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CHAPTER 12 RADIATION PROTECTION

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CHAPTER 12.0 RADIATION PROTECTION

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

12.1.1 Policy Considerations

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel will be kept ALARA.

12.1.1.1 Design and Construction Policies

The ALARA philosophy was applied during the initial design of the plant and implemented via internal design reviews. Each design was given an ALARA review by engineers in the project organization of Architect/Engineer/Constructor (A/E/C). The design was then reviewed independently for ALARA by the staff organization of the A/E/C. These reviewers included professional nuclear engineers and health physicists with a number of years of experience in ALARA nuclear power plant design and operation.

In addition, the design was reviewed for ALARA by the GGNS Nuclear Project Engineering group and the Grand Gulf Plant staff. These reviewers also included professional nuclear engineers and health physicists with a number of years of experience in ALARA nuclear power plant design and operation. These reviews were consistent with the recommendations of Regulatory Guide 8.8.

The plant design was reviewed, updated, and modified as necessary during the design and construction phases. Engineers reviewed the plant design and integrated the layout, shielding, ventilation, and monitoring designs with traffic control, security, access control, and health physics aspects to ensure that the overall design is conducive to maintaining exposures ALARA.

All pipe routings containing radioactive fluids were reviewed as part of the engineering design effort. This ensured that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure to personnel.

Onsite inspections are conducted, as necessary, to check the shielding and piping layout design. If any of these review steps indicate a necessity for modifying the shielding or routing of a given piping run, such modifications are made.

To comply with the ALARA policy, inspection and testing of plant shielding will be conducted to verify that the shielding performs its function of reducing radiation to design levels. During construction, a visual inspection was made to ensure that there were no major defects in the shield walls as they were poured. During initial power operations, radiation surveys, discussed in detail in subsection 14.2.12.3.2 Test Description for Radiation Measurements, were conducted to ensure that there were no defects in the shielding that might seriously affect personnel exposures during normal operation and maintenance of the plant.

12.1.1.2 Operation Policies

It is the responsibility of the General Manager, Plant Operations to ensure management's commitment to a program to maintain occupational radiation exposures ALARA, consistent with recommendations of Section C.1 of NRC Regulatory Guide 8.8 (Rev. 2). Plant design changes are reviewed as directed by the General Manager, Plant Operations.

To verify that the overall radiation protection program is functioning properly, independent reviews and audits required by QAPM are conducted.

The Grand Gulf Nuclear Station Plant Administrative and Radiation Protection Procedures are two of the means of instituting the operational ALARA policy, since they are available to each member of the station staff who will receive training covering the contents.

In addition to defining management's commitment to ALARA, the Plant Administrative and Radiation Protection Procedures designate the station personnel who have the responsibility and authority to implement the ALARA program. The General Manager, Plant Operations bears the final responsibility for implementing the ALARA program, but has delegated this responsibility to the Radiation Protection Manager. The authority to prevent unsafe practices and to direct steps to prevent any unnecessary radiation exposures lies within the Radiation Protection Manager or designee responsibilities. The Radiation Protection Supervisors will assist the Radiation Protection Manager in ensuring that radiation exposures are maintained ALARA. The Radiation Protection Supervisors handle the day-to-day operation of the site radiation protection program and report to the Radiation Protection Manager. He will ensure implementation of

the ALARA program and supervise the health physicists who perform the various surveys for radiation protection. For a more detailed discussion on the responsibility and authority of the key supervisory positions discussed above and the qualifications of the personnel who fill them see subsections 13.1.2 and 13.1.3.

It is the responsibility of the Radiation Protection Manager to see that company employees and contractors are trained in radiation protection requirements to comply with 10 CFR 19, 10 CFR 20, 10 CFR 55, and the recommendations of Regulatory Guides 8.8, 8.10, and 8.13, and to verify that personnel follow the Plant Administrative and Radiation Protection Procedures. To ensure compliance with this policy, the Radiation Protection Supervisors are charged with the responsibility to promptly advise higher management of any unsafe practices which exceed their authority to correct. They have the authority to halt any operation which is, in their judgment, unsafe. It is also the prerogative of the working health physicists to halt any operation which is, in their judgment, unsafe pending review by Radiation Protection Supervisor.

In addition to reviews by management, all employees are encouraged to submit suggestions relating to radiation protection to their immediate supervisors or Radiation Protection supervision. The supervisor is expected to remedy any unsafe condition promptly on valid suggestions if it is within the scope of his responsibility. Any problem that is not directly attributable to the supervisor's responsible area shall be reported to Radiation Protection who will take the suggestion to a Radiation Protection Supervisor.

Prior to initial startup, personnel were trained in radiation protection procedures and techniques if their assignments involved radiation work. Personnel were also tested to verify that they understood how these procedures related to the safe performance of their jobs. Station personnel whose assignments require it will be trained and tested in radiation protection procedures, techniques, and basic emergency preparedness on a two-year cycle and will be tested annually.

Maintenance, refueling, and radwaste system operating procedures which involve significant radiation exposures will be reviewed to verify adherence to ALARA policy prior to their use. This review will be performed by health physicists and/or a Radiation Protection Supervisor, as specified in plant administrative

procedures. The Radiation Protection Supervisors and/or health physicists will observe implementation of selected procedures to identify situations in which exposures can be reduced. Those operations with higher exposure potentials will receive greater attention.

12.1.1.3 Compliance with Regulatory Guide 8.8, Rev. 2 (March 1977), Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable

Grand Gulf's design has been reviewed to ensure operator doses will be as low as is reasonably achievable. Many of the recommendations of Revision 2 to Regulatory Guide 8.8 which was issued over four years after the construction permit issue date have been incorporated in the design. Specific exceptions or points of clarification of the regulatory positions are as follows:

Paragraph C.1.c	-	Instructors presenting radiation protection training will be under the supervision of the training supervisor. Course material and presentation will be subject to approval of the Radiation Protection Manager.
Paragraph C.2.e(3)	-	Grand Gulf does not use bright hydrogen-annealed tubing and piping in the primary coolant and feedwater systems. This could interfere with the important requirement of obtaining fully solution annealed stainless steel components to protect against stress corrosion cracking. There is also no evidence that it would be of value in reducing deposition of activated corrosion products.

Paragraph C.2.e(5)	-	Grand Gulf does not use live-loaded valve packings and bellow seals to reduce the leakage of contaminated coolant from the primary system. Two sets of packing with a leakoff connection between them are used.
Paragraph C.2.e(6)	-	There is no laminar flow in the primary coolant system. The primary system is designed for conventional turbulent flow as a matter of good engineering practice and economy. Further, laminar flow in the primary coolant system would increase the problems associated with the buildup of radioactive crud in the piping. Laminar flow conditions would necessitate the use of larger piping systems to achieve the low flow velocity of the coolant and thereby increase, significantly, the internal piping surfaces exposed to the coolant. Typically, crud traps occur where low velocity conditions exist and any piping system design that incorporates low velocity conditions would only increase the possibility of having crud traps.
Paragraph C.2.g	-	Refer to subsection 12.3.4.2.4.
Paragraph C.2.g(1)	-	The central monitoring system does not have a readout capability at the main radiation protection access control point. The readout is in the control room.

Paragraph C.2.i(7) - Grand Gulf only uses vertical can pumps in the ECCS sumps. Operating experience has shown that canned pumps are not desirable because the bearings in these pumps require servicing (with the accompanying radiation doses) more frequently than a typical conventionally designed centrifugal pump.

Paragraph C.2.i(9) - There are no spare connections on tanks or other components located in higher radiation zones since the equipment was purchased and, in many cases, already installed before Revision 2 of Regulatory Guide 8.8 was issued. The plant systems have been designed with sufficient flexibility to minimize if not eliminate the need to add additional connections later. It would also not be good ALARA practice to install these connections in the higher radiation zones if no specific need or basis is identified.

Paragraph C.2.i(13) - Lamps in higher radiation areas cannot be serviced from lower radiation areas.

Refer to subsection 12.1.2 for further discussion of radiation protection design considerations.

Section C.3 - Information contained in this section is considered guidance and will be used as such in establishing methods to maintain exposures ALARA. However, these methods will be used only as deemed appropriate and in accordance with station procedures.

Paragraph C.3a(8)(e) - This information will be entered on RWPs. RWPs will be utilized for preventive, routine, and special maintenance, and refueling work performed inside high radiation, high contamination, airborne and potential airborne radioactivity areas, and neutron exposure areas.

Refer to subsection 12.1.3 and Section 12.5 for a discussion of the Radiation Protection Program.

Paragraph C.4a - This equipment is included in Table 12.5-2 which lists minimum laboratory instrumentation available.

Paragraph C.4b - Equipment is included in Table 12.5-1 which lists minimum portable instrumentation available.

Paragraph C.4.c - Same as C.4.b.

Paragraph C.4.d - Equipment is included in Table 12.5-1. All equipment and programs will be used. Commitment to Regulatory Guide 8.15 is discussed in Section 12.5.

Refer to subsection 12.5.2 for further discussion on Radiation Protection Facilities, Instrumentation, and Equipment.

12.1.2 Design Considerations

This subsection discusses the methods and features by which the policy considerations of subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures ALARA are presented in detail in subsections 12.3.1, 12.3.2, and 12.5.3.

12.1.2.1 General Design Consideration for ALARA Exposures

General design considerations and methods employed to maintain inplant radiation exposures ALARA, consistent with the recommendations of Section C.2 of NRC Regulatory Guide 8.8, have two objectives:

a. Minimizing the necessity for and amount of personnel time spent in radiation areas

b. Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention

Both equipment and facility designs are considered in maintaining exposures ALARA during plant operations including: normal operation maintenance and repairs, refueling operations and fuel storage, inservice inspection and calibrations, radioactive waste handling and disposal, and other events of moderate frequency and certain infrequent occurrence events.

12.1.2.2 Equipment General Design Consideration for ALARA

a. Equipment general design consideration to minimize the necessity for and amount of personnel time spent in a radiation area include:

   1. Reliability, durability, construction, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance.

   2. Servicing convenience for anticipated maintenance or potential repair, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair.

   3. Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment (including inservice inspection in accordance with ASME Code, Section XI).

   4. Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high.

b. Equipment general design considerations directed toward minimizing radiation levels in proximity to equipment or components requiring personnel attention include:

   1. Provision for draining, flushing, or if necessary, remote cleaning of equipment and piping containing radioactive material.

2. Design of equipment, piping, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps.

3. Utilization of welded connectors instead of flanged or threaded connections, high quality valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials.

4. Provisions for minimizing the spread of contamination into equipment service areas, including direct drain connections.

5. Provisions for isolating equipment from radioactive process fluids.

12.1.2.3 Facility Layout General DesignConsiderations for ALARA

a. Facility general design considerations to minimize the amount of personnel time spent in radiation areas include:

   1. Locating equipment, instruments, and sampling stations, which will require routine maintenance, calibration, operation, or inspection, for ease of access and minimum of required occupancy time in radiation areas.

   2. Laying out plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment.

   3. Providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area.

b. Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include:

   1. Separating radiation sources and occupied areas where practicable (e.g. pipes or ducts containing potentially high radioactive fluids not passing through occupied areas).

2. Providing adequate shielding between radiation sources and access and service areas.

3. Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone.

4. Providing central control panels to permit remote operation of all essential instrumentation and controls from lowest radiation zone practicable.

5. Where practicable for package units, separating highly radioactive equipment from less radioactive equipment, instruments, and controls.

6. Providing means and adequate space for utilizing movable shielding for sources within the service area when required.

7. Providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable.

8. Providing means for decontamination of service areas.

9. Providing space for pumps and valves outside of highly radioactive areas.

10. Providing remotely operated centrifugal discharge and/or backflushable filter systems for highly radioactive radwaste and cleanup systems.

11. Providing labyrinth entrances to radioactive pump, equipment, and valve rooms.

12. Providing adequate space in labyrinth entrances for easy access.

13. Maintaining ventilation air flow patterns from areas of lower radioactivity to areas of higher radioactivity.

12.1.2.4 Illustrative Examples of ALARA Improvements

Design improvements, which will result in person-rem reductions during operation and maintenance, have been implemented following ALARA reviews. Some examples are discussed below.

a. Improvements Based on Dose Assessment

Investigation of the number of personnel and number of hours required to perform maintenance on equipment in high radiation areas led to the establishment of a design criterion to separately shield components in high radiation areas such that the total dose rate contribution from adjacent components or rooms would be less than 15 mr/hr. With this protection, most of the dose received would be from the component being repaired and not from other equipment not involved in the repair.

b. Improvement Based on Operational Experience

Operational experience with radwaste filters, sludge processing, and radwaste solidification systems was utilized in the design and selection of components for systems. Use was also made of studies of operating plant experiences with similar systems to determine system failure rates, down times, number of personnel and man-hours required to repair the systems, and the person-rem associated with repairs.

c. Improvement Based on ALARA Design Review

Radiation levels for radioactive system piping were provided to engineering designers for use in conjunction with the radiation zone drawings to route radioactive piping. Then an ALARA review was made of the proposed pipe routing and shield wall penetrations by shielding engineers. Following the review, pipes were shielded and/or rerouted and penetrations were shielded and/or rerouted as necessary.

As the design of the Grand Gulf plant progressed and personnel became more familiar with ALARA concepts, the number of pipes that had to be rerouted or shielded and penetrations that had to be relocated or shielded decreased, indicating improvements in the application of ALARA by the designers and placement of less dependence on the ALARA review process to uncover potential problem areas.

d. Operating experience from other units and the ALARA design features of GGNS will be utilized in the development and revision of station operating and maintenance procedures and instructions to ensure that occupational radiation exposure is maintained ALARA.

12.1.2.5 Decommissioning Design Considerations for ALARA

The radiation protection aspects of the plant design, as described throughout Section 12.1, which have been incorporated to ensure that the plant can be operated and maintained with ALARA exposures, will also aid in the ALARA aspects of decommissioning. These include the following:

a. Accessibility for maintenance or removal of equipment

b. Shielding to provide protection during maintenance or during storage after termination of plant operations

c. Provisions for draining, flushing, or decontaminating equipment or piping

d. Separation of more radioactive equipment from the less radioactive equipment

e. Features to minimize crud buildup

f. Coatings applied to surfaces likely to become contaminated to facilitate cleanup

12.1.3 Operational Considerations

In accordance with company policy and consistent with the recommendations of Regulatory Guides 8.8 and 8.10, the radiation exposure of plant personnel will be kept ALARA by means of the radiation protection program discussed in Section 12.5. The radiation protection policies and practices contained therein are initiated through the training program discussed in Section 13.2 and through the Radiation Protection Procedures discussed in subsection 12.1.1 and Section 12.5.

Procedures for radiation-related jobs that routinely occur at an operating boiling water reactor (BWR) will be written and approved for use at GGNS. For unusual or first-time operations

which will involve significant radiation exposure, operating procedures or work control documents will normally be prepared by or with the assistance of the section doing the work.

The Radiation Protection Manager or designee will review it for ALARA purposes.

Techniques which will be embodied in station procedures, training, and/or work practices for ALARA purposes are discussed below, as are the criteria and/or conditions for their use. These techniques will not be employed if it is determined by the Radiation Protection Manager or designee that the total dose received may be increased or that the dose reduction may be negligible compared to the effort involved to implement the technique.

From operating experience at other BWRs and the Atomic Industrial Forum (AIF) National Environmental Studies Project Report (Ref. 1), it has been determined that a large percentage of exposure at an operating BWR occurs during plant outages from maintenance and inspection activities and not from normal operating activities. This is to be expected since during operation, instrumentation and valves can be operated from outside the shield walls, and operators only have to enter cubicles containing radioactive equipment for short periods of time to check equipment. Maintenance and inspection personnel usually must be in proximity to lines, valves, instruments, or other pieces of equipment which are radiation sources in order to perform their job.

Operation of the hydrogen water chemistry system results in several changes to the station chemistry balance.

Injection of excess free hydrogen into reactor feedwater shifts the stoichiometric oxygen concentration in the reactor vessel to near zero concentrations. The lower oxygen levels result in free nitrogen combining with the excess hydrogen resulting in generation of volatile NH_3 and NH_4 compounds versus less volatile NOH compounds. These compounds are removed from the vessel in the main steam and result in increased main steam line dose rates from the additional nitrogen 16. The magnitude of this N-16 shift to steam is based on the feedwater hydrogen concentration.

Switching to a hydrogen water chemistry environment from a normal water chemistry environment also changes the chemical environment for the BWR from an oxidizing regime to a reducing regime. As

such, the iron-based oxide films existing in the primary system are gradually transformed from a hematite structure to a substituted spinel structure. This conversion process increases the transport of insoluble corrosion products, which subsequently deposit on the recirculation system piping and component surfaces, causing increased dose rates and exposures. The intensity of this process appears to be dependent on the amount of hematite-type iron in the plant, the amount of hydrogen injected, and the operation of the hydrogen injection system.

The injection of depleted zinc oxide (DZO) into the feedwater is a means of controlling Co-60 transport by retaining Co-60 on the fuel surfaces in a tenacious spinel form. However, the high crud loading from the feedwater input will lead to higher dose rates and increase the likelihood of hot spots. Laboratory studies and BWR HWC operating data has shown that shutdown dose rates can be 25-50% higher with frequent cycling of the hydrogen system compared to steady-state moderate HWC.

12.1.3.1 General ALARA Techniques

Described below are several general ALARA techniques. These methods will be incorporated into preplanning of tasks and procedure development. Further information on ALARA techniques incorporated into procedures is given in Section 12.5.

a. Permanent shielding is used, where possible, by having workers stay behind walls or in areas of low-level radiation areas when not actively involved in direct work in the radiation area. On some jobs, temporary shielding, such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment may be used. Temporary shielding will be used only if the total exposure, which includes exposure received during installation and removal, will be effectively reduced. Experience with such operations will assist in developing guidelines in this area (see subsection 12.5.3.2).

b. Systems and other pieces of equipment which are subject to crud buildup, such as reactor water cleanup system, residual heat removal system, liquid radwaste system, various pumps, filters, and demineralizers, have been equipped with connections which can be used for flushing the system to eliminate potential hot-spot buildup.

Prior to performing maintenance work, consideration will be given to flushing and/or chemically decontaminating the system or piece of equipment in order to reduce the crud levels and hence personnel exposure.

c. Work involving a projected collective whole body dose of greater than 1 person-rem will be reviewed for ALARA considerations prior to performing the work. The purpose of this review is to carefully prepare for the job so it can be performed in a proper, safe manner with a minimum of personnel exposure.

d. On complex jobs or jobs with exceptionally high radiation levels, dry-run training may be utilized, and in some cases mock-ups are used to familiarize the workers with the operations they must perform at the jobsite. These techniques will assist in improving worker efficiency and thus minimize the amount of time spent in the radiation field. Normally these efforts will be documented and the experience used to improve future efforts (see subsection 12.5.3.2).

