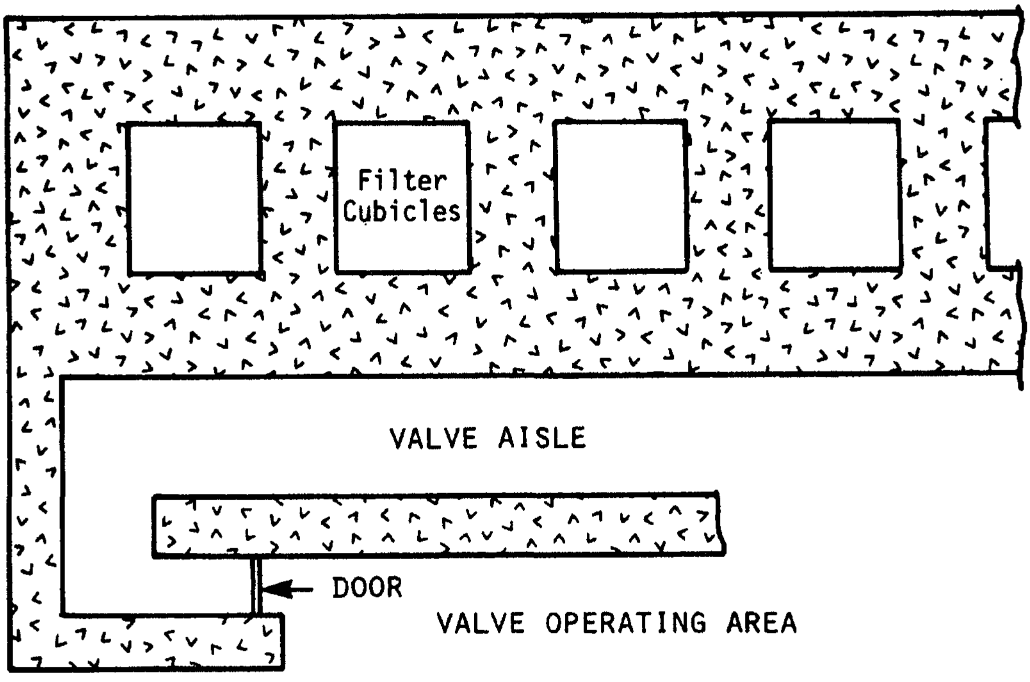
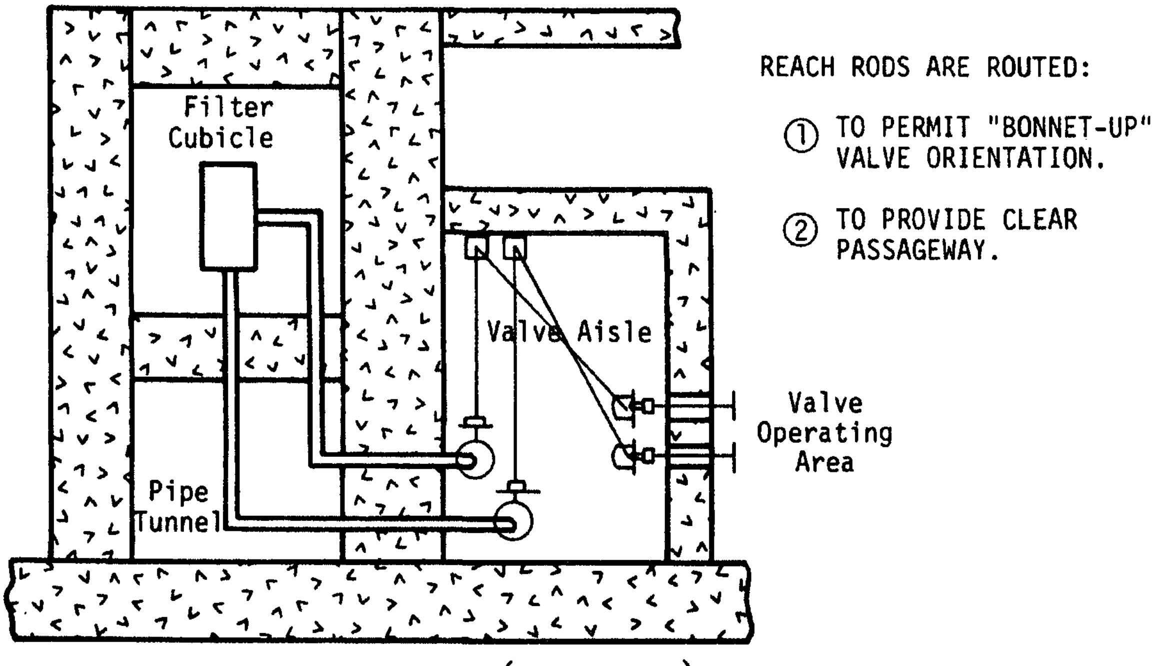
FIGURE 12.3-2

SKETCH OF A DOUBLE LABYRINTH ENTRANCE



LAYOUT FOR WALK-IN VALVE AISLE (PLAN VIEW)



WALK-IN VALVE AISLE (ELEVATION)

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FIGURE 12.3-3

TYPICAL WALK-IN VALVE AISLE

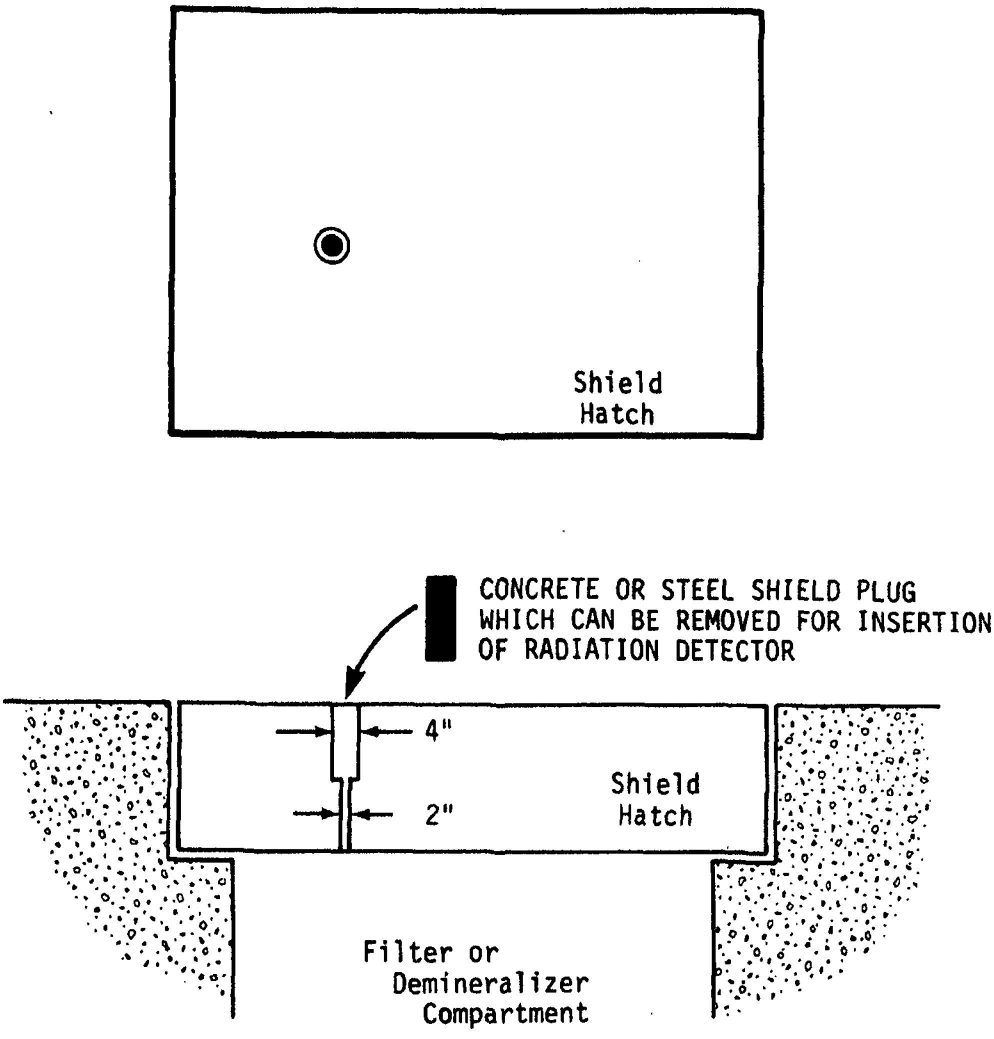

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FIGURE 12.3-4

SKETCH OF RADIATION DETECTOR PROBE ACCESS HOLE IN SHIELD HATCH FOR FILTER OR DEMINERALIZER

Figure 12.3-5 through 12.3-26a have been deleted intentionally.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 12.3-27

RADIATION ZONE MAP FOR NORMAL OPERATION
ROOF PLAN EL. 477 FT 0 IN. AND EL. 485 FT 0 IN.
COLUMNS 18 THROUGH 30

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FIGURE 12.3-28

RADIATION ZONE MAP FOR NORMAL OPERATION
ROOF PLAN EL. 477 FT 0 IN. AND EL. 485 FT 0 IN.
COLUMNS 6 THROUGH 18

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FIGURE 12.3-29

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 451 FT 0 IN.

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FIGURE 12.3-30

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 439 FT 0 IN.

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FIGURE 12.3-31

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 426 FT 0 IN.

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FIGURE 12.3-32

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 401 FT 0 IN.

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FIGURE 12.3-33

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 383 FT 0 IN.

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FIGURE 12.3-34

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 364 FT 0 IN.