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

f. For long-term repair jobs, consideration will be given to setting up a communications network, such as soundpowered telephones or closed circuit television, to assist support and supervisory personnel in checking on work progress from a lower radiation area.

g. On some jobs, special tools or jigs will be used when their use would permit the job to be performed more efficiently or would prevent errors, thus reducing the time spent in a radiation area. Special tools may also be used if their use would increase the distance from the source to the worker thereby reducing the exposure received. These special tools will be used only if the total exposure, including that received during installation and removal, is significantly reduced. Experience in this area will assist in improving future efforts.

h. Controlled access area entry, exit and changing areas will be established in low-level radiation areas to allow personnel access and exit in as low a level of radiation as practicable. This is done because personnel may spend a significant amount of time changing protective clothing and respiratory equipment in those access areas. These access points will be set up to limit the spread of contamination from the jobsite to as small an area as practicable.

i. Protective clothing and respiratory equipment will be selected to minimize the discomfort of workers so that efficiency will be increased and less time will be spent in radiation areas. Protective clothing prescribed by radiation protection will be commensurate with the hazards involved and the requirements cannot be modified by other personnel.

j. Contamination containments, i.e., glove bag, poly bottles, tents, etc., will be used where practicable to allow personnel to work on highly contaminated equipment while minimizing the spread of contamination during the work. Often these efforts will be documented and experience in this area will assist in improving future efforts (see subsection 12.5.3.2).

k. Individuals will be instructed to remain in low-level radiation areas as much as possible, consistent with performing their assigned jobs. In addition, on certain jobs, detailed maps will be provided with the Radiation Work Permit to clearly delineate the areas of high radiation levels to prevent inadvertent entry into an area of much higher radiation level and to identify lower-level radiation areas.

l. Personnel will be assigned self-reading dosimeters to allow determination of accumulated exposure at any time during the job.

m. On jobs where general area radiation levels are unusually high, a timekeeper will keep track of the total exposure. This will help ensure personnel do not exceed applicable dose limits.

n. On major maintenance jobs, especially those which involve high or complex-radiation levels, the job preplanning will include estimates of the person-rem needed to complete the job. At the completion of the work, a debriefing session will be held with the personnel who performed the work (when practical) in an effort to determine how the work could have been completed more efficiently, resulting in less accumulated exposure. This information, together with the procedures used and actual person-rem expended, will be compiled and filed for future reference. All radiation aspects, i.e., radiation, contamination, airborne radioactivity, and personnel contamination (external or internal), will be compiled and filed for future reference to provide guidance during preplanning of future similar work situations. This will incorporate experience gained in performing these tasks into future work efforts.

12.1.4 Reference

1. Pelletier, C.D., et al, National Environmental Studies Project, Computation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, Atomic Industrial Forum, Sept. 1974.

12.2 RADIATION SOURCES

12.2.1 Contained Sources

12.2.1.1 General

With the exception of the vessel and drywell shields, shielding designs are based on fission product and activation product sources consistent with Section 11.1. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience supports a lower annual average than the design average (Ref. 1). It should be noted that activation products, principally nitrogen-16, control shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transit decay, while at the same time providing for transient increase of the noble gas source, daughter product formation and energy level of emission. Areas where fission products are significant relative to nitrogen-16 include: the condenser offgas system downstream of the steam jet air ejector, liquid and solid radwaste equipment, portions of the reactor water cleanup system, and portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

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

12.2.1.2 Containment

12.2.1.2.1 Reactor Vessel Sources

12.2.1.2.1.1 Radiation from the Reactor Core

12.2.1.2.1.1.1 General

The information in this section defines a reactor vessel model and the associated gamma and neutron radiation sources. This section is designed to provide the data required for calculations beyond the vessel. The data selected were not chosen for any given program, but were chosen to provide information for any of several shield program types. In addition to the source data, calculated

radiation dose levels are provided at locations surrounding the vessel. This data is given as a potential check point for calculations by shield designers.

12.2.1.2.1.1.2 Physical Data

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

12.2.1.2.1.1.3 Core Boundary Neutron Fluxes

Table 12.2-2 presents peak axial neutron multigroup fluxes at the core equivalent radius. The core equivalent radius is a hypothetical boundary enclosing an area equal to the area of the fuel bundles and the coolant space between them. The peak radial flux occurs adjacent to the portion of the core with the greatest power. As shown by the data in Table 12.2-1, this point is below the core midplane. Since this data is calculated with a core equivalent radius, the flux represents a mean flux in the azimuthal angle around the core. While the flux within any given energy group is not known within a factor of 2, the total calculated core boundary flux is estimated to be within ± 50 percent.

12.2.1.2.1.1.4 Gamma Ray Source Energy Spectra

Core Spectra - Table 12.2-3 presents average gamma ray energy spectra per watt of reactor power in both core and non-core regions. In Table 12.2-3, part A, the energy spectra in the core are presented. The energy spectra in the core represent the average gamma ray energy released by energy group per watt of core thermal power. The energy spectra in MeV per sec per watt can be used with the total core power and power distributions to obtain the source in any part of the core.

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

Post Operation Gamma Ray Energy Spectra - Table 12.2-3, part B, gives a gamma ray energy spectrum in MeV per sec per watt in spent fuel as a function of time after operation. The data was prepared from tables of fission product decay gamma fitted to integral measurements for operation times of 108 seconds or approximately 3.2 years. To obtain shutdown sources in the core, the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectra and the thermal power the element contained during operation.

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

12.2.1.2.1.1.5 Gamma Ray and Neutron Fluxes Outside Vessel

Tables 12.2-4 and 5 present the maximum neutron and gamma ray fluxes outside the vessel. The maximum flux occurs on the vessel opposite the portion of the core with the maximum power level.

This elevation can be located using the data from Table 12.2-1. The fluxes at this elevation are based on a mean radius core and do not show azimuth angle variations.

12.2.1.2.1.1.6 Gamma Ray Dose Rates and Fast Neutron Fluxes at the Vessel

The calculated fast neutron flux of neutrons with energy greater than 100 keV outside of the reactor vessel is 1.36 x 10^9 neutrons/ cm^2-sec. The calculated gamma ray dose rate is 2.69 x 10^4 rad/ hour.

12.2.1.2.2 Radioactive Sources in the Reactor Water, Steam, and Offgas

The radioactive sources in the reactor water, steam and offgas are discussed in Chapter 11. This material provides the concentrations during normal operation of the radioisotopes in the reactor vessel or leaving the reactor vessel.

12.2.1.2.3 Reactor Water Cleanup System

The radioactive sources in the cleanup system are the result of the activity in the reactor water in transit through the system or accumulation of radioisotopes removed from the water. Components for this system include regenerative and non-regenerative heat exchangers, pumps, valves, filter demineralizers, and the backwash receiving tank. The system is described in subsection 5.4.8. The N-16 sources in the regenerative heat exchanger are 0.75 μCi/cc and in the non-regenerative heat exchanger are 0.40 μCi/cc. The other gamma sources are given in Table 12.2-6. These sources are present in the filters and receiving tank during all modes of operation. Therefore, backwashing capability is provided to remove the residual activity for effective radwaste handling.

12.2.1.2.4 Main Steam System

All radioactive materials in the main steam system result from radioactive sources carried over from the reactor during plant operation. In most of the components carrying live steam, the source is dominated by N-16. In components where the N-16 has decayed, the other activities carried by the steam become significant. During plant shutdown, there is a residual activity resulting from prior plant operations. The main steam pipe chase sources are in Table 12.2-7.

12.2.1.2.5 Transverse Incore Probe System

The radiation source for the transverse incore probe system (TIP) is provided in Table 12.2-8. The radiation source is based upon location within the core and residence time. As indicated in the tables, the TIP system consists of three components for shielding calculations, the fissionable material, nonfissionable material, and the cable. Sources are provided for each component as a function of irradiation and decay times.

12.2.1.2.6 Reactor Startup Source

The reactor startup source is shipped to the site in a special cask designed for shielding. The source is transferred under water while in the cask and loaded into beryllium containers.

This is then loaded into the reactor while remaining under water. The source remains within the reactor for its lifetime. Thus, no unique shielding requirements after reactor operation are required.

12.2.1.2.7 Additional Sources in the Containment

The radiation source terms for other components in the containment are provided in Table 12.2-9.

12.2.1.3 Auxiliary Building

12.2.1.3.1 Radioactive Sources in the Engineered Safeguard Systems

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

Although the RHR steam condensing mode is permanently disabled, the design gamma source strengths in the residual heat removal (RHR) and reactor core isolation cooling (RCIC) systems were based upon the reactor hot standby (5 percent power) mode. In this mode these systems condense the reactor steam and recirculate the reactor coolant to remove reactor heat. The sources in the RHR and

RCIC systems are the nitrogen-16 activities in the volumes of reactor steam contained in these systems. These sources are provided in Table 12.2-10.

The design gamma source strengths from fission products in the engineered safeguard systems following shutdown is typified by the source strength and fission product inventories for the system given in Table 12.2-11. In the shutdown mode the RHR system recirculates reactor coolant to remove decay heat. The system is operated from approximately 2-4 hours after shutdown until the end of the refueling period. The source in the RHR system is the activity in the volume of reactor water contained in the system. This includes the increase of activity as a result of depressurization.

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

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

12.2.1.3.2 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in subsection 12.2.1.2.1.1.4, Post-Operation, (Table 12.2-3) in terms of MeV per sec per watt.

12.2.1.3.3 Fuel Pool Cooling and Cleanup System

Sources in the fuel pool cooling and cleanup (FPCC) system are a result of transfer of radioactive isotopes from the reactor coolant into the spent fuel pool during refueling operations. The reactor coolant activities for fission, corrosion, and activation products (Section 11.1) are decayed for the amount of time required to remove the reactor vessel head following shutdown, are reduced by operation of the RWCU system filter demineralizers following shutdown, and are diluted by the total volumes of the water in the reactor vessel, refueling pool, and spent fuel pool. This activity then undergoes subsequent decay and accumulation on the FPCC filters and demineralizers. The activities in the FPCC components are given in Table 12.2-12.

12.2.1.3.4 Additional Sources in the Auxiliary Building

The radiation source terms used for shielding of other components in the auxiliary building are provided in Table 12.2-13.

12.2.1.4 Turbine Building

12.2.1.4.1 Turbine System

Piping and equipment which contain steam are sources of radiation due to the presence of nitrogen-16, the predominant source of activity during operation. Fission product gases (xenon and krypton) and gas daughter products (oxygen-19 and nitrogen-17) present in steam and condensate are considered as activity sources. Another source is the carryover of iodine and other fission products. The carryover is conservatively assumed to be two percent by weight for halogens and 0.1 percent by weight for other fission products. The N-16 concentrations in equipment are listed in Table 12.2-14. (Due credit for decay transit for the short-lived N-16 has been factored into the evaluation.)

12.2.1.4.2 Condensate System

The sources in the condensate system are based on decayed main steam activities.

Almost all noncondensables entering the condenser are removed by the air ejectors. Because of this and the relatively long holdup time, the nitrogen-16 and other gaseous activity will be very minor in the hotwell and almost negligible in the remainder of the power conversion system. However, the activity caused by activated corrosion and fission products must be considered. The activity of the water in the hotwell is determined by considering it as a continuously operated tank.

There is some activity in the condensate system, the source being primarily the activated corrosion and fission products. These sources are reduced from reactor water concentrations by the carryover factors. The sources are in the condensate lines and the precoat filters and demineralizers. The accumulation of activity on a filter or demineralizer can be determined by the following equation for isotopic buildup:

$$D_i(t) = \frac{KA_i F f_i}{\lambda_i}(1 - e^{-\lambda_i t})$$

where:

$D_i(t)$ = activity of isotope, i, present on the filter or demineralizer at time, t(μCi)

t = time of operation (sec)

λ_i = decay constant of isotope, i (sec^{-1})

f_i = fraction of isotope, i, retained by filter

F = flow rate of fluid (gal/min)

A_i = activity of isotope, i, in coolant entering demineralizer or filter (μCi/cc)

K = conversion constant = 63.08

For the condensate demineralizer, f_i is assumed to be one for corrosion and fission products and zero for gases.

The activities in the condensate and feedwater systems are listed in Table 12.2-15.

The activities used to determine shielding for the Advanced Resin Cleaning subsystem of the Condensate Cleanup System were based upon actual GGNS plant data in lieu of the activities listed in Table 12.2-15. The use of plant historical data is consistent with Position 2 of NRC Regulatory Guide 8.8 which indicates that design basis assumptions for activation source terms should be based on measurements and experience gained from operating stations of similar design.

12.2.1.5 Radwaste Building

12.2.1.5.1 Liquid and Solid Radwaste Systems

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

The radionuclide sources in the process fluids at various locations in the radwaste systems such as pipes, tanks, filters, demineralizers, and evaporators used in shielding calculations are listed in Section 11.2.

12.2.1.5.2 Offgas System

The offgas filters are located in the radwaste building and contain sources of radiation based on the extraction of noncondensable gases from the main condenser by the main condenser air removal system. The N-16 activity in the radwaste building is negligible due to decay. Therefore, the predominant radiation sources are the fission product gases, xenon and krypton, and their daughter products. The source terms used for shielding are given in Section 11.3.

12.2.1.5.3 Sources Resulting from Design Basis Accidents

The radiation sources from design basis accidents include the design basis inventory of radioactive isotopes in the reactor coolant, plus postulated fission product releases from the fuel. Accident parameters and sources are discussed in Chapter 15.

12.2.1.6 Stored Radioactivity

The principal sources of activity not stored inside the plant structures are the refueling water storage tank (RWST), the condensate storage tank (CST) and the Large Component Storage Building (LCSB). The CST is expected to contain concentrations of radionuclides which yield a surface dose rate of 0.5 mrem/hr or less. The RWST is expected to have a maximum contact dose of less than 10 mrem/hr when the water is returned from the refueling pool. This will be rapidly reduced by processing through the fuel pool cooling and cleanup filter and demineralizer. If dose rates on the outside of the CST or RWST exceed those stated above, the CST/RWST will be posted in accordance with 10CFR20. The area monitoring program will be used to verify compliance with 10CFR20. Because the refueling water storage tank (RWST) and the condensate storage tank (CST) are outside the plant structures, airborne concentrations in the vicinity of these tanks will be minimal. Any possible airborne activity will be dispersed in the open atmosphere. As a result, the inhalation exposures to plant personnel in the vicinity of these tanks will be negligible. The Large Component Storage Building (LCSB) is a radioactive materials storage area located in the Northwest laydown area as shown in Figure 2.1-001. Several components were replaced during

the Extended Power Uprate (EPU) at GGNS. This building serves as permanent storage for these components until decommissioning. They include the steam dryer, both moisture separator reheaters, 9 feedwater heaters, both reactor feedpump turbines and their inner casings and the high pressure turbine rotor. The total expected volume of the major components contributing to offsite dose is approximately 39,000 cubic feet. The principal sources of radioactivity are from solid activated corrosion product buildup on the steam dryer, moisture separator heaters, and the feedwater heaters. The maximum total quantity of stored radioactivity in the LCSB contributing to offsite dose is 960 curies. The activities in the stored components are listed in Table 12.2-19. The LCSB is designed to limit calculated dose rates to within the limits of 10CFR20 and 40CFR190. The calculated dose rate at the site area boundary, approximately 400 feet north of the LCSB, is less than 0.5 mrem/yr. Other sources of activity not stored in the plant include storage areas for radioactive material, sources, holding areas for receiving radioactive sources, and holding areas for radioactive waste shipments. No other radioactive wastes are normally stored outside the plant structures. All spent fuel is stored in the spent fuel pool until it is placed in dry fuel storage casks and moved to the ISFSI, or in the spent fuel shipping cask for transport offsite.

The GGNS ISFSI (Independent Spent Fuel Storage Installation) storage pad is located at the north end of the GGNS plant site and at a location north of the canceled Unit 2 Containment and Turbine Building (see UFSAR Figure 1.2-001 and 3.4-001). The pad stores spent nuclear fuel. Detailed design and radiological information is provided in the NRC Certificate of Compliance (CoC) 72-1014, HI-STORM 100 FSAR HI-2002444, and the GGNS HI-STORM 100 10CFR72.212 Evaluation Report. Additional discussions are also provided UFSAR Chapters 1.2, 3.4, and 9.1. The ISFSI FSAR is maintained in accordance with 10CFR72.

Storage space is allocated in the radwaste building for storage of solidified spent resins, evaporator bottoms, and chemical wastes. Radioactive wastes stored inside plant structures are shielded such that there is zone A access outside the structure. If it becomes necessary temporarily to store radioactive wastes or to store radioactive materials outside plant structure, adequate radiation protection measures will be taken by the radiation protection staff.

12.2.2 Airborne Radioactive Material Sources

This subsection deals with the source models, and parameters required to evaluate airborne concentrations of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected. Expected reactor coolant specific activities for water and steam listed in Tables 11.2-9 and 11.3-9 were used as sources for leakage from components. During movement of spent fuel, the iodines are expected to be the largest contributors to airborne concentrations, since the partition factor for particulates between the vapor and liquid phases at the pool surfaces will be much lower than the partition factor for the iodines. Expected iodine concentrations in the spent fuel pool and upper containment pool during refueling are shown in Table 12.2-17. The airborne radioactive material sources resulting from reactor vessel head and internals removal have been determined from operating plant experience. The major radioisotopes found were I-131, Co-60, and Mn-54, with Nb-95, Zr-95, Ru-103, and Ce-144 at moderate concentrations, and with Ce-141, Cs-137, Co-58, and Cr-51 at low concentrations. The radioactive particulates ranged as high as 2×10^{-8} μCi/cc and the I-131 as high as 4×10^{-8} μCi/cc.

To minimize the containment airborne radioactivity contribution due to removal of the reactor pressure vessel head, the head will be vented either to the drywell purge exhaust system or to the main condenser with vacuum supplied by the mechanical hoggers. Portable HEPA systems may be used, if necessary, with supply trunks creating a small negative pressure on the open vessel cavity. The release points to atmosphere will be monitored by the radiation monitoring system to ensure that radiological effluent limits will not be exceeded. The head will be shrouded when necessary to reduce contamination potential to acceptable levels on the operating floor. Both of these methods will significantly reduce the airborne radioactivity in the containment.

The airborne radioactive material sources resulting from relief valve venting are discussed in subsection 12.4.1.2, and the isotopic sources are listed in Table 12.4-6.

Radwaste building tanks are filled from the top and, as the water splashes into the tanks, dissolved and entrained radioactivity will be liberated. This activity will not be released into the atmosphere in the rooms because the tank vents are connected directly to the building ventilation system. More radioactivity

is expected to be released in this manner than would result from small leaks from the shaft seals in the pump rooms. Furthermore, more than 90 percent of the pumps have flushing lines connected to the pump shaft seals. Condensate water via the condensate and refueling water transfer and storage system is used to prevent process fluids from coming into contact with the seals.

Therefore, 90 percent of the radwaste building ventilation releases were assumed to come from the tank vents. The remaining 10 percent were assumed to be equally divided between the valve and processing equipment rooms and the pump rooms. The airborne concentrations of radionuclides in the pump rooms were calculated by assuming that 5 percent of the releases from the radwaste building ventilation system for iodines and particulates given in Table 2-9 of NUREG-0016 (Ref. 2) are from the pump room ventilation system. It was conservatively assumed that all of the pump room releases were from the regeneration solution receiving pump room. Since this room is small and has a low ventilation exhaust rate, the airborne concentration in this room is expected to be conservatively higher than other pump rooms. The results are shown in Table 12.2-18.

Measurements from operating BWRs indicate airborne concentrations are approximately 10^{-9} and 10^{-11} μCi/cc for particulate and I-131, respectively, in the radwaste building pump rooms. Calculated airborne concentrations (Table 12.2-18) for Grand Gulf yield values of approximately 3.2×10^{-9} and 9.6×10^{-10} μCi/cc for particulates and I-131 for the rooms. Therefore, it can be concluded that the choice of 5 percent of the releases from the radwaste building ventilation system to be from the pump room is conservative.

Equipment cubicles, corridors, and operating areas normally occupied by operating personnel do not contain significant radioactive sources. As indicated in subsection 12.3.3.3.2.a, the air flow for airborne radioactivity is from the area of lowest potential radioactivity to areas of increasing potential for airborne radioactivity until the air is finally exhausted. In this manner, the normally occupied areas which do not have sources will normally not have airborne radioactivity. Data from operating BWRs corroborates the general lack of airborne activity in corridors and normally occupied operating areas. Routine surface contamination swipe samples will be performed to verify the general lack of surface contamination in corridors and normally occupied operating areas. Equipment which is radioactive

and has the potential for leakage has been installed in separate shielded compartments which are not normally occupied due to radiation levels.