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FIGURE 12.3-35

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING EL. 346 FT 0 IN.

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FIGURE 12.3-36

RADIATION ZONE MAP FOR NORMAL OPERATION
MISCELLANEOUS PLANS

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FIGURE 12.3-37

RADIATION ZONE MAP FOR NORMAL OPERATION
PIPE TUNNELS

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FIGURE 12.3-38

RADIATION ZONE MAP FOR NORMAL OPERATION
AREAS BETWEEN AUXILIARY BUILDING AND
CONTAINMENT BUILDING

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FIGURE 12.3-39

RADIATION ZONE MAP FOR NORMAL OPERATION
FUEL HANDLING BUILDING EL. 426 FT 0 IN.

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FIGURE 12.3-40

RADIATION ZONE MAP FOR NORMAL OPERATION
FUEL HANDLING BUILDING EL. 401 FT 0 IN.

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FIGURE 12.3-41

RADIATION ZONE MAP FOR NORMAL OPERATION
CONTAINMENT BUILDING EL. 377 FT 0 IN.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 12.3-42

RADIATION ZONE MAP FOR NORMAL OPERATION
CONTAINMENT BUILDING EL. 390 FT 0 IN.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 12.3-43

RADIATION ZONE MAP FOR NORMAL OPERATION
CONTAINMENT BUILDING EL. 401 FT 0 IN.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 12.3-44

RADIATION ZONE MAP FOR NORMAL OPERATION
CONTAINMENT BUILDING EL. 426 FT 0 IN.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 12.3-45

RADIATION ZONE MAP FOR NORMAL OPERATION
RADWASTE/SERVICE BUILDING EL. 397 FT 0 IN.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 12.3-46

RADIATION ZONE MAP FOR NORMAL OPERATION
RADWASTE/SERVICE BUILDING EL. 433 FT 0 IN.