Radiation exposures due to airborne radioactivity are expected in the containment, SFP area of the auxiliary building and the pump rooms in the radwaste building. The assumptions and parameters required to calculate isotopic airborne concentrations are listed in Table 12.2-16. A detailed listing of the expected airborne isotopic concentrations in these regions is given in Table 12.2-18.

The design of the plant, along with the use of proper radiological controls, personal protection methods, and the expected weekly occupancy in the regions, ensures that the expected airborne isotopic concentration will be below the DAC value listed in 10CFR20, Appendix B, Table 1.

12.2.2.1 Models for Calculating Airborne Concentrations

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

$$C_i(t) = (L \cdot R)_i \quad A_i \quad (P \cdot F)_i \quad (1 - e^{-\lambda_{Ti} t})/V\lambda_{Ti} \qquad (1)$$

where:

$(L \cdot R)_i$ = leak or evaporation rate of the i^{th} radioisotope in gm/sec, in the applicable region

and

A_i = activity concentration of the i^{th} leaking or evaporating radioisotope in μCi/gm

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

λ_{Ti} = total removal rate constant for the i^{th} radioisotope in sec^{-1} from the applicable region

$(\lambda d_i + \lambda e)$ = (λd_i and λ^e are the removal rate constants in sec^{-1} due to radioactive decay and the exhaust from the applicable region respectively for the i^{th} radioisotope)

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

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

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

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

$$C_{Eqi} = (L \cdot R)_i \quad A_i \quad (P \cdot F)_i \quad / V\lambda_{Ti} \qquad (2)$$

With high exhaust rates, this peak concentration will be reached within a few hours. The airborne concentrations in the containment and in the refueling area of the auxiliary building are calculated using Eq. (2).

The equilibrium air concentrations in the pump room are calculated as follows:

$$C_{Eqi} = \frac{(RR_i)\frac{\mu Ci}{min}}{(ER)\frac{cc}{min}}$$

where

RR_i = Release rate of the i^{th} radioisotope from the room due to room exhaust in $\frac{\mu Ci}{min}$

ER = Exhaust rate from the room in $\frac{cc}{min}$

and

C_{Eqi} = Equilibrium room air concentration of the i^{th} radionuclide in μCi/cc.

12.2.3 References

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

2. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)", NUREG-0016, April 1976.

TABLE 12.2-1: BASIC REACTOR DATA (INITIAL CORE DESIGN AND POWER)

A. Reactor thermal power: 3833 MW

B. Average core power density: 56 Watts/cm^3

C. Physical dimensions*

Radii	Inches
1. Core equivalent radius	95.78
2. Inside shroud radius	105.625
3. Outside shroud radius	107.625
4. Inside vessel radius - nominal	125.5
5. Outside vessel radius - nominal	131.69
6. Outside vessel radius - reinforced - nominal	133.78
7. Shroud head inside radius	204.00
8. Vessel top head inside radius	129.50
9. Vessel bottom head inside radius	138.00

Elevation	Inches
10. Outside of vessel bottom head	-8.77
11. Inside of vessel bottom head	-1.02
12. Vessel bottom head tangent	136.98
13. Bottom of core support plate	205.37
14. Top of core support plate	207.37
15. Bottom of active fuel	216.31
16. Top of reinforced vessel wall	223.58
17. Top of active fuel	364.31
18. Bottom of top guide	374.51
19. Top of fuel channel	380.69
20. Shroud head tangent	426.70
21. Inside of shroud head	456.78
22. Outside of shroud head	458.78
23. Normal vessel water level	553.9
24. Top of steam dryer	724.00
25. Vessel top head tangent	745.00
26. Inside of vessel top head	875.50

TABLE 12.2-1: BASIC REACTOR DATA (INITIAL CORE DESIGN AND POWER) (CONTINUED)

27. Outside of vessel top head	877.50

*Figure 12.2-1 shows the relative locations of the dimensions.

D. Material densities* - gm/cm^3 of region volume

Region	Coolant	UO_2	Zircaloy	304 L Stainless
A	0.740	0.0	0.0	0.600
B	0.599	0.0	0.0	3.160
C	0.306	2.602	0.994	0.056
C-1	0.697	0.0	1.211	0.901
C-2	0.298	0.0	1.168	0.0
D	0.324	0.0	1.141	1.285
E	0.27	0.0	0.0	0.0
F	0.591	0.0	0.0	0.421
G	0.036	0.0	0.0	0.0
H	0.74	0.0	0.0	0.0
I	0.74	0.0	0.0	0.0

TABLE 12.2-1: BASIC REACTOR DATA (INITIAL CORE DESIGN AND POWER) (CONTINUED)

E. Typical core power distributions

Radial Power Distribution		Axial Power Distribution (Typical End-of-Life)	
% of Equivalent Radius	**Relative Power**	**Elevation** in.**	**Relative Power**
0	1.2	-74	0.43
20	1.2	-72	0.45
35	1.19	-60	0.62
50	1.172	-48	0.84
60	1.149	-36	1.06
70	1.096	-24	1.25
80	0.992	-12	1.37
90	0.794	0	1.39
92.4	0.700	12	1.34
94.3	0.600	24	1.20
97.5	0.516	36	1.05
98.6	0.546	48	0.88
99.3	0.587	60	0.71
99.5	0.649	72	0.41
100	0.757	74	0.38

*Figure 12.2-1 shows the region locations.
**Elevations are measured from the midplane of the core.

TABLE 12.2-2: CORE BOUNDARY NEUTRON FLUXES (INITIAL CORE DESIGN AND POWER)

Energy Bounds	Neutron Flux (Neutrons/cm^2-sec)
16.5 MeV	4.6 E + 10
10.0 MeV	6.0 E + 11
6.07 MeV	2.1 E + 12
3.68 MeV	4.2 E + 12
2.23 MeV	4.4 E + 12
1.35 MeV	3.9 E + 12
821 KeV	3.9 E + 12
498 KeV	2.8 E + 12
302 KeV	2.3 E + 12
183 KeV	1.8 E + 12
111 KeV	1.4 E + 12
67.4 KeV	1.1 E + 12
40.8 KeV	1.0 E + 12
24.8 KeV	1.0 E + 12
15.0 KeV	9.5 E + 11
9.12 KeV	9.4 E + 11
5.53 KeV	9.4 E + 11
3.35 KeV	9.1 E + 11
2.03 KeV	1.3 E + 12
1.01 KeV	2.4 E + 12
249 eV	2.5 E + 12
55.6 eV	2.3 E + 12
12.4 eV	4.0 E + 12
0.625 eV	2.4 E + 13

TABLE 12.2-3: GAMMA RAY SOURCE ENERGY SPECTRA (INITIAL CORE DESIGN AND POWER)

A. Gamma Ray Sources in the Core During Operation

Energy Bounds (MeV)	Gamma Ray Source MeV/sec-Watt
16.5	3.0 E + 8
8.0	4.4 E + 9
6.0	3.9 E + 10
4.0	4.9 E + 10
3.0	4.6 E + 10
2.6	6.1 E + 10
2.2	6.8 E + 10
1.8	8.0 E + 10
1.4	9.0 E + 10
1.0	1.4 E + 11
0.5	7.6 E + 10
0.0	

B. Post Operation Gamma Sources in Core* (MeV/sec-Watt)

Energy Bounds (MeV)	Time After Shutdown			
	0 sec	1 Day	1 week	1 month
6.0	8.2 E + 9	<1.0 E + 6	<1.0 E + 6	<1.0 E + 6
4.0	1.8 E + 10	7.0 E + 6	4.6 E + 6	<1.0 E + 6
3.0	1.1 E + 10	5.7 E + 6	3.7 E + 6	<1.0 E + 6
2.6	1.7 E + 10	2.9 E + 8	1.7 E + 8	<1.0 E + 6
2.2	2.1 E + 10	4.5 E + 8	4.0 E + 7	5.2 E + 6
1.8	3.3 E + 10	3.1 E + 9	2.1 E + 9	6.4 E + 8
1.4	3.7 E + 10	2.3 E + 9	1.6 E + 9	1.1 E + 9
0.9	5.1 E + 10	7.5 E + 9	3.8 E + 9	2.1 E + 9

TABLE 12.2-3: GAMMA RAY SOURCE ENERGY SPECTRA (INITIAL CORE DESIGN AND POWER) (CONTINUED)

0.4	1.2 E + 10	1.8 E + 9	8.7 E + 8	3.6 E + 8
0.1				

*Operating history of 3.2 years

C. Gamma Ray Sources in Shroud and Vessel During Operation* (MeV/cm³-sec-Watt)

	Shroud		Vessel	
Energy Bounds (MeV)	Inside Radius	Outside Radius	Inside Radius	Outside Radius
16.5	8.9 E + 1	1.2 E + 1	8.8 E - 2	7.5 E - 3
8.0	1.9 E + 2	2.6 E + 1	5.4 E - 1	4.6 E - 2
6.0	6.7 E + 1	9.4 E + 0	1.9 E - 1	1.6 E - 2
4.0	3.3 E + 1	4.8 E + 0	1.1 E - 1	8.9 E - 3
3.0	2.2 E + 1	3.3 E + 0	7.2 E - 2	5.8 E - 3
2.6	1.4 E + 0	3.1 E + 0	4.0 E - 2	3.2 E - 3
2.2	2.1 E + 1	3.1 E + 0	6.1 E - 2	5.0 E - 2
1.8	2.9 E + 1	4.2 E + 0	9.0 E - 2	7.4 E - 3
1.4	1.5 E + 1	2.4 E + 0	4.3 E - 2	3.5 E - 3
1.0	2.1 E + 1	3.8 E + 0	5.8 E - 2	4.8 E - 3
0.5	4.7 E + 1	6.4 E + 0	8.5 E - 2	7.2 E - 3
0.0				

*Shroud and vessel sources are at core midplane.

TABLE 12.2-4: NEUTRON FLUX OUTSIDE REACTOR VESSEL (INITIAL CORE DESIGN AND POWER)

Upper Energy (eV)	Neutron Flux Vessel Outer Surface	(Neut./cm2-sec) Shield Wall Outer Surface
15.0 + 6	5.80 + 05	1.03 + 02
12.2 + 6	1.92 + 06	3.47 + 02
10.0 + 6	4.28 + 06	8.34 + 02
8.18 + 6	8.41 + 06	1.90 + 03
6.36 + 6	1.26 + 07	3.06 + 03
4.96 + 6	9.68 + 06	2.26 + 03
4.06 + 6	1.62 + 07	2.88 + 03
3.01 + 6	1.85 + 07	6.17 + 03
2.46 + 6	6.00 + 06	3.89 + 03
2.35 + 6	3.64 + 07	1.12 + 04
1.83 + 6	1.17 + 08	2.82 + 04
1.11 + 6	3.17 + 08	6.25 + 04
5.50 + 5	8.15 + 08	1.47 + 05
1.11 + 5	5.56 + 08	1.39 + 05
3.35 + 3	1.17 + 08	6.40 + 04
5.83 + 2	9.25 + 07	8.04 + 04
1.01 + 2	6.12 + 07	7.57 + 04
2.90 + 1	4.00 + 07	5.48 + 04
1.07 + 1	3.74 + 07	6.58 + 04
3.06 + 0	2.12 + 07	4.93 + 04
1.12 + 0	1.27 + 07	4.02 + 04
0.414 + 0	4.48 + 06	2.28 + 04

TABLE 12.2-5: GAMMA FLUX OUTSIDE REACTOR VESSEL (INITIAL CORE DESIGN AND POWER)

Upper Energy(MeV)	Gamma Flux Vessel Outer Surface	(gammas/cm²-sec) Shield Wall Outer Surface
10.0	4.37 + 07	2.47 + 04
8.0	1.31 + 08	2.19 + 05
6.5	1.45 + 08	1.67 + 05
5.0	2.70 + 08	1.35 + 05
4.0	5.86 + 08	1.75 + 05
3.0	4.88 + 08	1.04 + 05
2.5	9.24 + 08	1.24 + 05
2.0	7.05 + 08	1.03 + 05
1.66	8.15 + 08	1.37 + 05
1.33	1.04 + 09	1.59 + 05
1.0	8.08 + 08	1.22 + 05
0.8	1.08 + 09	1.58 + 05
0.6	2.07 + 09	4.07 + 05
0.4	1.38 + 09	2.13 + 05
0.3	2.09 + 09	2.81 + 05
0.2	1.94 + 09	2.32 + 05
0.1	9.27 + 07	1.07 + 04
0.05	8.58 + 04	3.55 + 01

TABLE 12.2-6: RWCU SOURCES ACTIVITIES (MCI/ML)

Isotope	RWCU Backwash Receiving Tank	RWCU Demineralizer
Ru - 103	1.14 + 00	4.01 - 02
Ru - 106	2.70 - 01	5.35 - 03
La - 140	2.55 - 02	0.0
La - 141	3.14 - 03	0.0
Ce - 141	2.01 + 00	7.90 - 02
Ce - 143	8.75 - 02	2.10 - 02
Ce - 144	3.61 + 02	7.30 - 02
Pr - 143	9.34 - 01	6.95 - 02
Nd - 147	2.87 - 01	2.55 - 02
Nb - 95m	5.27 - 02	0.0
Nb - 95	3.83 + 00	8.25 - 02
Mo/Tc - 99	1.67 + 03	2.31 + 01
Y - 89m	1.58 - 02	0.0
Y - 90	2.41 + 01	0.0
Y - 91M	3.10 - 04	0.0
Y - 91	2.86 + 00	0.0
Y - 92	1.07 - 12	0.0
Cs - 134	1.79 + 01	3.40 - 01
Cs - 135	3.67 - 09	0.0
Cs - 136	2.44 + 00	1.91 - 01
Cs - 137	2.79 + 01	5.10 - 01
Cs - 138	1.00 + 01	2.56 - 00
Na - 24	2.95 + 00	7.50 - 01
Cr - 51	2.91 + 01	1.30 - 00
Mn - 54	5.58 + 00	1.13 - 01
Mn - 56	1.27 + 01	3.23 - 00
Fe - 59	6.52 + 00	2.16 - 01
Co - 58	5.03 + 02	1.37 + 02
Co - 60	7.64 + 01	1.42 + 00
Ni - 65	7.54 - 02	1.92 - 02
Zr - 95	2.93 + 00	8.20 - 02
Ag - 110m	8.28 + 00	1.69 - 01

**TABLE 12.2-6: RWCU SOURCES
ACTIVITIES (ΜCI/ML) (CONTINUED)**

Isotope	RWCU Backwash Receiving Tank	RWCU Demineralizer
W - 187	7.07 + 00	1.79 + 00
I - 131	0.0	2.36 - 01
I - 132	0.0	6.35 + 00
I - 133	0.0	3.88 + 01
I - 134	0.0	5.10 + 00
I - 135	0.0	1.85 + 01
Br - 83	0.0	7.85 - 01
Br - 84	0.0	3.61 - 01
Br - 85	0.0	2.26 - 02
Sr - 89	0.0	6.25 + 00
Sr - 90	0.0	4.82 - 01
Sr - 91	0.0	1.37 + 01
Sr - 92	0.0	6.75 + 00
Te - 129m	0.0	7.95 - 02
Te - 132	0.0	5.55 + 01
BA - 139	0.0	5.20 + 00
BA - 140	0.0	1.59 + 01
BA - 141	0.0	1.36 + 00
BA - 142	0.0	8.25 - 01

TABLE 12.2-7: MAIN STEAM PIPE CHASE U.R. N-16 SOURCES

Four 28" OD main steam pipes:	50 µCi/gm steam
One 10" OD RCIC steam pipe:	50 µCi/gm steam
One 4" OD RWCU steam pipe:	50 µCi/gm steam

TABLE 12.2-8: MATERIAL COMPOSITION OF THE TIP SYSTEM COMPONENTS AS USED IN ACTIVATION CALCULATIONS

Detector Region

Material	Weight
AISI 304 stainless steel	4gm
Commercially pure titanium	3gm
Fosterite ceramic	0.5gm
Nichrome	0.02gm
Uranium - 235	0.001gm

Cable Region

AISI 3041 stainless steel	0.12gm/in.
AISI C1070 carbon steel	2.1gm/in.
Magnesium oxide	0.12gm/in.

TABLE 12.2-8: MATERIAL COMPOSITION OF THE TIP SYSTEM COMPONENTS AS USED IN ACTIVATION CALCULATIONS (CONTINUED)

TRAVERSING INCORE PROBE DETECTOR DECAY GAMMA ACTIVITIES IN MeV/SEC of 0.001gm of U-235

Decay Time = 10^0 Seconds

Activation Time = 10^2 Seconds

Energy-MeV	
0.1 - 0.4	3.4 + 9
0.4 - 0.9	1.5 + 10
0.9 - 1.35	1.2 + 10
1.35 - 1.8	1.1 + 10
1.8 - 2.2	8.0 + 9
2.2 - 2.6	6.4 + 9
2.6 - 3.0	5.6 + 9
3.0 - 3.5	5.1 + 9
3.5 - 4.0	4.3 + 9
4.0 - 4.5	2.5 + 9
4.5 - 5.0	1.5 + 9
5.0 - 5.5	7.9 + 8

TABLE 12.2-8: MATERIAL COMPOSITION OF THE TIP SYSTEM COMPONENTS AS USED IN ACTIVATION CALCULATIONS (CONTINUED)

TRAVERSING INCORE PROBE DETECTOR DECAY GAMMA ACTIVITIES OF MATERIALS IN THE DETECTOR (EXCLUDING U-235)IN MICROCURIES IN THE IRRADIATED DETECTOR

Decay Time = 10^{0} Seconds

Activation Time = 10^{2} Seconds

Activated Isotopes	
Fe - 59	1.1 + 1
Mn - 56	1.7 + 5
Cr - 51	7.0 + 1
Mn - 54	2.1 + 0
Co - 58M	3.5 + 3
Co - 58	2.2 - 2
Ni - 57	1.1 - 1
Co - 57	6.0 - 7
Ni - 65	4.0 + 2
Co - 60M	7.6 + 3
Co - 60	1.8 - 3
Co - 61	9.6 + 0
Si - 31	2.9 + 1

TABLE 12.2-8: MATERIAL COMPOSITION OF THE TIP SYSTEM COMPONENTS AS USED IN ACTIVATION CALCULATIONS (CONTINUED)

DECAY GAMMA ACTIVITIES OF MATERIALS IN THE CABLE
IN MICROCURIES PER INCH OF IRRADIATED CABLE

Decay Time = 10° Seconds

Activation Time = 10^2 Seconds

Activated Isotopes	
Fe - 59	8.2 + 0
Mn - 56	7.4 + 4
Cr - 51	3.7 + 0
Mn - 54	1.6 + 0
Co - 58M	1.0 + 2
Co - 58	6.5 - 4
Ni - 57	3.3 - 3
Co - 57	1.8 - 8
Ni - 65	1.2 + 1
Co - 60M	2.2 + 2
Co - 60	5.1 - 5
Co - 61	2.8 - 1
Si - 31	8.7 - 1

TABLE 12.2-9: ADDITIONAL SOURCES IN THE CONTAINMENT

Activities (µCi/m1)