Security - Related Information Figure Withheld Under 10 CFR 2.390

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-47

RADIATION ZONE MAP FOR NORMAL OPERATION
CONDENSATE POLISHING/TECHNICAL SUPPORT
CENTER

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-48

RADIATION ZONE MAP FOR NORMAL OPERATION
AUXILIARY BUILDING ELEVATIONS 459 FT 2 IN.,
463 FT 5 IN., AND 475 FT 6 IN.

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FIGURE 12.3-49

RADIATION ZONE MAP FOR SHUTDOWN
ROOF PLAN EL. 477 FT 0 IN. AND EL. 485 FT 0 IN.
COLUMNS 18 THROUGH 30

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-50

RADIATION ZONE MAP FOR SHUTDOWN
ROOF PLAN EL. 477 FT 0 IN. AND EL. 485 FT 0 IN.
COLUMNS 6 THROUGH 18

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FIGURE 12.3-51

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL 451 FT 0 IN.

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FIGURE 12.3-52

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL. 439 FT 0 IN.

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FIGURE 12.3-53

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL. 426 FT 0 IN.

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-54

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL. 401 FT 0 IN.

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FIGURE 12.3-55

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL. 383 FT 0 IN.

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FIGURE 12.3-56

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL. 364 FT 0 IN.

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FIGURE 12.3-57

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING EL. 346 FT 0 IN.

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FIGURE 12.3-58

RADIATION ZONE MAP FOR SHUTDOWN
MISCELLANEOUS PLANS

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FIGURE 12.3-59

RADIATION ZONE MAP FOR SHUTDOWN
PIPE TUNNELS

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FIGURE 12.3-60

RADIATION ZONE MAP FOR SHUTDOWN
AREAS BETWEEN AUXILIARY BUILDING
AND CONTAINMENT BUILDING

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FIGURE 12.3-61

RADIATION ZONE MAP FOR SHUTDOWN
FUEL HANDLING BUILDING EL. 426 FT 0 IN.

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FIGURE 12.3-62

RADIATION ZONE MAP FOR SHUTDOWN
FUEL HANDLING BUILDING EL. 401 FT 0 IN.

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FIGURE 12.3-63

RADIATION ZONE MAP FOR SHUTDOWN
CONTAINMENT BUILDING EL. 377 FT 0 IN.

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FIGURE 12.3-64

RADIATION ZONE MAP FOR SHUTDOWN
CONTAINMENT BUILDING EL. 390 FT 0 IN.

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FIGURE 12.3-65

RADIATION ZONE MAP FOR SHUTDOWN
CONTAINMENT BUILDING EL. 401 FT 0 IN.

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FIGURE 12.3-66

RADIATION ZONE MAP FOR SHUTDOWN
CONTAINMENT BUILDING EL. 426 FT 0 IN.

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FIGURE 12.3-67

RADIATION ZONE MAP FOR SHUTDOWN
RADWASTE/SERVICE BUILDING EL. 397 FT 0 IN.

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FIGURE 12.3-68

RADIATION ZONE MAP FOR SHUTDOWN
RADWASTE/SERVICE BUILDING EL. 433 FT 0 IN.

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FIGURE 12.3-69

RADIATION ZONE MAP FOR SHUTDOWN
CONDENSATE POLISHING/TECHNICAL SUPPORT
CENTER

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-70

RADIATION ZONE MAP FOR SHUTDOWN
AUXILIARY BUILDING ELEVATIONS 459 FT 2 IN.,
463 FT 5 IN., AND 475 FT 6 IN.