Isotope	North-East Equipment Drain Tank	South-East Equipment Drain Tank	Containment Floor Drain Tank
I - 131	2.17 - 04	4.77 - 03	5.28 - 06
I - 132	2.17 - 03	4.77 - 02	5.28 - 05
I - 133	1.46 - 03	3.21 - 02	3.55 - 05
I - 134	4.33 - 03	9.55 - 02	1.06 - 04
I - 135	2.17 - 03	4.77 - 02	5.28 - 05
Br - 83	2.56 - 04	5.64 - 03	6.24 - 06
Br - 84	4.93 - 04	1.09 - 02	1.20 - 05
Br - 85	3.15 - 04	6.95 - 03	7.67 - 06
Sr - 89	4.69 - 05	1.03 - 03	1.15 - 06
Sr - 90	3.52 - 06	7.75 - 05	8.63 - 08
Sr - 91	1.12 - 03	2.46 - 02	2.73 - 05
Sr - 92	1.90 - 03	4.18 - 02	4.65 - 05
Te - 129m	6.07 - 07	1.34 - 05	1.49 - 08
Te - 132	7.45 - 04	1.64 - 02	1.82 - 05
BA - 139	2.93 - 03	6.46 - 02	7.20 - 05
BA - 140	1.37 - 04	3.01 - 03	3.36 - 06
BA - 141	3.33 - 03	7.33 - 02	8.15 - 05
BA - 142	3.13 - 03	6.89 - 02	7.67 - 05
Ru - 103	2.93 - 07	6.46 - 06	7.27 - 09
Ru - 106	3.91 - 08	8.62 - 07	9.59 - 10
Ce - 141	6.07 - 07	1.34 - 05	1.49 - 08
Ce - 143	5.48 - 07	1.21 - 05	0.0
Ce - 144	5.28 - 07	1.16 - 05	1.34 - 08
Pr - 143	5.87 - 07	1.29 - 05	1.44 - 08
Nd - 147	2.15 - 07	4.74 - 06	5.28 - 09
Nb - 95	6.26 - 07	1.38 - 05	1.53 - 08
Mo/Te - 99	5.04 - 04	1.11 - 02	8.63 - 06
Cs - 134	2.35 - 06	5.17 - 05	5.76 - 08

TABLE 12.2-9: ADDITIONAL SOURCES IN THE CONTAINMENT (CONTINUED)

Activities (μCi/m1)

Isotope	North-East Equipment Drain Tank	South-East Equipment Drain Tank	Containment Floor Drain Tank
Cs - 136	1.60 - 06	3.53 - 05	3.93 - 08
Cs - 137	3.71 - 06	8.18 - 05	9.11 - 08
Cs - 138	3.52 - 03	7.75 - 02	8.63 - 05
F - 18	7.87 - 05	1.73 - 03	1.92 - 06
NA - 24	3.91 - 05	8.62 - 04	9.59 - 07
Cr - 51	9.80 - 06	2.16 - 04	2.40 - 06
Mn - 54	7.80 - 07	1.72 - 05	1.92 - 08
Mn - 56	9.80 - 04	2.16 - 02	2.40 - 05
Fe - 59	1.57 - 06	3.45 - 05	3.84 - 08
Co - 58	9.80 - 05	2.16 - 03	2.40 - 06
Co - 60	9.80 - 06	2.16 - 04	2.16 - 04
Ni - 65	5.87 - 06	1.29 - 04	1.44 - 07
Zr - 95	6.07 - 07	1.34 - 05	1.49 - 08
Ag - 110m	1.17 - 06	2.58 - 05	2.88 - 08
W - 187	5.87 - 05	1.29 - 03	1.44 - 06
Total	301. - 02	6.62 - 01	7.32 - 04
Volume (ft^3)	133.7	133.7	133.7

TABLE 12.2-10: RCIC AND RHR SOURCES IN HOT STANDBY (5% POWER CONDENSING) MODE

Entrance Point to RCIC turbine = 12.9 $\mu Ci/gm_{steam}$ N-16

RHR heat exchanger = 2.5 $\mu Ci/gm_{steam}$ N-16

TABLE 12.2-11: RHR SYSTEM ACTIVITIES

Activities (µCi/m1)

Isotope	RHR Pumps	RHR Sumps A & B
I - 131	1.10 - 02	2.20 - 04
I - 132	1.10 - 01	2.20 - 03
I - 133	7.40 - 02	1.48 - 03
I - 134	2.30 - 01	4.60 - 03
I - 135	1.10 - 01	2.20 - 03
Br - 83	1.30 - 02	2.60 - 04
Br - 84	2.80 - 02	5.60 - 04
Br - 85	1.90 - 02	3.80 - 04
Sr - 89	2.30 - 03	2.30 - 06
Sr - 90	1.70 - 04	1.70 - 07
Sr - 91	5.70 - 02	5.70 - 05
Sr - 92	1.00 - 01	1.00 - 04
Te - 129M	2.60 - 04	2.60 - 07
Te - 132	1.10 - 02	1.10 - 05
Ba - 139	1.60 - 01	1.60 - 04
Ba - 140	6.70 - 03	6.70 - 06
Ba - 141	1.90 - 01	1.90 - 04
Ba - 142	1.90 - 01	1.90 - 04
Ru - 103	1.50 - 05	1.50 - 08
Ru - 106	1.90 - 06	1.90 - 09
Ce - 141	3.00 - 05	3.00 - 08
Ce - 143	2.70 - 05	2.70 - 08
Ce - 144	2.60 - 05	2.60 - 08
Pr - 143	2.90 - 05	2.90 - 08
Nd - 147	1.10 - 05	1.10 - 08
Nb - 95	3.10 - 05	0.0
Mo/Tc - 99	3.40 - 02	3.40 - 05
Cs - 134	1.20 - 04	1.20 - 07
Cs - 136	8.00 - 05	8.00 - 08
Cs - 137	1.80 - 04	1.80 - 07
Cs - 138	2.00 - 01	2.00 - 04
Na - 24	2.00 - 03	2.00 - 06
Cr - 51	5.00 - 04	5.00 - 07
Mn - 54	4.00 - 05	4.00 - 08
Mn - 56	5.00 - 02	5.00 - 05
Fe - 59	8.00 - 05	8.00 - 08

TABLE 12.2-11: RHR SYSTEM ACTIVITIES (CONTINUED)

Isotope	RHR Pumps	RHR Sumps A & B
Co - 58	5.00 - 03	5.00 - 06
Co - 60	5.00 - 04	5.00 - 07
Ni - 65	3.00 - 04	3.00 - 07
Zr - 95	6.00 - 05	9.00 - 08
Ag - 110m	6.00 - 05	6.00 - 08
W - 187	3.00 - 03	3.00 - 06
Volume 24.25 ft^3		Volume 224.9 ft^3

TABLE 12.2-12: FUEL POOL COOLING AND CLEANUP SYSTEM ACTIVITIES (µCi/ml)

Isotope	Spent Fuel Pool Filter/ Demineralizer	Backwash Receiving Tank	Fuel Pool Heat Exchanger	Fuel Pool Drain Tank
I - 129	1.79 + 01	5.60 + 01	1.73 - 03	9.36 - 16
I - 131	5.54 - 14	1.73 - 13	1.73 - 02	1.49 - 03
I - 132	3.52 + 00	1.10 + 01	1.16 - 02	4.13 - 03
I - 133	0 + 00	0 + 00	0 + 00	2.90 - 03
I - 134	0 + 00	0 + 00	3.61 - 02	1.66 - 16
I - 135	1.15 - 03	3.60 - 03	1.73 - 02	2.26 - 04
Br - 83	2.97 - 14	9.27 - 14	2.04 - 03	1.11 - 08
Br - 84	0 + 00	0 + 00	4.24 - 03	7.55 - 27
Br - 85	0 + 00	0 + 00	2.83 - 03	0 + 00
Sr - 89	5.38 + 00	1.68 + 01	3.61 - 04	3.53 - 04
Sr - 90	4.26 - 01	1.33 + 00	2.67 - 05	2.67 - 05
Sr - 91	2.67 - 02	8.33 - 02	8.79 - 03	4.38 - 04
Sr - 92	1.04 - 11	3.24 - 11	1.57 - 02	3.25 - 07
Te - 129m	6.75 - 02	2.11 - 01	4.71 - 06	4.54 - 06
Te - 129	0 + 00	0 + 00	0 + 00	2.90 - 06
Te - 132	3.19 + 01	9.97 + 01	5.81 - 03	4.00 - 03
Ba - 137m	0 + 00	0 + 00	0 + 00	2.65 - 05
Ba - 139	0 + 00	0 + 00	2.36 - 02	1.76 - 11
Ba - 140	1.28 + 01	4.00 + 01	1.05 - 03	9.56 - 04
Ba - 141	0 + 00	0 + 00	2.83 - 02	0 + 00
Ba - 142	0 + 00	0 + 00	2.83 - 02	0 + 00
Ru - 103	3.46 - 02	1.08 - 01	2.36 - 06	2.28 - 06
Ru - 106	4.70 - 03	1.47 - 02	2.98 - 07	2.97 - 07
Ce - 141	6.75 - 02	2.11 - 01	4.71 - 06	1.51 - 05
Ce - 143	5.44 - 03	1.70 - 02	4.24 - 06	1.76 - 06
Ce - 144	6.43 - 02	2.01 - 01	4.08 - 06	4.07 - 06
Pr - 143	5.63 - 02	1.76 - 01	4.55 - 06	1.87 - 05
Nd - 147	2.02 - 02	6.30 - 02	1.73 - 06	1.55 - 06
Nb - 95m	0 + 00	0 + 00	0 + 00	2.57 - 08
Nb - 95	7.04 - 02	2.20 - 01	4.87 - 06	4.85 - 06
Mo/Tc - 99	1.21 - 01	3.77 + 01	2.67 - 03	1.71 - 03
Cs - 134	2.99 - 01	9.35 - 01	1.88 - 05	1.88 - 05
Cs - 135	1.54 - 01	4.80 - 01	1.26 - 05	5.90 - 14
Cs - 136	0 + 00	0 + 00	0 + 00	0 + 00
Cs - 137	4.51 - 01	1.41 + 00	2.83 - 05	2.83 - 05
Cs - 138	0 + 00	0 + 00	2.98 - 02	0 + 00
Na - 24	1.96 - 02	6.12 - 02	3.14 - 04	4.51 - 05
F - 18	0 + 00	0 + 00	6.28 - 04	0 + 00

TABLE 12.2-12: FUEL POOL COOLING AND CLEANUP SYSTEM ACTIVITIES (μCi/ml) (CONTINUED)

Cr - 51	1.10 + 00	3.44 + 00	7.85 - 05	7.50 - 05
Mn - 54	9.89 - 02	3.09 - 01	6.28 - 06	6.25 - 06
Mn - 56	1.23 - 12	3.84 - 12	7.85 - 03	9.81 - 08
Fe - 59	1.86 - 01	5.80 - 01	1.26 - 05	1.22 - 05
Co - 58	1.19 + 00	3.72 + 01	7.85 - 04	7.71 - 04
Co - 60	1.25 + 00	3.91 + 00	7.85 - 05	7.85 - 05
Ni - 65	5.92 - 15	1.85 - 14	4.71 - 05	5.46 - 10
Zr - 95	7.14 - 02	2.23 - 01	4.71 - 06	4.62 - 06
Ag - 110m	1.48 - 01	4.64 - 01	9.42 - 06	9.37 - 06
W - 187	2.35 - 01	7.35 - 01	4.71 - 04	1.40 - 04
Kr - 83m	0 + 00	0 + 00	0 + 00	4.79 - 08
Kr - 85m	0 + 00	0 + 00	0 + 00	4.30 - 08
Kr - 85	0 + 00	0 + 00	0 + 00	1.50 - 09
Xe - 131m	0 + 00	0 + 00	0 + 00	9.22 - 07
Xe - 133m	0 + 00	0 + 00	0 + 00	6.86 - 05
Xe - 133	0 + 00	0 + 00	0 + 00	1.22 - 03
Xe - 135m	0 + 00	0 + 00	0 + 00	2.04 - 05
Xe - 135	0 + 00	0 + 00	0 + 00	1.36 - 03
La - 140	0 + 00	0 + 00	0 + 00	5.13 - 04
La - 141	0 + 00	0 + 00	0 + 00	1.36 - 06
La - 142	0 + 00	0 + 00	0 + 00	3.71 - 12
Y - 89m	0 + 00	0 + 00	0 + 00	3.17 - 08
Y - 90	0 + 00	0 + 00	0 + 00	9.73 - 06
Y - 91m	0 + 00	0 + 00	0 + 00	2.92 - 04
Y - 91	0 + 00	0 + 00	0 + 00	5.64 - 05
Y - 92	0 + 00	0 + 00	0 + 00	1.24 - 05
Volume (ft^3)	18	36	54.98	668.5

TABLE 12.2-13: AUXILIARY BUILDING DRAIN TANKS

Activities (µCi/ml)

Isotope	Equipment Drain Transfer Tank	Floor Drain Transfer Tank
I - 131	4.44 - 03	4.44 - 05
I - 132	4.44 - 02	4.44 - 04
I - 133	2.99 - 02	2.99 - 04
I - 134	8.89 - 02	8.89 - 04
I - 135	4.44 - 02	4.44 - 04
Br - 83	5.25 - 03	5.25 - 05
Br - 84	1.01 - 02	1.01 - 04
Br - 85	6.46 - 03	6.46 - 05
Sr - 89	9.62 - 04	9.62 - 06
Sr - 90	7.22 - 05	7.22 - 07
Sr - 91	2.29 - 02	2.29 - 04
Sr - 92	3.89 - 02	3.89 - 04
Te - 129m	1.24 - 05	1.24 - 07
Ba - 139	6.01 - 02	6.01 - 04
Ba - 140	2.81 - 03	2.81 - 05
Ba - 141	6.82 - 02	6.82 - 04
Ba - 142	6.41 - 02	6.41 - 04
Ru - 103	6.01 - 06	6.01 - 08
Ru - 106	8.02 - 07	8.02 - 09
Ce - 141	1.24 - 05	1.24 - 07
Ce - 143	1.12 - 05	1.12 - 07
Ce - 144	1.08 - 05	1.08 - 07
Pr - 143	1.20 - 05	1.20 - 07
Nd - 147	4.41 - 06	4.41 - 06
Nb - 95	1.28 - 05	1.28 - 07
Mo/Tc - 99	7.22 - 03	7.22 - 05
Cs - 134	4.81 - 04	4.81 - 06
Cs - 136	3.29 - 05	3.29 - 07
Cs - 137	7.62 - 05	7.62 - 07
Cs - 138	7.22 - 02	7.22 - 04
F - 18	1.62 - 03	1.62 - 05
Na - 24	8.02 - 04	8.02 - 06
Cr - 51	2.00 - 04	2.00 - 06
Mn - 54	1.60 - 05	1.60 - 07
Mn - 56	2.00 - 02	2.00 - 04

TABLE 12.2-13: AUXILIARY BUILDING DRAIN TANKS (CONTINUED)

Activities (µCi/ml)

Isotope	Equipment Drain Transfer Tank	Floor Drain Transfer Tank
Fe - 59	3.21 - 05	3.21 - 07
Co - 58	2.00 - 03	2.00 - 05
Co - 60	2.00 - 04	2.00 - 06
Ni - 65	1.20 - 04	1.20 - 06
Zr - 95	1.24 - 05	1.24 - 07
Ag - 110m	2.41 - 05	2.41 - 05
W - 187	1.20 - 03	1.20 - 05
Volume (ft^3)	668.5	668.5

TABLE 12.2-14: N-16 CONCENTRATIONS IN EQUIPMENT IN THE TURBINE BUILDING

Turbines	50 μCi/gm
Condenser	50 μCi/gm
Moisture separator/reheater	50 μCi/gm
Reactor feed pump turbine	0.010 μCi/cc
Heater drain pump	0.272 μCi/cc
High pressure feedwater heater #5	0.135 μCi/cc
High pressure feedwater heater #6	0.347 μCi/cc
Offgas condenser	86 μCi/cc
Offgas preheater	20 μCi/cc
Offgas recombiner	13 μCi/cc

TABLE 12.2-15: CONDENSATE AND FEEDWATER SOURCES (µCi/ML)

Isotope	Condensate Precoat Filter	Condensate Demineralizer	Reactor Feed Pumps	Heater Drain Tank
I - 129	0 + 00	0 + 00	0 + 00	1.78 - 21
I - 131	8.06 - 02	4.33 + 01	2.20 - 04	2.20 - 04
I - 132	4.69 - 01	5.11 + 00	2.20 - 03	2.19 - 03
I - 133	5.11 - 01	3.13 + 01	1.48 - 03	1.48 - 03
I - 134	4.89 - 01	3.91 + 00	4.60 - 03	4.52 - 03
I - 135	6.65 - 01	1.50 + 01	2.20 - 03	2.20 - 03
Br - 83	5.69 - 02	6.35 - 01	1.30 - 05	1.29 - 05
Br - 84	3.51 - 02	2.69 - 01	2.70 - 05	2.63 - 05
Br - 85	2.13 - 03	1.63 - 02	1.80 - 05	1.37 - 05
Sr - 89	8.84 - 04	2.22 + 00	2.30 - 06	2.30 - 06
Sr - 90	6.64 - 05	3.03 - 01	1.70 - 07	1.70 - 07
Sr - 91	1.82 - 02	5.61 - 01	5.60 - 05	5.59 - 05
Te - 129m	1.14 - 05	2.24 - 02	3.00 - 08	3.00 - 08
Te - 129	0 + 00	0 + 00	0 + 00	1.90 - 11
Te - 132	1.38 - 02	3.01 + 00	3.70 - 05	3.70 - 05
Ba - 137m	0 + 00	0 + 00	0 + 00	3.42 - 09
Ba - 139	2.39 - 02	2.11 - 01	1.50 - 04	1.48 - 04
Ba - 140	2.57 - 03	2.18 + 01	6.70 - 06	6.70 - 06
Ba - 141	6.90 - 03	5.27 - 02	1.80 - 04	1.71 - 04
Ba - 142	3.80 - 03	2.90 - 02	1.80 - 04	1.66 - 04
Ru - 103	5.53 - 06	1.20 - 02	1.50 - 08	1.50 - 08
Ru - 106	7.37 - 07	3.08 - 03	1.90 - 09	1.90 - 09
La - 140	0 + 00	0 + 00	0 + 00	1.92 - 10
La - 141	0 + 00	0 + 00	0 + 00	5.30 - 08
La - 142	0 + 00	0 + 00	0 + 00	1.47 - 07
Ce - 141	1.14 - 05	2.17 - 02	3.00 - 08	3.00 - 08
Ce - 143	9.91 - 06	9.39 - 04	2.70 - 08	2.70 - 08
Ce - 144	1.00 - 05	4.05 - 02	2.60 - 08	2.60 - 08
Pr - 143	1.10 - 05	9.89 - 03	2.90 - 08	2.90 - 08
Nd - 147	4.04 - 06	2.96 - 03	1.10 - 08	1.10 - 08
Nb - 95m	0 + 00	0 + 00	0 + 00	7.68 - 15
Nb - 95	1.18 - 05	2.36 - 02	3.10 - 08	3.10 - 08
Mo/Tc - 99	7.74 - 02	1.22 + 00	1.70 - 05	1.70 - 05
Cs - 134	4.43 - 05	1.94 - 01	1.20 - 07	1.20 - 07
Cs - 136	3.01 - 05	2.58 - 02	8.00 - 08	8.00 - 08
Cs - 137	7.01 - 05	3.21 - 01	1.80 - 07	1.80 - 07
Cs - 138	1.28 - 02	9.83 - 02	1.90 - 04	1.85 - 04
F - 18	1.52 - 02	1.49 - 01	0 + 00	0 + 00
Na - 24	6.73 - 04	3.06 - 02	2.00 - 06	2.00 - 06
Cr - 51	1.84 - 04	3.12 - 01	5.00 - 07	5.00 - 07
Mn - 54	1.47 - 05	6.07 - 02	4.00 - 08	4.00 - 08

TABLE 12.2-15: CONDENSATE AND FEEDWATER SOURCES (µCi/ML)

Isotope	Condensate Precoat Filter	Condensate Demineralizer	Reactor Feed Pumps	Heater Drain Tank
Mn - 56	1.18 - 02	1.31 - 01	5.00 - 06	4.97 - 05
Fe - 59	2.94 - 05	6.90 - 02	8.00 - 08	8.00 - 08
Co - 58	1.84 - 04	5.42 + 00	5.00 - 06	5.00 - 06
Ni - 65	6.74 - 05	7.78 - 04	3.00 - 07	2.98 - 07
Zr - 95	1.14 - 05	3.23 - 02	3.00 - 08	3.00 - 08
Ag - 110m	2.21 - 05	8.87 - 02	6.00 - 80	6.00 - 08
W - 187	1.05 - 03	7.29 - 02	3.00 - 06	3.00 - 06
N - 13	0 + 00	0 + 00	5.20 - 05	4.76 - 05
O - 19	0 + 00	0 + 00	2.20 - 04	2.32 - 04
Kr - 83m	0 + 00	0 + 00	0 + 00	8.01 - 09
Kr - 85m	0 + 00	0 + 00	0 + 00	4.66 - 09
Kr - 85	0 + 00	0 + 00	0 + 00	2.89 - 17
Xe - 131m	0 + 00	0 + 00	0 + 00	2.64 - 13
Xe - 133m	0 + 00	0 + 00	0 + 00	4.38 - 11
Xe - 133	0 + 00	0 + 00	0 + 00	6.47 - 10
Xe - 135	0 + 00	0 + 00	0 + 00	4.01 - 08
Xe - 137	0 + 00	0 + 00	0 + 00	1.26 - 08
Y - 89m	0 + 00	0 + 00	0 + 00	1.14 - 11
Y - 90	0 + 00	0 + 00	0 + 00	3.06 - 12
Y - 91m	0 + 00	0 + 00	0 + 00	4.67 - 08
Y - 91	0 + 00	0 + 00	0 + 00	1.78 - 11
Y - 92	0 + 00	0 + 00	0 + 00	3.26 - 08
Volume (ft^3)	381.6	300	95.6	2072.4

TABLE 12.2-16: PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVITY CONCENTRATIONS

			Pounds/Hour
a.	Leak Rates		
1.	Reactor water leak into the containment during power operation		50
2.	Reactor steam leak into the containment during power operation		5
3.	Safety/relief valve leak into the suppression pool during power operation		2000
b.	Evaporation Rates		
1.	Evaporation during refueling from upper containment pool into the containment (Temperature of containment pool = 125°. Temperature of air in the containment = 80°. Relative humidity of containment = 60%.)		1063
2.	Evaporation during refueling from spent fuel pool. (Temperature of the SFP = 125°. Temperature and Relative Humidity of air in refueling area of auxiliary building are 80° and 50%, respectively.)		913
c.	Partition Factors	**Halogens**	**Particulates**
1.	Internal partition in the reactor vessel	2×10^{-2}	1×10^{-3}
2.	Liquid leak into the containment	2×10^{-2}	1×10^{-3}
3.	Steam leak into the containment	1	1
4.	Upper containment pool surface	1×10^{-2}	Negligible
5.	Spent fuel pool surface	1×10^{-2}	Negligible
d.	Partition coefficients from Suppression Pool to Building Air		

TABLE 12.2-16: PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVITY CONCENTRATIONS (CONTINUED)

Element	Suppression Pool Partitioning* (liquid/gas)
Iodine	0.999935/0.000065
Krypton	.46/.54
Xenon	.48/.52

* Significant volatile radioisotope species taken from Table 12.2.5, GESSAR II-251

e.	Ventilation Rates	**CFM**
1.	Exhaust rate from the containment during power operation	500*
2.	Internal recirculation rate in the containment during power operation	3,000
3.	Exhaust rate from the containment during refueling	6,000
4.	Exhaust rate from refueling area in the auxiliary building during refueling (this includes a sweep system flow rate of 14,360 cfm)	15,045
5.	Exhaust rate from the regenerant solution receiving tank pump room	175
f.	Volumes of the Regions	**Ft^3**
1.	Containment	1.4×10^6
2.	Refueling area of the auxiliary building	5.5×10^5
3.	Regenerant solution receiving tank pump room	3.78×10^3
4.	Containment annulus (homogeneous air mixing)	3.00×10^5
g.	Maximum Weekly Occupancy/Person**	**Hrs**

TABLE 12.2-16: PARAMETERS AND ASSUMPTIONS FOR CALCULATING AIRBORNE RADIOACTIVITY CONCENTRATIONS (CONTINUED)

1.	Containment during power operation	40
2.	Containment during refueling	56
3.	Refueling area in the auxiliary building during refueling	56
4.	Pump room in the radwaste building	40
h.	Miscellaneous Information	
1.	Iodine halving time in the suppression pool	10,000 hrs
2.	Mixing efficiency in the containment for recirculation during power operation	70%
3.	Charcoal adsorber efficiency for removal of iodine during recirculation	90%
4.	HEPA efficiency for removal of particles during recirculation***	90%
5.	Water to air partition factor for tritium	1.0
6.	Average radiohalogen (reactor water to steam) carry-over by weight	2%

* If needed, containment air would be exhausted at 6,000 cfm in which case there would be no recirculation.

** Indicates only maximum weekly occupancy/person in any one week. However, for most of the weeks, the occupancy for items 1 and 4 will be much less than the maximum.

***Realistically, 99 percent efficiency would be more appropriate. However, conservatively 90 percent efficiency is assumed.

TABLE 12.2-17: MAXIMUM IODINE AND TRITIUM CONCENTRATIONS IN THE SPENT FUEL POOL & UPPER CONTAINMENT POOL DURING REFUELING

Concentration µCi/cc

Isotope	Upper Containment Pool**	Upper Containment Pool Spent Fuel Pool***
I - 131	3.86×10^{-4}	6.94×10^{-6}
I - 133	1.44×10^{-3}	2.04×10^{-5}
I - 135	5.29×10^{-4}	5.11×10^{-6}
H - 3*	1.99×10^{-3}	1.59×10^{-3}

* Equilibrium activity concentrations achieved after fifteenth refueling

** Concentrations prior to mixing with the Spent Fuel Pool.

*** Concentrations after complete mixing between the Upper Containment Pool and the Spent Fuel Pool.

TABLE 12.2-18: AIRBORNE RADIOACTIVITY CONCENTRATIONS IN APPLICABLE REGIONS

Airborne Radioactivity concentration (μCi/cc)

Isotope	Containment During Power Operation*	Containment During Refueling	Auxiliary Building Refueling Area	Radwaste Building Pump Rooms
H - 3	1.57×10^{-7}	8.09×10^{-8}	2.59×10^{-8}	0
Kr - 83m	6.43×10^{-8}	0	0	0
Kr - 85m	2.44×10^{-7}	0	0	0
Kr - 85	6.51×10^{-9}	0	0	0
Kr - 87	2.67×10^{-7}	0	0	0
Kr - 88	5.62×10^{-7}	0	0	0
Kr - 89	7.19×10^{-8}	0	0	0
Xe - 131m	4.35×10^{-9}	0	0	0
Xe - 133m	6.05×10^{-8}	0	0	0
Xe - 133	2.23×10^{-6}	0	0	0
Xe - 135m	7.33×10^{-9}	0	0	0
Xe - 135	1.71×10^{-6}	0	0	0
Xe - 137	9.92×10^{-8}	0	0	0
Xe - 138	2.68×10^{-7}	0	0	0
I - 131	5.39×10^{-10}	9.36×10^{-11}	1.13×10^{-12}	9.60×10^{-10}
I - 132	7.92×10^{-10}	0	0	0
I - 133	1.65×10^{-9}	2.99×10^{-10}	3.27×10^{-12}	3.46×10^{-9}
I - 134	8.17×10^{-10}	0	0	0
I - 135	1.07×10^{-9}	8.25×10^{-11}	7.79×10^{-13}	0
Rb - 88	5.38×10^{-7}	0	0	0
Rb - 89	6.93×10^{-8}	0	0	0
Sr - 89	3.69×10^{-10}	0	0	8.64×10^{-12}
Sr - 90	3.35×10^{-14}	0	0	5.76×10^{-12}
Y - 89m	4.99×10^{-17}	0	0	0
Y - 90	3.19×10^{-15}	0	0	0
Zr -95	3.89×10^{-14}	0	0	9.60×10^{-13}
Nb - 95	2.16×10^{-17}	0	0	0
Nb - 95m	2.71×10^{-15}	0	0	0
Cs - 134	1.67×10^{-13}	0	0	8.64×10^{-11}
Cs - 135	5.74×10^{-16}	0	0	0
Cs - 136	1.09×10^{-13}	0	0	8.64×10^{-12}
Cs - 137	2.92×10^{-12}	0	0	1.73×10^{-10}
Cs - 138	2.49×10^{-7}	0	0	0
Ba - 137m	2.73×10^{-12}	0	0	0
Ba - 140	2.18×10^{-12}	0	0	1.92×10^{-12}
La - 140	3.16×10^{-13}	0	0	0
Ce - 141	1.66×10^{-13}	0	0	4.99×10^{-11}
Cr - 51	2.77×10^{-11}	0	0	1.73×10^{-10}

TABLE 12.2-18: AIRBORNE RADIOACTIVITY CONCENTRATIONS IN APPLICABLE REGIONS (CONTINUED)

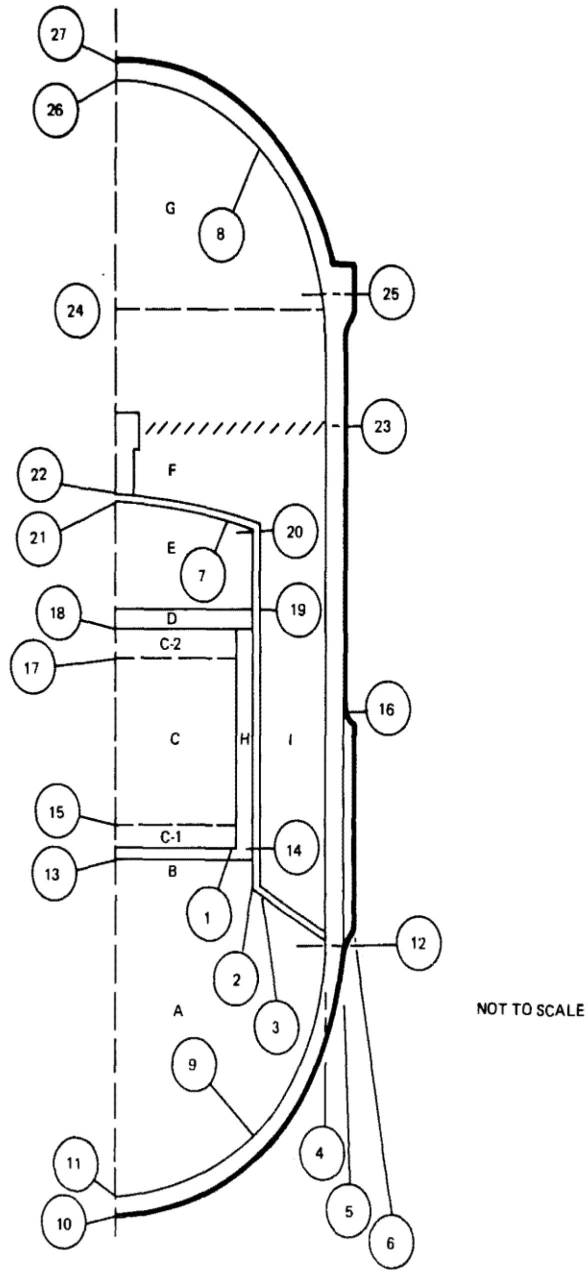
Airborne Radioactivity concentration (μCi/cc)

Mn - 54	3.35×10^{-13}	0	0	5.76×10^{-10}
Fe - 59	1.66×10^{-13}	0	0	2.88×10^{-10}
Co - 58	1.11×10^{-12}	0	0	8.64×10^{-11}
Co - 60	2.23×10^{-12}	0	0	1.73×10^{-9}
Zn - 65	1.12×10^{-12}	0	0	2.88×10^{-11}
Sb - 124	0	0	0	9.60×10^{-13}

*Based on 500 cfm exhaust rate during normal power operation. If needed, this can be increased to 6,000 cfm.

TABLE 12.2-19: ISOTOPES ABOVE 100 mCi FOR LCSB

Isotope	Activity (mCi)
H-3	9.42E+07
Fe-55	4.85E+05
Cr-51	2.30E+05
I-131	2.34E+03
Co-60	1.71E+05
Zn-65	3.98E+03
Cs-137	1.38E+03
Co-58	7.87E+03
Sr-89	4.17E+02
Mn-54	2.51E+04
Cs-134	5.24E+02
Y-91	1.87E+02
Ba-140	1.46E+02
La-140	1.46E+02
Fe-59	1.23E+04
Te-129m	1.10E+02
Sr-90	2.25E+02
Zr-95	6.04E+02
Ce-144	3.63E+02
Ni-63	1.63E+04



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	RADIATION SOURCE MODEL FIGURE 12.2-1

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

In this section, specific design features to maintain personnel exposures ALARA are discussed.

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

The equipment and plant design features employed to maintain radiation exposures as low as is reasonably achievable are based upon the design considerations of subsection 12.1.2 and are outlined in this subsection for several general classes of equipment (subsection 12.3.1.1.1), several typical layout situations (subsection 12.3.1.1.2), and the reactor coolant system (subsection 12.3.1.1.3).

12.3.1.1.1 Common Equipment and Component Designs for as Low as is Reasonably Achievable

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

a. Filters

   Filters which accumulate radioactivity carried in liquids are supplied with the means to remotely backflush the filter. The equipment and floor drain filters are centrifugally cleared of sludge by remote means.

b. Demineralizers

   Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks prior to solidification and that fresh resin can be remotely loaded into the demineralizer. Underdrains and downstream strainers are designed for full system pressure drop. The demineralizers and piping are designed with provisions to flush with condensate. Combination vent overflow lines prevent entry of spent resin into the exhaust duct.

c. Evaporators

Evaporators are provided with chemical addition connections to allow the use of chemicals for descaling operations. Space is provided to allow uncomplicated removal of heating tube bundles. The more radioactive components are separated from those that are less radioactive by a shield wall. Wherever practicable, instruments and controls are located on the accessible side of the shield wall. Wherever practicable, valves in radioactive lines are located on the accessible side of the shield wall. Valves in nonradioactive lines are located outside of the room.

d. Pumps

Where practicable, pumps are purchased with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate spaces for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal, if necessary. All pumps in radioactive waste systems are provided with flanged connections for ease in removal. Pump casings are provided with drain connections for draining the pump for maintenance.

e. Tanks

Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the floor and equipment drain system in order to control any contamination within plant structures, provisions such as curbing or dikes around the tanks are made to contain overflows and accidental spills.

f. Heat Exchangers

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

g. Instruments

Instrument devices are located in low radiation zones and away from radiation sources whenever practical. Primary instrument devices which, for functional reasons, are located in high radiation zones are designed for easy removal to a lower radiation zone for calibration if possible. Transmitters and readout devices are normally located in low radiation zones for servicing, such as corridors and the control room. Some instruments (such as thermocouples) in high radiation zones are provided in duplicate to reduce the access and service time required.

Instrument sensing lines on process piping which may contain highly radioactive liquids and solids are provided with remote back flushing capability to reduce the servicing time required to keep the lines free of solids. Instrument and sensing line connections are located in such a way as to avoid corrosion product and radioactive gas buildup.

h. Valves

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

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

When equipment in Zone E is operated infrequently, only those manual valves associated with safe operation, shutdown, and draining of the equipment are provided with remote-manual operators or reach rods. All other valve operations are performed with equipment in the shutdown mode. Simple reach rods are used to allow operators to retain the feel of whether the valves are tightly closed or not.

For valves located in radiation areas, provisions are made to drain adjacent radioactive components when maintenance is required.

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

All manually operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are remotely operated. Personnel do not enter the valve gallery during flushing operations. The valve gallery shield walls are designed for maximum expected filter backflush activities during flushing operations.

For most larger valves (2 1/2 inches and larger) in lines carrying radioactive fluids, a double set of packing with lantern ring is provided. A stuffing box is also provided with a leak-off connection which may be piped to a drain header. Full ported valves are used in systems expected to contain radioactive solids.

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

i. Piping

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

j. Floor Drains

Floor drains and properly sloped floors are provided in each room or cubicle containing serviceable components containing radioactive liquids. When practicable, shielded pipe chases are used for both major and field-run radioactive pipes. If a radioactive drain line must pass through a zone lower than that at which it will terminate, proper shielding is provided. Local gas traps or porous seals are not used on radwaste floor drains. Gas traps are provided at the common sump or tank.

k. Lighting

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

l. HVAC

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

m. Sample Stations

Sample stations for routine sampling of process fluids are located in accessible areas. Shielding is provided at the local sample stations as required to maintain radiation sampling. The counting room and laboratory facilities are described in Section 12.5.

n. Clean Services

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

12.3.1.1.2 Common Facility and Layout Designs for as Low as is Reasonably Achievable

This subsection describes the design features utilized for Grand Gulf processes and layout situations. These features are employed in conjunction with the general equipment designs described in subsection 12.3.1.1.1 and include the features discussed in the following paragraphs.

a. Valve Galleries

Valve galleries are provided with shielded entrances for personnel protection. The galleries are divided into subcompartments which service only two or three components so that personnel are only exposed to a few valves and some piping associated with these few components at any given location. Threshold berms and floor drains are provided to control radioactive leakage. To facilitate decontamination in valve galleries, concrete floors are covered with a smooth surface coating which allows easy decontamination.

b. Piping

Pipes carrying radioactive materials are routed through controlled access areas properly zoned for that level of activity. Each piping run is individually analyzed to determine the potential radioactivity level and surface dose rate. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipeways are provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Whenever practicable, equipment compartments are used as pipeways only for those pipes associated with equipment in that compartment.

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

Potentially radioactive piping is located in appropriately zoned and restricted areas. Process piping is monitored to ensure that access is controlled to limit exposure (see Section 12.5).

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Thermal expansion loops are raised rather than dropped, where possible. In radioactive systems, the use of nonremovable backing rings in the piping joints is prohibited to eliminate a potential crud trap for radioactive materials. Piping carrying resin

slurries or evaporator bottoms is run vertically whenever possible and large radius bends are utilized instead of elbows.

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

c. Penetrations

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

Penetrations through the drywell wall are tabulated on Figure 12.3-28. Those penetrations requiring shielding to maintain the radiation levels specified on the zone maps are so noted. Shielding design details for the penetrations are shown on Figures 3.8-61, 12.3-28, and 12.3-29.

Shielding for the drywell personnel lock, shown on Figure 12.3-30, is provided by the design of the personnel lock. The inner door and its bulkhead are 3-3/4-inch-thick steel. The outer door and its bulkhead are 1-inch-thick steel. Shielding of the drywell equipment hatch is shown on Figure 3.8-61.

d. Contamination Control

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

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

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

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

Figures 12.3-12 through 12.3-17 and 12.3-21 show controlled access areas, personnel and equipment decontamination areas, contamination control areas, location of the radiation protection facilities, location of area radiation monitors, location of control panels for radwaste equipment and components, location of the onsite laboratory, and location of the counting room.