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BYRON / BRAIDWOOD
GENERAL ARRANGEMENT-RADIATION
SHIELDING OLD STEAM
GENERATOR STORAGE FACILITY

M-24

SHEET NUMBER: 23 SIZE: F

UFSAR FIGURE: 12.03-26A SHT: 23

UFSAR FIGURE: 12.03-71 SHT: 23

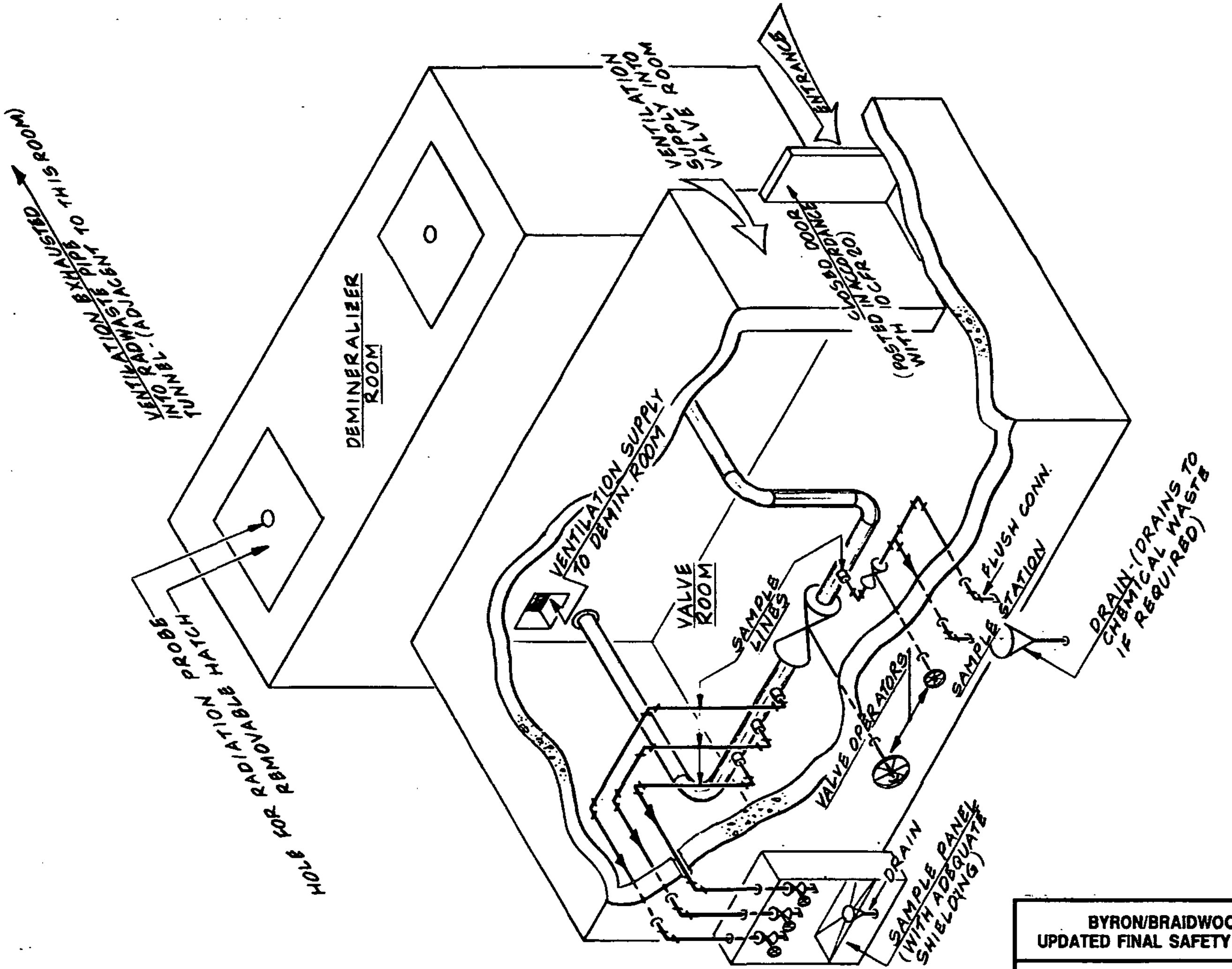

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FIGURE 12.3A-1

FILTER/DEMINERALIZER EQUIPMENT, SAMPLING STATION, AND PANEL

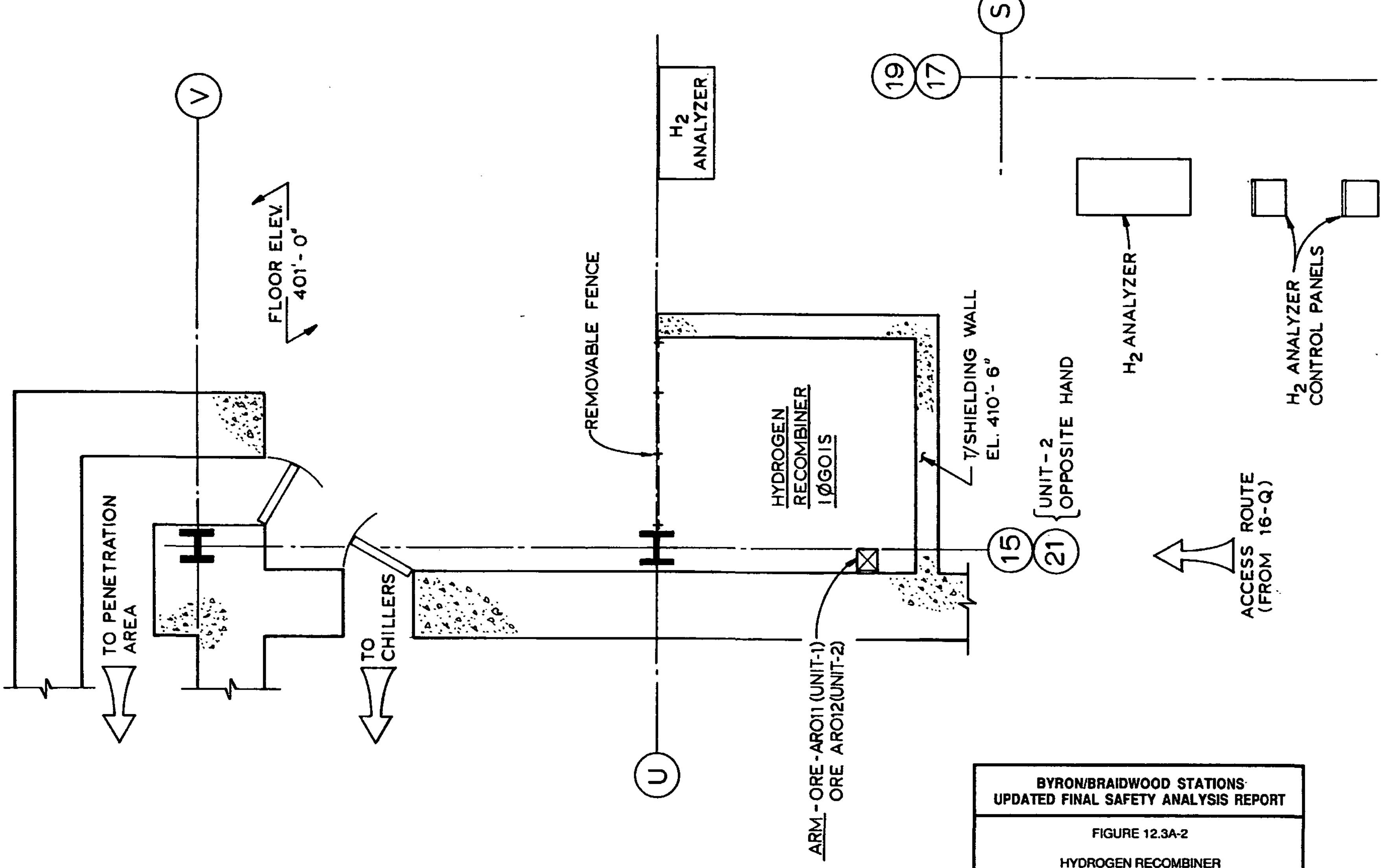

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FIGURE 12.3A-2