Figures 12.3-22 through 12.3-27 provide radiation zones, both units in operation cross sections, and radiation zone tabulation.

e. Equipment Layout

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

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

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

Labyrinth entrance way shields or shielding doors are provided for each compartment from which radiation could stream to access areas and exceed the radiation zone dose limits for those areas. With the exception of Radwaste Building equipment/floor drain FUNDA filters, potentially high radiation components (such as filters and demineralizers), are completely enclosed in shielded compartments with hatch openings are used.

The concrete hatch covers for the Radwaste Building equipment/floor drain FUNDA filters were determined to be unnecessary during the initial operating history of these components; and were stored nearby on the filter maintenance floor area at Elevation 151'-6". Subsequently the hatch covers have been permanently removed. Applicable shielding calculations do not identify any inter-dependencies between nearby radiation levels and the installation of the hatch covers. The filter maintenance area dose rates are normally categorized as radiation zone “E” (greater than 100 mrem/hr.) for all plant conditions except when the filters are undergoing maintenance or component testing. The area is normally inaccessible. During maintenance or testing, radiation exposure levels are reduced to acceptable limits by draining and flushing the applicable components. The Radwaste Building roof (Elev. 179') above the hatches is accessible in a controlled, limited manner. As confirmed by Radiation Protection surveys, the Radwaste Building roof above the hatches remains radiation zone “B” (no more than 2.5 mrem/hr.) with the hatches removed and the filters in operation.

Figures 12.3-1 through 12.3-11 provide typical layout arrangements for demineralizers, spent resin storage tanks, hydrogen recombiners, sample racks, and their associated valve compartments or galleries.

Exposure from routine inplant inspection is controlled by locating, whenever possible, inspection points in properly shielded low background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for

routine inspection are required, and permanent shielding is not feasible, sufficient space for portable shielding is provided. In high radiation areas where routine inspection is required, remote viewing devices may be provided as needed. When this is not practicable, written procedures which reduce radiation exposure by reducing the total time exposed to the radiation field are used, and access to high radiation areas is under the direction and control of the unit's radiation protection personnel.

f. Field Run Piping

To minimize radiation exposure to plant personnel from field run piping that contains radioactive material, the following steps are taken:

1. Only piping 2 inches and smaller is permitted to be field routed.

2. Routing procedures are listed by the design and shielding engineers for use in the field. These procedures include:

   (a) The field engineer is furnished with the radiation zoning and minimum shielding requirements for the areas of his concern.

   (b) Field engineers are provided with sets of piping classification tables that give the expected radiation level category for pipes in radioactive systems (i.e., high, moderate, or low radioactivity level). Piping and instrumentation diagrams have the radiation zone marked on each field run pipe.

   (c) Field engineers route radioactive pipes through the highest zoned (lowest access) areas feasible, even if this requires a greater length of pipe. Radioactive pipes are routed through radioactive pipe chases as much as practicable. Space is allowed for local shielding of all radioactive pipes in areas that may require access while the pipe is “on line” and active.

3. Field-routed radioactive process piping is reviewed by design and shielding engineers prior to permanent installation.

12.3.1.1.3 Reactor Coolant System Design for as Low as is Reasonably Achievable (ALARA)

This subsection describes the features incorporated in the plant design to maintain occupational radiation exposure ALARA by minimizing and controlling the buildup, transport, and deposition of activated corrosion products in reactor coolant and auxiliary systems:

a. The majority of the materials for the piping and fittings in contact with the reactor coolant are carbon steel SA-106 (Grade B or C) and SA-105, respectively. These materials contain only residual amounts of nickel or cobalt. The remaining piping and fittings, which are stainless steel, contain 8 to 10.5 percent nickel. Stainless steel is used in instrument piping to prevent (or minimize) corrosion products from entering the instrument from the piping. Stainless steel is also used for the reactor recirculation system. Nickel content of these materials is low, and it is controlled in accordance with applicable ASME material specifications.

   Most of the reactor internal components are stainless steel. A small amount of nickel base material (Inconel 600) is employed in the reactor vessel internal components. Inconel 600 is required where components are attached to the reactor vessel shell, and the coefficient of expansion must match the thermal expansion characteristics of the low alloy vessel steel. Inconel 600 was selected because it provides the proper thermal expansion characteristics, adequate corrosion resistance, and can be readily fabricated and welded. Alternate low nickel materials which meet the above requirements and are suitable for long term reactor service are not available.

b. The use of carbon steel reduces the cobalt as well as the nickel content in systems in contact with reactor coolant as indicated in a. above.

c. Stellite seats containing approximately 55 percent cobalt have been used for valves. Industry and Government studies have shown that stellite has superior wear-resistant

qualities when compared to alloys having a lower cobalt content. The stellite seats will deposit less material into the reactor coolant system than would the alternate materials.

d. High temperature filtration is not used in the BWR. A full-flow condensate demineralizer has been used for Grand Gulf. The reactor water cleanup system flow rate is adequate to maintain low impurity levels within the reactor vessel.

e. The following criteria were applied for the design of the valves and packing:

   1. Valves:

      (a) Valves having a minimal amount of internal crevices were used.

      (b) Full ported valves were used where possible.

   2. Packing:

      (a) Braided packing without any loose filler material was used.

      (b) The packing is chloride free with minimal halogens to prevent stem pitting.

f. The recirculation system is equipped with decontamination flanges for decontamination of the recirculation pump and associated hardware. The cleanup system has chemical cleaning and decontamination connections to enable separate decontamination of system hardware. The ECC systems have flushing connections to permit decontamination of the system piping and components.

g. Two cleanup systems are used for removal of impurities or crud from the reactor coolant during operation. The condensate demineralizer system is used to clean up the coolant before it is returned to the reactor vessel. The reactor water cleanup system is used to remove impurities which may have been concentrated in the reactor vessel during the production of steam.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yards is regulated and controlled by radiation zoning and access control (see subsection 12.5.2). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding.

All plant areas are categorized into radiation zones according to the expected maximum total (neutron plus gamma) radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures as low as is reasonably achievable and within the standards of 10 CFR 20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal operation, shutdown, and emergency operation; for radiation streaming that might occur through penetrations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Radiation zone categories employed and their descriptions are given in Table 12.3-1 and the specific zoning for each plant area is shown in Figures 12.3-12 through 12.3-17. All frequently accessed areas, i.e. corridors, are shielded for Zone A or B access under normal conditions.

The control of ingress or egress of plant operating personnel to controlled access areas and procedures employed to ensure that radiation levels and allowable working time are within the limits prescribed by 10 CFR 20, are described in Section 12.5.

Any area having a radiation level which could cause a whole-body exposure in any one hour in excess of 5.0 mrem is posted with signs bearing the radiation symbol and words of warning. Access alert symbols (e.g., signs, chain, rope, door, etc.) are provided for radiation areas. Locations of these barriers are shown on Figures 12.3-12 through 12.3-17. Any area having a radiation level which could cause a whole-body exposure in any one hour in excess of 100 mrem is posted with the radiation symbol and the words, CAUTION, HIGH RADIATION AREA. LOCKED HIGH RADIATION AREAS and Very high radiation areas are kept locked, except during periods when access to the area is required, in which case positive control is exercised over each individual entry.

The radiological hazards at GGNS are communicated to workers in accordance with plant procedures and applicable regulations.

12.3.2 Shielding

12.3.2.1 Design Objectives

The objective of the shielding in the Grand Gulf Nuclear Station is to protect operating personnel and the general public from radiation emanating from the reactor, power-conversion, process, and auxiliary systems, including equipment and piping. Shielding requirements in the plant are determined so as to perform the following functions:

a. Ensure that exposure to radiation of plant personnel, contractors and visitors is ALARA and within 10 CFR 20 limits.

b. Limit exposure to radiation of plant personnel in the control room to within the limits of 10 CFR 50, Appendix A, Criterion 19, to ensure that the plant can be maintained in a safe condition under accident conditions.

c. Limit exposure to radiation of certain components with specified radiation tolerances if they are in high-radiation areas.

d. Limit exposure to persons at the site boundary to a small fraction of 10 CFR 20 as a result of direct radiation during normal operation.

12.3.2.2 Design Description

12.3.2.2.1 General Design Guides

In order to meet the design objectives, the following general design guides are used in the shielding analysis of the plant.

a. All systems containing radioactivity are identified and shielded, based on the access requirements of the area.

   The access zones for each area are determined and the amounts of shielding needed to ensure these zones are determined. Effort is made to locate processing systems in such a manner as to minimize shielding. Use of labyrinths is made in order to eliminate any streaming radiation from equipment.

b. Penetrations are placed so that they do not pass through the shield wall in a direct line with the radiation source in order to prevent streaming. If this is not feasible, adequate shielding is provided.

c. Wherever possible, radioactive piping runs in such a manner as to minimize radiation exposure to plant personnel. This involves (a) minimizing radioactive pipe routing in corridors, (b) avoiding the running of high-activity pipes through low radiation zones, (c) use of shielded pipe trenches where routing of high-activity pipes in low level areas cannot be avoided, and (d) separating radioactive and nonradioactive pipes for maintenance purposes.

   Prior to routing field run piping, construction engineering is given a complete set of piping diagrams that have been marked up with the expected radiation levels. They are also supplied with a complete set of radiation zone drawings. Guidance is given as to the best methods for routing and shielding the field run piping. Field run lines requiring special consideration (highly radioactive lines, seismic Category I lines, etc.) are drawn diagrammatically by Bechtel engineering. Other field run radioactive lines are reviewed by the nuclear engineering staff of Bechtel engineering to determine the adequacy of the field run proposals. Revisions are made as necessary to these proposals before any such piping can be installed.

d. To maintain acceptable levels at valve stations, motor-operated or full ported valves are used where practical. For valve maintenance, provision is made for drainage of associated equipment so that radiation exposure is minimized. If manual valves are used, provision is made to shield the operator from the valve by use of shield walls and valve step extensions.

e. The dose at the site boundary as a result of skyshine from the turbine is determined.

f. The principal shield material is concrete of density 140 lb/ft^3. Where necessary, water, steel, high-density concrete, or lead is occasionally used. Sandbags and sand-filled tanks are not used as shielding inside the containment.

g. Provision is made to shield major sources during inservice inspection to allow sufficient access.

12.3.2.2.2 Plant Shielding Description

Radiation zones are shown on Figures 12.3-12 through 12.3-17. Figure 6.4-6 shows a shielding isometric of the control room area.

The general description of plant shielding in the different plant buildings is as follows:

a. Containment

Shielding in the containment includes the reactor vessel shield, drywell, and the containment walls.

The reactor vessel shield surrounds the reactor and serves the function of reducing gamma heating in the drywell concrete wall; reducing activation of, and radiation effects on, materials and equipment in the drywell; and provides limited access in the drywell for shutdown period inspection and maintenance. The drywell region is Zone E during normal operation.

The drywell wall provides additional shielding in order to permit limited access in the containment during normal operation. The open containment area is a Zone B area.

Within the containment, there are several shielded rooms. These rooms enclose the reactor water cleanup system equipment and piping, traversing incore probe system, and safeguard and process piping systems and ensure that radiation levels in the containment are low enough for Zone B access. In addition, the main steam lines are within the shielded guard pipes.

The containment wall is a reinforced concrete structure that completely surrounds the nuclear steam supply system. This wall attenuates the system radiation to ensure that levels outside the building are less than 0.5 mrem/hr. In addition, in the unlikely event of an accident, the containment shields personnel and the public from radiation sources inside the containment. The containment wall is three and one-half feet thick.

Details on the construction of these walls are presented in Section 3.8.

A portable, trough-shaped radiation shield will be utilized during spent fuel transfer. The shield, when in place, will span the distance between the reactor vessel flange and the fuel pool gate. The shield assembly has 6 inches of lead on the bottom, 4 inches of steel on the sides to a height of 4.5 feet and 1/2-inch of steel to a height of 8 feet. The dose rate during normal fuel transfer to the head of a 6-foot person standing at elevation 161'-10" is 11.2 mrem/hr. The dose rate to a person in the same location during an accident condition with a fuel assembly lying flat on the portable shield is 249 mrem/hr.

b. Turbine Building

The anticipated major radiation source in the turbine building is the primary steam containing activation gases, principally nitrogen-16, and possibly fission products. Radiation shielding is provided around the following equipment in order to ensure that required access zone areas are met around the shielded areas:

1. Main steam lines

2. Primary and extraction steam piping

3. High and low pressure turbines

4. Moisture separators and reheaters

5. Feedwater system heaters and heater drains

6. Main condenser and hotwell

7. Air ejectors, steam packing exhauster

8. Condensate demineralizer

9. Condensate demineralizer backwash system

10. Turbine-driven reactor feed pumps

11. Offgas equipment and piping

Areas within these shields are high radiation levels and have limited access.

c. Auxiliary and Radwaste Buildings

Concrete walls, covers, removable blocks, labyrinths, and pipe trenches are used to shield the safety features equipment in the auxiliary building and processing equipment in the radwaste building, including valves and piping, in accordance with general guides of subsection 12.3.2.2.1. The access zones are determined as necessary and required shielding is provided.

Shielding also is required in the fuel-handling area in the auxiliary building. The spent fuel pool contains highly radioactive spent fuel assemblies, control rods, and instrument strings. Concrete, 4-1/2 feet thick, is used for radiation protection at the sides and bottom of the storage pool. A minimum cover of 7 feet 9 inches of water above the active fuel in each assembly will be maintained for the shielding of plant personnel during fuel transfer operations.

Access to the fuel transfer tube area would be through an opening that is 2'-0" wide by 3'-4" high. This opening is blocked with removable solid masonry blocks to a thickness of 4'-0" to maintain the radiation levels in the adjacent occupied areas at Zone B or lower. The removable blocks also allow access to the tube area. Administrative procedures will be employed to control this access by providing instructions to ensure that the fuel transfer tube room is unoccupied, locked, and posted. Access control and shielding removal are controlled by station administrative procedures. Access to the transfer tube area is prohibited during fuel movement through the transfer tube. When required for maintenance and

inspection activities, access to this area is controlled by a Radiation Work Permit and fuel movement through the tube is stopped. All appropriate areas shall be marked in accordance with 10 CFR 20 Subparts F, G, and J to indicate the radiation hazard. The gap between the containment wall and the structures inside the containment is shielded with a lead collar, and the gap between the containment wall and the auxiliary building is shielded with solid masonry blocks. Refer to Figure 12.3-20 for the structural barriers and shielding in the area of the fuel transfer tube.

d. Control Room

The control room shielding design is based on the requirements of 10 CFR 50, Appendix A, Criterion 19, which requires that personnel can occupy and have access to the control room following a maximum hypothetical accident, maintain full control, and shut down the plant. The accident analysis in Chapter 15 indicates that the dose to personnel would be less than 5 rem for the duration of the accident.

Direct shielding of the control room from the fission product inventory in the containment is provided by the concrete walls between them. Emergency air conditioning and filtration systems are provided for post-LOCA conditions in the control room as described in Section 9.4. Figure 6.4-6 shows an isometric view of the control room shielding.

The control room is located in an area well removed from potential accident sources. There are many intervening walls and floors between the control room and the sources or potential sources of radiation. As noted in subsection 12.3.1.1.2.c, all penetrations through shield walls were investigated and shielded as necessary to prevent radiation streaming. Therefore, during postulated accidents, there will be no radiation streaming through penetrations into the control room.

e. General Plant Yard Areas

Plant yard areas which are frequently occupied by plant personnel receive a radiation field of less than 0.5 mr/hr. These areas are surrounded by a security fence and closed off from areas accessible to the general public for reasons of general safety.

12.3.2.3 Method of Shielding Design

Shield wall thicknesses are determined by using basic shielding data and equations. The methods are taken from the Reactor Shielding Design Manual (Ref. 9) by T. Rockwell and The Engineering Compendium of Radiation Shielding (Ref. 1). Data are taken from the Table of Isotopes, Reactor Physics Constants, ANL-5800, XDC-59-8-179, and other pertinent texts (Refs. 2 and 7).

All shielding is done using the design basis of a noble gas fission product release rate 100,000 μCi/sec after 30 minutes decay, and of the corresponding activation and corrosion product reactor water concentrations found in Tables 11.1-1 through 11.1-5. Radiation sources for various pieces of equipment in the plant are determined as indicated in subsection 12.2.1.

The geometry used for shielding evaluation is that of a finite shielded cylinder for tanks, demineralizers, filters, heat exchangers, etc., and an infinite shield cylinder in the case of piping carrying radioactive fluids. Special cases require different geometric assumptions as determined in individual cases.

Calculations are done to determine the gamma dose rate through a laminated shield at the sides and ends of a cylinderized source to obtain exposure values. Source strengths are divided into energy bins for different gamma energies. Credit is taken for self-attenuation, except for gaseous sources. Buildup is determined using Broder's laminated shield formula. Shielding materials are concrete and steel and, if appropriate, water.

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall. Therefore, the actual anticipated radiation levels in the greater region of each plant area are less than this maximum dose and therefore less than the radiation zone upper limit.

Where shielded entry ways to compartments containing high radiation sources are necessary, labyrinths or mazes are designed using a general purpose gamma-ray scattering code. The mazes are so constructed such that the scattered dose rate plus the transmitted dose rate through the shield wall from all contributing sources are below the upper limit of the radiation zone specified for each plant area.

A list of computer codes used is provided in Table 12.3-2.

12.3.3 Ventilation

12.3.3.1 Design Objective

The plant ventilation systems for normal plant operation and anticipated operational occurrences are designed to meet the requirements of 10 CFR 20, Standards for Protection Against Radiation, 10 CFR 50, Licensing of Production and Utilization Facilities, and Regulatory Guide 8.8.

12.3.3.2 Design Criteria

The plant ventilation systems, in addition to their primary function of preventing extreme thermal environmental conditions for operating personnel and equipment, provide effective protection for operating personnel against possible airborne radioactive contamination in areas where this may occur.

The systems operate to ensure that the maximum airborne radio activity levels for normal operation, including anticipated operational occurrences, are within the limits of 10 CFR 20, Appendix B, Table I, for areas within plant structures and on the plant site where construction workers and visitors are permitted. The maximum levels correspond to design-bases reactor coolant inventory. The average airborne radioactivity levels are considerably smaller since average coolant inventories and actual equipment leakages are small.

The control room ventilation system also operates to provide a suitable environment for equipment and continuous personnel occupancy in the control room under post-accident conditions in accordance with 10 CFR 50, Appendix A, Criterion 19.

12.3.3.3 Design Guidelines

In order to accomplish the design objectives, certain general design guidelines are followed where practicable.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

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

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

c. All welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the radwaste collection system.

d. The valves in some systems are provided with leak-off connections piped directly to the liquid radwaste collection system.

e. To minimize the amount of airborne radioactivity as a result of valve leakage, most larger valves (2-1/2 inches and larger) are provided with a double set of packing with a lantern ring in lines carrying radioactive fluids. A stuffing box is also provided with a leak-off connection that is piped to a drain header. Metal diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated.

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

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

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

b. A greater volumetric flow is exhausted from the area than is supplied to the area, where applicable, to prevent uncontrolled exfiltration of contamination. Slightly positive pressure is maintained in the control room during normal operation to prevent infiltration of potential contaminants.

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

d. HEPA filters are provided on exhausts from the radwaste and containment buildings (including drywell), to remove airborne activity and to reduce onsite and offsite radiation levels. The containment (including drywell), and portions of the radwaste building exhausts are also provided with charcoal filters. These filter units generally comply with the access and service requirements of Regulatory Guide 1.52.

e. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered through HEPA filters to prevent contamination of the control room.

f. Suitable containment isolation valves are installed in accordance with General Design Criteria 54 and 56, including valve controls, to assure that the containment integrity is maintained. See additional discussion in Sections 3.1 and 6.2.4.

g. Redundant, seismic Category I systems and/or components are provided for portions of the control room, SGTS and ESF rooms ventilation systems. The extent to which redundant components are provided is discussed in Sections 9.4 and 6.5.

h. Atmospheric tanks which contain radioactive materials are vented to the respective building ventilation systems through charcoal filters in the Turbine Building.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment

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

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

c. Ventilating air is recirculated in clean areas only.

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

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

   2. Local cooling equipment, servicing the normal building requirements, is generally located in areas of low contamination potential, radiation Zones A, B or C. The drywell and steam tunnel coolers are located in Zone E. However, fully redundant components are provided, allowing servicing to be deferred and performed during shutdown periods. Several turbine building vents are located in Zone E, however these are expected to require a minimum of maintenance which can be scheduled during shutdown.

12.3.3.4 <u>Design Description</u>

The previously discussed guidelines have been incorporated in the heating and ventilation design described in Section 9.4. The following is a brief summary of those systems.

12.3.3.4.1 Control Room Ventilation

During normal plant operation, control room air is recirculated through an air-conditioning unit to maintain control room design conditions. Fresh air makeup is provided from a single intake located on the roof of the control building at El. 206. An airborne radiation monitoring system and a control room filter recirculation system have been provided to detect and reduce airborne radiation levels in the control room in the event of a LOCA. Redundant airborne radiation detectors in the fresh air intake duct monitor the fresh air supply. A high airborne radiation signal alarm in the control room automatically closes the fresh air intake damper, shuts down and isolates the utility exhaust fan, and starts the emergency filtration unit trains. An area radiation monitor is provided in the control room to detect high radiation levels in the control room area. After a prolonged isolation of the control room and when conditions permit, fresh air can be manually brought in through the HEPA filter system.

There are two 100 percent capacity HEPA filter trains which operate following control room isolation. A complete description of control room ventilation is found in subsection 9.4.1.

12.3.3.4.2 Drywell

The drywell cooling system consists of recirculation, fan-coil units to maintain the design drywell temperature and relative humidity. Six fan-coil units are provided to distribute cooling air effectively with minimum ductwork. Cooling water for the drywell cooling system is provided by the drywell chilled water system.

The drywell purge system is designed to purge the drywell at a minimum rate of 1-1/3 air changes per hour. The drywell filtration purge system operates by closing the dampers to the containment cooling and distribution system and opening the dampers to the drywell. The ventilation filtration system can be used for drywell purge, recirculation cleanup, or a ratio of purging and recirculation in the same manner as described in subsections 9.4.7.2.2 and 9.4.8.2.2 for the containment and drywell ventilation and filtration systems. The discharge is to the containment vent. The activity of the air is continuously monitored.

A complete description of drywell ventilation is found in subsection 9.4.8.

12.3.3.4.3 Containment

The containment cooling system consists of recirculating air-conditioning units which maintain the design containment temperature and relative humidity.

During normal system operation, the containment coolers recirculate the air through chilled water cooling coils. One of the charcoal filter trains draws 3,000 cfm out of the main containment cooler discharge plenum, filters the air through HEPA filters and deep-bed charcoal filters, and returns the air to the containment coolers return plenum. The exhaust air is discharged to the containment vent and then to the environment if the filters are utilized in an exhaust mode.

A complete description of the containment ventilation system is found in subsection 9.4.7.

12.3.3.4.4 Auxiliary Building

The auxiliary building is divided into 6 zones for air-conditioning purposes.

Zones 1 through 4 are the lower four floors, which are maintained at design temperature and relative humidity by recirculating air through air-conditioning units. Outdoor air is used and mixed with the return air for ventilation control.

The fifth zone is the fuel handling areas which is maintained at design temperature and relative humidity by recirculating air through air-conditioning units. Evaporation from the fuel handling pool is controlled by return ducting on one side of the pool and a sweep system on the other side. The fuel handling area is ventilated such that a greater volumetric flow is exhausted from the area than is supplied to the area. Exhaust air from the fuel handling area is discharged to the auxiliary building vent stack during normal operation. The activity of the exhaust air is continuously monitored.

The sixth zone is the steam tunnel which is maintained at design temperature by recirculating air through coolers containing redundant fans and coils.

A complete description of auxiliary building ventilation is found in subsection 9.4.6 and the fuel handling area ventilation is described in subsection 9.4.2.

12.3.3.4.5 Turbine Building

Outside air is induced into the turbine building above the operating floor and then drawn down to the areas below the floor by the exhaust system. A greater volumetric flow is exhausted from the turbine building than is supplied to it to assure that no outleakage of air will occur.

Air flow control is from areas of low potential radioactivity to areas of high potential radioactivity. In this way, clean area passageways are kept free of radioactive contaminants.

Space fan-coil terminal units are located throughout the turbine building to provide the cooling capacity required to maintain design temperatures.

The turbine building exhaust air system exhausts air from the condenser area, turbine building equipment compartments, and the turbine building equipment drain sumps. Air is drawn out and exhausted to the vent.

A complete description of turbine building ventilation system is found in subsection 9.4.4.

12.3.3.4.6 Radwaste Building

Outside air is filtered, tempered, and delivered to the clean areas such as the lower level corridors. A pressure gradient is maintained to create air flow from the corridors into the equipment cells, where it is exhausted after removing airborne contaminants. A greater volumetric flow is exhausted from the radwaste building than is supplied to it.

Tanks with potentially high levels of radioactivity, such as the equipment drain collector tank, are vented to a charcoal filter.

A complete description of the radwaste building ventilation system is found in subsection 9.4.3.

12.3.3.4.7 Hot Machine Shop/Decontamination Facility

Fresh air is supplied to the hot machine shop via the access control area fan coil unit. A greater volumetric flow is exhausted from the hot machine shop and decontamination facility than is supplied to it to assure that no outleakage of air into the control building will occur. Air from the hot machine shop and decontamination facility is exhausted through the turbine building exhaust vent.

Air flow is from areas of low potential radioactivity to areas of high potential radioactivity. Ventilation hoods are provided where the potential for airborne radioactivity is great.

A complete description of this system is found in subsection 9.4.10.

12.3.3.5 Air Cleaning System Design

The guidance and recommendations of Regulatory Guide 1.52, concerning maintenance, in-place testing provisions for atmospheric cleanup systems and air filtration and adsorption units have been used as a reference in the design of the various safety-related charcoal filter systems. The extent to which Regulatory Guide 1.52 has been followed is discussed in Appendix 3A.

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

a. The loading of the filters with radioactive material during normal plant operation is a slow process; therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity on a scheduled maintenance basis with portable equipment, and the filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard. Filters whose radioactivity level (due to a postulated accident) is such that a change of filter elements would constitute a personnel hazard can be removed intact. No shielding is provided since it is not required for the level of radioactivity developed during normal operation. It will not be necessary for workers to handle filter units

immediately after a design basis accident so that exposure can be minimized by allowing the short-lived isotopes to decay before changing the filter.

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

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

d. Typical layout with minimum distances for access and servicing is shown on Figures 12.3-18 and 12.3-19. No filter bank is more than three filter units high; each filter unit is 2 ft by 2 ft. The access to the level or platform at which the filter is serviced is by stairs or ladders.

e. The clear space for doors throughout the plant is a minimum of 2 ft-6 in. by 7 ft.

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

12.3.4 Area Radiation And Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring

The area radiation monitoring system is provided to ensure compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, 10 CFR 72 and Regulatory Guides 8.2 and 8.8.

Grand Gulf Nuclear Station has adopted the provisions of 10CFR50.68 in lieu of the monitoring system capable of detecting a criticality as described in 10CFR70.24. Compliance with the requirements of 10CFR50.68 negates prior NRC exemptions to 10CFR70.24. 10CFR50.68 stipulates that licensees comply with either 10CFR70.24 or 10CFR50.68. In order to comply with

10CFR50.68, GGNS will limit the total quantity of Special Nuclear Material (SNM) present in the not-in-use in-core nuclear instrumentation to less than a critical mass as defined by Section 1.1 of NRC Regulatory Guide 10.3 (Revision 1, dated April 1977).

Consistent with this purpose, the area radiation monitors function to:

a. Immediately alert plant personnel entering or working in nonradiation or low radiation areas of increasing or abnormally high radiation levels which, if unnoticed, could possible result in inadvertent over exposures.

b. Inform the control room operator of the occurrence and approximate location of an abnormal radiation increase in nonradiation or low radiation areas.

c. Comply with the requirements of 10 CFR 50, Appendix A, General Design Criterion 63, for monitoring fuel and waste storage and handling areas.

d. Certain monitors located near the spent fuel pool act as criticality alarm monitors and conform to the requirements of 10 CFR 72 (Specifically 10CFR72.124).

e. To detect excessive radiation levels and to initiate appropriate safety actions during receipt and handling of unirradiated fuel as described in Section 9.1.4. Monitoring complies with the requirements of 10CFR50.68.

f. In general, assist in maintaining personnel exposures as low as reasonably achievable (ALARA).

The area radiation monitoring system has no function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment.

12.3.4.1.1 Criteria for Area Monitor Selection

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

RANGEABILITY - Five decades of range with the alarm set point preferably not lower than the second decade and set at the maximum dose rate for the area being monitored. The lower range limit is

either natural background or one decade below the normal operating level of each particular area. The system continues to read upscale if exposed to radiation levels above maximum range.

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

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

ENERGY DEPENDENCE - The computerized dose rate (mrem/hr) readout is within 15 percent of the actual dose rate on each instrument range area from photon energies between 100 keV and 1.5 MeV.

ENVIRONMENTAL DEPENDENCE - The system meets the above requirements for all variations of temperature, pressure, and relative humidity within each area monitored which includes 95 percent relative humidity and temperatures between 32 F and 120 F.

EXPOSURE LIFE - Each detector maintains its characteristics up to an integrated dose of 10^5 rads.

12.3.4.1.2 Criteria for Location of Area Monitors

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

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

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

c. Areas in which the new fuel is received and stored. Radiation monitors are required by 10CFR50.68 when new fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

d. Area monitors are provided in accordance with General Design Criterion 63 of 10 CFR 50 Appendix A.

12.3.4.1.3 System Description (Area Radiation Monitoring)

The area radiation monitoring system detects, measures, and records ambient gamma radiation levels at various locations. It also provides audible and visual alarms in areas monitored and in the control room, if gamma radiation exceeds a specified limit. It provides visual indication in the area monitored and at a control room annunciator if there is a malfunction in any area monitor.

Each area radiation monitoring channel consists of a detector and a local alarm unit at a remote location and an indicator and trip unit in the control room. A control room channel is provided with a detector in the control room and no local alarm unit. The area radiation monitor provided in the control room has no local alarm unit since the control room annunciator system provides alarms to the operators.

The area radiation monitor channels share a multipoint recorder in the control room. Each monitor is provided with two alarms. The failure alarm is activated if high voltage, signal, or line voltage fails. Green lights on the affected indicator trip unit and local alarm unit turn off, and a control room annunciator is activated when an area monitor fails. The high radiation alarm setpoint is adjustable, and the alarm is activated when the radiation level exceeds the set point. High radiation turns on a red light on the affected indicator trip unit and causes a red lamp to flash and an audible alarm to sound on the affected local alarm unit, as well as activating a control room annunciator. In addition the indicator trip units are provided with a logarithmic meter which reads in mr/hr.

All channels have a five decade range as appropriate for their detector locations and as specified in Tables 12.3-3 and 12.3-4.

With the exception of the Geiger-Muller detector tubes, all electronics are solid-state, and the system is designed for high reliability. All monitors are independent and failure of one monitor has no effect on any other.

The area radiation monitors are powered from the uninterruptible ac bus. Emergency power to this bus is provided by the station battery through an inverter for balance of plant equipment.

The location of each area radiation detector is indicated on the radiation zoning and access control drawings, Figures 12.3-12 through 12.3-17 and are listed in Table 12.3-3. Consistent with the above criteria, the following general areas are monitored:

a. Control room

b. Radwaste building corridors

c. Auxiliary building corridors

d. Fuel storage and handling area

e. Valve operating stations

f. Containment

g. Radwaste solidification area

h. Sampling rooms

12.3.4.1.4 Safety Evaluation

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

The area radiation monitoring system is designed to operate unattended for extended periods of time detecting and measuring ambient gamma radiation. Ambient radiation dose rate at the detector is indicated remotely in the control room. These monitors cause an audible and visual alarm at the detector and in the control room if the radiation levels exceed preset limits.

12.3.4.1.5 Calibration and Testing

Each of the monitors is calibrated by the instrument manufacturer prior to shipment using sources certified by National Institute of Standards and Technology (NIST) or traceable to NBS. In-plant calibration, using a standard radioactive point source traceable to NBS, is performed as required by the TRM or whenever maintenance work is done on the detectors.

The proper functioning of each monitor is verified periodically by checking instrument response to the remotely operated radioactive check source provided with each detector. Proper operation of the monitor's electronics is verified periodically by use of internal check circuitry provided in each unit.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

Airborne radioactivity monitoring is provided in compliance with 10 CFR 20 and Regulatory Guides 8.2. and 8.8. The purpose of the airborne radioactivity monitoring system is to monitor the air within an enclosure by either direct measurement of the enclosure atmosphere or the exhaust air from this enclosure. The system indicates and records the levels of airborne radioactivity, and, if abnormal levels occur, actuates alarms. Alarms are provided to alert personnel that airborne radioactivity is at or above the selected set point level to ensure that personnel are not subjected to airborne radioactivity above the limits in 10 CFR 20. The system provides a continuous record of airborne radio activity levels which will aid operating personnel in maintaining airborne radioactivity at the lowest practicable level.

12.3.4.2.1 Criteria for Selection of Airborne Radioactivity Monitors

The criteria for determining the type of airborne radioactivity monitoring system are based upon the nature and type of radioactive releases expected, and the location being monitored.

Where ingestion of radioactive airborne materials by plant personnel is a possibility, monitors are used to analyze, record and alarm should the radioactivity approach the limits established by 10 CFR 20.

In the case of the drywell radioactivity monitor which is used to detect leakage from the reactor coolant pressure boundary, the guidance of Regulatory Guide 1.45 is followed, as discussed in Appendix 3A.

The drywell radioactivity monitoring system provides a means to detect leakage of steam or reactor water. A continuous air sampling system is provided for gross counting of particulate and noble gas radioactivity.

The drywell radioactivity monitoring system functions to inform the control room operator immediately of abnormal radioactivity levels in the drywell.

The in-plant airborne radiological monitoring and sampling systems are provided to allow determination of the content of radioactive material in various rooms throughout the plant. The design objectives and criteria are primarily determined by the system safety functions.

The main objectives of those in-plant airborne radioactivity monitoring systems which are required for safety are to initiate appropriate protective actions to limit the potential release of radioactive materials from the primary and secondary containment, and spent fuel pool area or the intake of radioactivity into the control room if predetermined radiation levels are exceeded in major HVAC streams. Additional objectives are to have those systems available under all operating conditions including accidents and to provide control room personnel with an indication of the radiation levels in the major HVAC streams plus alarm annunciation if high radiation levels are detected.

The radiation monitoring systems (RMS) provided to meet these objectives are:

a. Containment and Drywell Ventilation Exhaust RMS

b. Auxiliary Building and Fuel Handling Area Ventilation Exhaust RMS

c. Auxiliary Building Fuel Handling Area Pool Sweep Exhaust RMS

d. Control Room Ventilation RMS

The design criteria for the safety-related in-plant airborne radioactivity monitoring systems are that they shall:

a. Withstand the effect of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions

b. Perform their intended safety functions under normal and postulated accident conditions

c. Meet the reliability, testability, independence and failure mode requirements of engineered safety features

d. Provide continuous output on control room panels

e. Permit checking of the operational availability of each channel during reactor operation with provision for calibration function and instrument checks

f. Assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences

The drywell radioactivity monitoring system is designed to remain functional when subjected to the safe shutdown earthquake. It can operate under the following environmental conditions:

a.	Temperature	40 to 140 F
b.	Humidity	20 to 90% RH
c.	Pressure	-1/4 inch H_2O to atmospheric
d.	Radiation	4×10^3 Rads*

The safety-related, in-plant airborne radioactivity monitors are designed to remain functional when subjected to the safe shutdown earthquake. They can operate under the following environmental conditions:

a.	Temperature	32 - 140 F
b.	Humidity	20 - 98% RH
c.	Pressure	-1/4 inch H_2O to atmospheric
d.	Radiation	2×10^5 Rads*

*Integrated exposure

12.3.4.2.2 Criteria for Airborne Radioactivity Monitor Locations

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

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

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

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

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

e. The outside air intake duct for the control room area is monitored to measure possible introduction of radioactive materials into the control room to ensure habitability of those areas requiring personnel occupancy for safe shutdown.

The drywell radioactivity monitor samples from the drywell.

The containment and drywell ventilation exhaust monitor measures radioactivity in the containment atmosphere upstream of the filters and the exhaust ventilation isolation valves.

The auxiliary building and fuel handling area vent exhaust monitor measures radioactivity in the combined flow from both areas from the exhaust duct upstream of all filters and the exhaust ventilation isolation valves.

The auxiliary building fuel handling area pool sweep exhaust monitor measures radioactivity in the pool sweep exhaust upstream of the isolation valves.

The control room ventilation monitor measures radioactivity in the control room supply air upstream of all filters and the isolation valves.

12.3.4.2.3 System Description (Airborne Radioactivity Monitors)

12.3.4.2.3.1 Drywell Radioactivity Monitor

The drywell radioactivity monitoring system includes a pumping system to draw a continuous sample of the drywell atmosphere through the particulate monitor and gaseous monitor and return it to the drywell. The two channels of the system are as follows:

a. Particulate: The particulate monitor collects and measures airborne radioactive particulate matter by means of a continuously moving filter and a beta sensitive scintillation detector.

b. Deleted

c. Gaseous: The gaseous monitor measures radioactivity in the sample after it leaves the particulate monitor. The detector is a beta sensitive scintillation detector.

Ratemeters are provided for each channel for control room indication of drywell radioactivity levels. Alarms are provided for each channel for high and high-high radioactivity and for analyzer failure. Alarms are also provided for failure of the filter in the particulate monitor.

A recorder is provided in the control room to provide a permanent record of drywell radioactivity.

The drywell radioactivity monitoring system is powered from the ESF Division I bus. The sample line containment isolation valves are powered from redundant ESF buses. The ranges and set points for particulate and gaseous radioactivity monitoring are shown in Table 12.3-5.

12.3.4.2.3.2 Containment and Drywell Ventilation Exhaust Radiation Monitoring System

This system monitors the radiation level exterior to the containment ventilation system exhaust duct upstream of the filters and the exhaust ventilation isolation valve. A high activity level in the ductwork could be due to fission gases from a leak or an accident.

The system consists of four redundant instrument channels. Each channel consists of a local detection assembly (a sensor and converter unit containing a GM tube and electronics) and a control room radiation monitor. Power for each of the 4 channels is supplied from a separate ESF Uninterruptible Power Supply (UPS) inverter bus (1Y87, 1Y88, 1Y96 and 1Y95). Channels A and C are powered by ESF UPS Division 1 and 3 buses 1Y87 and 1Y96 respectively. Channels B and D are powered by ESF UPS Division 2 and 4 buses 1Y88 and 1Y95 respectively. See Section 8.3.1.1.4.1.4 for a description of the 120 Volt AC Class 1E Uninterruptible Power System. One recorder powered from the 125-V dc bus A allows the output of all channels to be recorded. The detection assemblies are physically located outside and adjacent to the exhaust ducting upstream of the containment discharge isolation valves.

Each radiation monitor provides both an analog output signal and a contact which opens on upscale (high-high) radiation or an inoperative circuit. Two-out-of-two upscale/inoperative trips in channels A and D initiate closure of the containment ventilation outboard isolation valves and the drywell inboard isolation valves. The same condition for channels B and C initiates closure of the containment inboard valves and drywell outboard valves.