HYDROGEN RECOMBINER

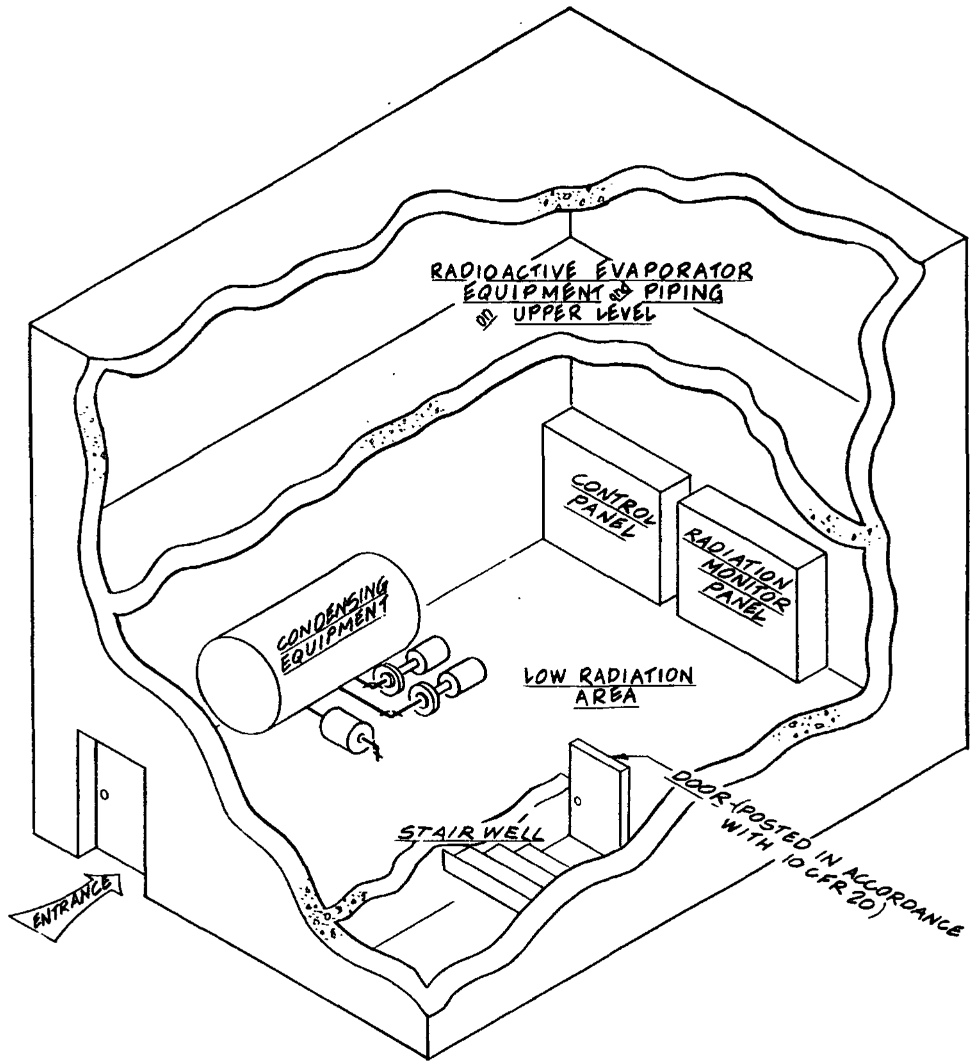

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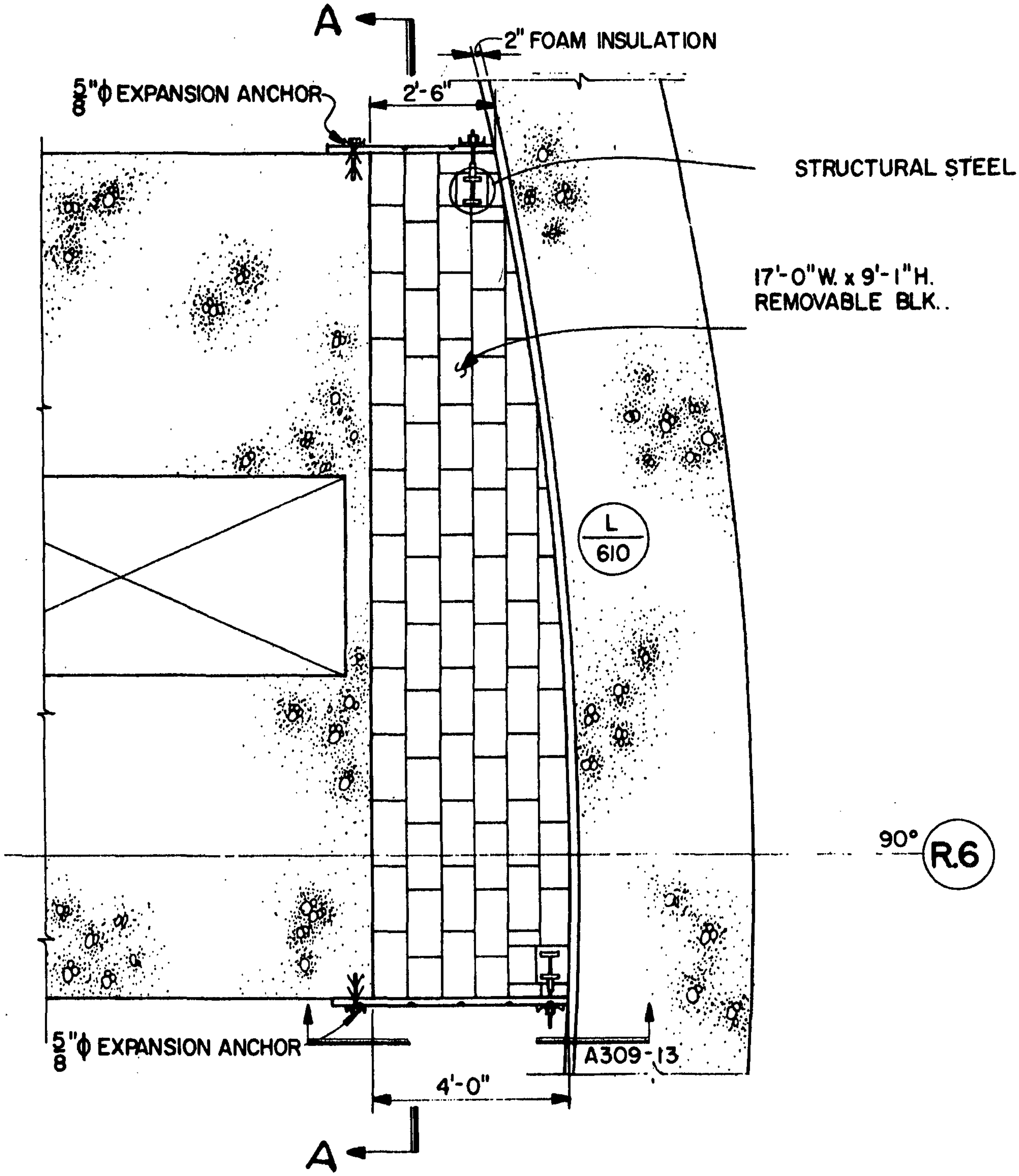
FIGURE 12.3A-3

EVAPORATOR EQUIPMENT



PARTIAL PLAN AT EL. 389'-11" UNIT 1

SCALE: 3/8" = 1'-0"

UNIT 2 IS OPPOSITE HAND

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FIGURE 12.3A-4

REMOVABLE BLOCK WALL PLAN

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FIGURE 12.3A-5

REMOVABLE BLOCK WALL SECTIONS