An upscale/inoperative trip is visually displayed on the affected radiation monitor and actuates a containment and drywell ventilation exhaust high-high radiation control room annunciator. A downscale trip is also visually displayed on the radiation monitor. Containment and drywell ventilation high radiation and

downscale control room annunciators common to all channels are generated from the analog signal. Each radiation monitor visually displays the measured radiation level.

12.3.4.2.3.3 Auxiliary Building Fuel Handling Area Ventilation Exhaust Radiation Monitoring System

This system monitors the radiation level exterior to the auxiliary building fuel handling area ventilation exhaust duct upstream of the filters and the exhaust ventilation isolation valve. The system consists of four channels identical to the channels in the containment and drywell ventilation exhaust radiation monitoring system with the same arrangement and power sources, corresponding annunciators, and recorder.

Two-out-of-two upscale (high-high)/inoperative trips in channels A and D initiate closure of the inboard isolation valves of the auxiliary building and fuel handling area ventilation systems and initiate startup of standby gas treatment system (SGTS) train A. The same condition for channels B and C initiates closure of the corresponding outboard isolation valves and initiates startup of SGTS train B.

12.3.4.2.3.4 Auxiliary Building Fuel Handling Area Pool Sweep Exhaust Radiation Monitoring System

This system monitors the radiation level exterior to the pool sweep exhaust duct upstream of the isolation valve. The system is identical to the auxiliary building fuel handling area ventilation exhaust radiation monitoring system with the same channel trip logic and protective action initiation. The recorder is powered from 125 V dc bus B.

12.3.4.2.3.5 Control Room Ventilation Radiation Monitoring System

This system monitors the radiation level exterior to the inlet ducting of the control room ventilation system. The system consists of four channels identical to the channels in the containment and drywell ventilation exhaust radiation monitoring system. The recorder is powered from 125 V dc bus B.

Two-out-of-two upscale (high-high)/inoperative trips in channels A and D initiate shutdown and outboard isolation valve closure of the control room ventilation system and initiate startup of the emergency air filtration fan (unit A). The same condition for

channels B and C initiates shutdown and inboard isolation valve closure of the control room ventilation system and initiates startup of the emergency air filtration fan (unit B).

12.3.4.2.4 Regulatory Guide Compliance

Refer to Section 11.5 for details of compliance with NRC Regulatory Guide 1.21 and ANSI Std N13.1. Refer to Section 12.5 for details of compliance with Regulatory Guide 8.2.

The airborne radioactivity monitoring systems comply with the recommendations for radiation monitoring systems presented in Regulatory Guide 8.8 as follows:

a. Readout capability is provided in the control room.

b. The placement of monitors is as shown on the radiation zone drawings, Figures 12.3-12 through 12.3-17.

c. Component failure is detected by a downscale reading from the monitor. An immediate control room annunciator informs the operator of a monitor failure. In addition, differential pressure switches are provided in the particulate monitor of the drywell radioactivity monitor to indicate malfunction of the filters.

d. Panel meters, audible alarms, and alarm lights provide clear and unambiguous indication of alarm conditions.

e. Ranges are chosen in accordance with expected radioactivity levels in the areas being monitored. Positive readout is assured from the lowest anticipated level to full-scale deflection.

f. Control room recorders are provided to permit recording of the readout of all systems.

12.3.4.2.5 Safety Evaluation

The following monitors are located upstream of filters and, therefore, are effective for monitoring in-plant airborne radioactivity levels.

a. Drywell radioactivity

b. Containment and drywell ventilation exhaust

c. Auxiliary building fuel handling area ventilation exhaust

d. Auxiliary building fuel handling area pool sweep exhaust

e. Control room ventilation supply

The drywell monitoring system serves no safety function. Failure of the system will not compromise other safety-related systems or prevent safe shutdown. The containment and drywell penetrations for the drywell monitoring system sample lines are of seismic Category I design, and the containment penetrations are provided with redundant isolation valves. The system will remain functional when subjected to the safe shutdown earthquake.

The in-plant HVAC airborne radioactivity monitors have safety related functions of isolating their particular ventilation systems and actuating the associated filtered emergency systems as has been discussed in subsections 12.3.4.2.1, 12.3.4.2.2, and 12.3.4.2.3.2 through 12.3.4.2.3.5. These monitors are redundant, seismic Category I and are powered from the emergency power system.

The combination of the airborne radioactivity monitoring system in conjunction with administrative controls restricting and limiting personnel access, standard radiation protection practices, ventilation flow patterns throughout the plant, plant equipment layout, lack of sources in radiation Zone B areas, and restricted radiation Zone E areas, is sufficient to ensure that airborne radioactivity levels are safe in terms of the required duration of personnel access throughout all areas of the plant. A general review of these concepts follows:

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

b. Air flow patterns are consistent with the basic ventilation design criteria of the plant. Clean filtered outside air is supplied to Zone B areas (corridors, clean areas); these areas are exhausted by drawing air into

rooms and areas of successively higher potential for airborne contamination. Air flow is such that air flow reversal or exfiltration from potentially contaminated areas is precluded. This ventilation arrangement eliminates the possibility of personnel exposure to airborne radioactivity in continuous occupancy areas such as radiation Zone B areas. With the exception of the auxiliary building fuel handling area vent exhaust, the HVAC airborne radioactivity monitors are located in the system exhaust from areas that have no subcompartments so that there is no dilution of the possibly contaminated air prior to the air flow reaching the monitor. The auxiliary building fuel handling area ventilation exhaust airborne radioactivity monitor is located at a point downstream of where the smaller exhaust ducts join together to form a common exhaust duct (see Figures 9.4-2 and 9.4-11). The detectors are not capable of distinguishing which area of the building contains the high radioactivity; however, the exhaust process radioactivity monitors have sufficient sensitivity to detect 10 MPC-hours of airborne radioactivity exhausting from the area with the lowest exhaust flow rate, considering dilution from the other areas being exhausted.

In addition to these monitors, area radiation monitors with alarm capability, are placed at various locations throughout the plant. Also, effluent radioactivity monitors with alarm capability are provided for the containment ventilation exhaust system and the fuel handling ventilation exhaust system. These effluent monitors sample and analyze the same atmosphere as that being monitored by the process radioactivity monitors. Each effluent radioactivity monitoring system contains iodine, particulate, and noble gas detectors to monitor radioactivity levels from normal operation through post-accident.

Table 12.3-5 indicates that the radiation levels for 10 $(MPC)_a$-hours of airborne radioactivity are within the ranges of all the monitors. The initial values for the warning alarms are provided in the TRM and were chosen based on past BWR operating experience. The final values will be determined after the plant is in operation based on actual background levels to ensure exposures are ALARA.

c. High radiation areas (radiation Zone E areas) where dose levels may exceed 100 mrem/hr are kept restricted and conspicuously posted in accordance with GGNS Technical Specifications. These areas are not normally entered. Authorization must be obtained before entry. Prior to entry, a high volume portable air sampler may be used by the Radiation Protection group to collect a representative air sample. Gaseous and particulate activity of the area may be analyzed before entry.

d. Radiation Protection programs are discussed in Section 12.5.

To completely identify any area of the plant which may contain high activity, alarm response instructions will be followed upon receipt of any radiation monitor annunciator. These instructions, in general, direct the operator to check the other appropriate indications (such as high radiation alarms, sump level monitors, etc.) to aid in identifying the specific area with high airborne activity. As a follow-up procedure, the radiation protection staff would be directed to conduct portable air sampling and visual surveys to verify/identify the area with high airborne activity.

12.3.4.2.6 Sensitivities

Each monitoring system has a minimum detectable concentration within the limits established by 10 CFR 20.

12.3.4.2.7 Calibration and Testing

The installed airborne radioactivity monitoring systems are calibrated at frequencies specified in ODCM using calibration standards which are traceable to the National Institute of Standards and Technology.

A record is maintained of the results of channel checks, functional checks, calibrations and, where applicable, background levels. Following repairs or modifications, the monitors are recalibrated at the plant with the secondary radionuclide standards.

Decay curves are normally provided for the sources to permit correction for source decay. Electronic circuit check is performed through the use of an internal oscillator or pulse generating circuit.

12.3.4.3 In-Containment Area Radiation Monitoring

The ICARM system is provided to detect, indicate, and record gamma radiation levels in the containment and the drywell, over a range of 1 to 10^7 R/hr during and following a loss-of-coolant accident. The system meets the design and qualification criteria as outlined in IEEE 323-1974 and IEEE 344-1975.

The ICARM system has no function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment. The system's only purpose is to monitor high range radiation levels in the containment and drywell after an accident.

12.3.4.3.1 Criteria for Monitor Selection

The following design criteria are applicable to the ICARM system:

a. Rangeability - Seven decades of range are available on each monitor.

b. Sensitivity - The ICARM is gamma sensitive to photon energies of 60 keV to 3 MeV.

c. Response - System response time is less than 1 second.

d. Energy Dependence - The dose rate (mr/hr) readout is within +/-20% percent of the actual dose rate in each detected area from photon energies between 0.1 MeV and 3.0 MeV.

e. Environmental Dependence - The system meets the requirements for all variations of temperature, pressure, and relative humidity within each area monitored. Qualification includes 100 percent relative humidity and temperatures between 32 F and 357 F.

f. Exposure Life - Each detector maintains its characteristics up to an integrated gamma dose of 2×10^8 Rads.

12.3.4.3.2 Criteria for Location of Monitors

Two radiation detectors are mounted at north and south locations inside the containment wall and two are mounted at east and west locations inside the drywell wall. The detector locations are given on the radiation zoning and access control drawings, Figures 12.3-15 and 12.3-17.

12.3.4.3.3 System Description (ICARM)

The ICARM system continuously detects, indicates, and records high-range gamma radiation levels in the containment and the drywell during and following an accident.

Each radiation monitoring channel consists of a local ion chamber detector, a control room indicator trip unit, and a control room recorder. Each monitor is provided with radiation and circuit failure alarm indicating lights. In addition, each monitor is provided with a seven-decade logarithmic meter which reads in R/hr.

Each monitor is provided with an electronic “check source” test which is automatically initiated every 17 minutes to assure the integrity of the electrode configuration and electrical operation of the detector/cable/readout system. Only a successful check turns on a front panel indicator light until the next check is received to provide indication of proper operation of the system. An unsuccessful check initiates a control room failure light on the front of the trip unit.

All electronics are solid state. All monitors are independent, and failure of one monitor has no effect on any other.

The radiation monitors are powered from a Class 1E power supply. Emergency power to this bus is provided by the station battery through an inverter.

System characteristics are given in Tables 12.3-6 and 12.3-7.

12.3.4.3.4 Safety Evaluation

The ICARM system is qualified to function in a loss-of-coolant accident environment. The system serves to monitor high range radiation levels in the containment and drywell during accident conditions. The system is not essential for safe shutdown of the plant.

The system is designed to operate unattended while detecting, indicating, and recording high range gamma radiation, primarily post-accident. The radiation dose rate from each detector is indicated and recorded remotely in the control room. The trip unit circuitry is such that an alarm light will be illuminated on the monitor if radiation levels exceed preset limits or if the system malfunctions or loses its power source.

12.3.4.3.5 Calibration and Testing

Each of the monitors is calibrated at 12, 35, and 300 R/hr by the instrument manufacturer prior to shipment using standards certified by the National Institute of Standards and Technology (NIST) or traceable to NBS. The manufacturer has stated that the equipment supplied does not utilize a radioactive check source for in-situ calibration.

The system includes an internal test trip circuit. The test signal is electrically fed into the unit so that a meter reading is provided in addition to a real trip. The test control is located on the front of the indicator trip unit. In addition, an automatic electronic “check source” test provides visual control room indication on system malfunction.

A channel calibration of the instruments will be performed in accordance with the Technical Specifications. It shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

12.3.5 References

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TABLE 12.3-1: RADIATION ZONE CLASSIFICATIONS

Designation	Maximum Dose Rate (mrem/hr)	Description
A	0.5	Uncontrolled, unlimited access
B	2.5	Controlled, limited access, 40 hr/wk
C	15	Controlled, limited access, 6-40 hr/wk
D	100	Controlled, limited access, 1-6 hr/wk
E	Over 100	Normally inaccessible, access for short periods for essential activities

TABLE 12.3-2: LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

GRACE I	Multigroup, multiregion, gamma-ray attenuation code used to compute gamma heating and gamma dose rates in slab geometry (Ref. 13)
GRACE II	Multigroup, multiregion, gamma-ray attenuation code used to compute the dose rate or heat generation rate for a spherical or a cylindrical source with slab or concentric shields (Ref. 14)
ANISN	Multigroup, multiregion code solving the Boltzman transport equation for neutrons or gamma-rays in one dimensional slab, cylindrical, or spherical geometry (Ref. 15)
SDC	Multigroup, multiregion, Kernal integration gamma-ray, shield design code which calculates dose rates for 13 geometry options (Ref. 17)
QAD	Multigroup, multiregion, three-dimensional, point Kernal code which calculates fast neutron and gamma-ray dose and heat generation rates (Ref. 18)
NAPD	Determines activation emission source strengths as a function of neutron exposure and decay time (Ref. 19)
MORSE-CG	Three-dimensional Monte Carlo neutron and gamma ray general transport code (Ref. 20)
DOT III	Two-dimensional neutron, gamma ray, discrete ordinate, transport code (Ref. 21)
ORIGEN	Isotope generation and depletion code which solves equations of radioactive growth and decay for isotopes of arbitrary coupling. (Ref. 23)
G^3	A general purpose gamma-ray scattering code. (Ref. 22)
MICROSHIELD	A general purpose gamma-ray scattering code. (Ref. 26)

TABLE 12.3-3: AREA RADIATION MONITOR LOCATIONS

Location Containment/Auxiliary Building	Elevation (ft)	Range (mR/hr)	Initial Set point
RHR room A	93	1-100,000	15
RHR room B	93	1-100,000	15
RCIC room	93	1-100,000	15
Component cooling water heat exchanger area	93	0.01-1,000	2.5
TIP drive mechanism area	119	0.01-1,000	15
Drywell equipment hatch	119	0.01-1,000	100
Drywell personnel airlock	119	1-100,000	100
Containment personnel airlock	119	0.01-1,000	2.5
CRD hydraulic units, north	139	0.01-1,000	2.5
CRD hydraulic units, south	139	0.01-1,000	2.5
RHR heat exchanger A removal hatch	139	1-100,000	15
RHR heat exchanger B removal hatch	139	1-100,000	15
SGTS filter train area	139	0.01-1,000	2.5
CRD repair room	166	0.01-1,000	2.5
Outside CRD repair room	166	0.01-1,000	2.5
Auxiliary building sample station	166	0.01-1,000	2.5
Containment ventilation equip. room	166	0.01-1,000	2.5
Hydrogen sample panel A	166	0.01-1,000	2.5
Hydrogen sample panel B	166	0.01-1,000	2.5
Drywell - east	166	1000-10E10	100-1000
Drywell - west	166	1000-10E10	100-1000
Containment ventilation filter train area	185	0.01-1,000	2.5
Containment sample station	185	0.1-10,000	15
Fuel handling area	208	0.01-1,000	2.5

TABLE 12.3-3: AREA RADIATION MONITOR LOCATIONS (Continued)

Location Containment/Auxiliary Building	Elevation (ft)	Range (mR/hr)	Initial Set point
Fuel handling area	208	0.01-1,000	2.5
Fuel handling area	208	0.01-1,000	2.5
Fuel handling area	208	0.01-1,000	2.5
Dryer storage area	208	0.01-1,000	15
Separator storage area	208	0.01-1,000	15
Containment fuel area - north	208	0.01-1,000	15
Containment fuel area - south	208	0.01-1,000	15
Containment personnel airlock	208	0.01-1,000	2.5
Containment - north	208	1000-10E10	10-100
Containment - south	208	1000-10E10	10-100
Turbine Building			
Turbine building filter train area	93	0.1-10,000	15
Turbine building sample station	93	0.1-10,000	15
Post-accident sample station	93	1-100,000	15
Mech vacuum pump area	113	0.1-10,000	2.5
Instrument rack area	113	0.01-1,000	2.5
Reactor feed pump area	133	1-100,000	15
Operating floor, turbine end	166	0.01-1,000	2.5
Operating floor, turbine end	166	0.01-1,000	2.5
Operating floor, generator end	166	0.01-1,000	2.5
Operating floor, generator end	166	0.01-1,000	2.5
Control Building			
Hot machine shop	93	0.01-1,000	2.5
Remote shutdown area	111	0.01-1,000	0.5
Control room	166	0.01-1,000	0.5
Technical Support Center	177	0.01-1,000	0.5

TABLE 12.3-3: AREA RADIATION MONITOR LOCATIONS (Continued)

Radwaste Building

Instrument rack area	93	0.01-1,000	2.5
Radwaste sample station	118	0.01-1,000	15
Control station	118	0.01-1,000	2.5
Distillate sample tank room	118	0.01-1,000	2.5
HVAC equipment room	136	0.1-10,000	15
Solid radwaste area	118	1-100,000	15

TABLE 12.3-4: AREA RADIATION MONITORING EQUIPMENT CHARACTERISTICS

Parameter	Characteristics
Detector Assembly	
Type	Gamma (Argon-filled, Halogen-quenched Geiger Muller tube)
Quantity	54
Mounting	Wall
Range	0.01 to 10^3 mR/hr 0.1 to 10^4 mR/hr as applicable 1.0 to 10^5 mR/hr 0.1 to 10^7 mR/hr
Energy dependence	±15% from 100 keV to 1.5 MeV
Local Alarm Unit	
Alarm	Audible and visual on high radiation visual on failure
Mounting	Wall
Indicator Trip Unit	
Range	0.01 to 10^3 mR/hr 0.1 to 10^4 mR/hr as applicable 1.0 to 10^5 mR/hr 0.1 to 10^7 mR/hr
Indicator	3.5-in scale panel meter
Indicator accuracy	±20% of reading (logarithmic)
Set points	
Accuracy	±20% of reading (logarithmic)
Range	Adjustable over full scale of indicator trip unit

TABLE 12.3-4: AREA RADIATION MONITORING EQUIPMENT CHARACTERISTICS (Continued)

Parameter	**Characteristics**
Adjustments	Internally mounted potentiometer with screwdriver adjustment
Overall Channel Accuracy	+/- 20% of reading (logarithmic)

TABLE 12.3-5: Airborne Radioactivity Monitors

					Upscale Set Point					
Monitored Process	**No of Channels**	**Detector Type**	**of Detector Channel Location Range**	**Channel Range**	**Warning Alarm**	**Trip**	**Scale**	**Purpose or Measurement**	**Principal Radionuclides Detected**	**Radiation level for 10 (MPC)= hours**
Drywell Radioactivity Monitoring System					Twice Normal		5 dec. log	Alarm	Cs-137	$1x10^6$
					Twice Normal	Not Applicable	5 dec. log	Alarm	I-131	$5x10^5$
					Twice Normal (See Note 2)		5 dec. log	Alarm	Xe-133, Kr-85	$4x10^3$
Particulate	1	Beta scintillator	Drywell	$10-10^6$ cpm						
Noble gas	1	Beta scintillator	Drywell	$10-10^6$ cpm						
Containment and Drywell Vent Exhaust	4	Geiger-Muller Tube	Exhaust duct upstream of exhaust ventilation isolation valve	0.01 mr/hr to 100 mr/hr	Technical Specification	Technical Specification	4 dec. log	Monitor exhaust - Isolates containment ventilation	Xe133,	2.5 mr/hr
Auxiliary Building and Fuel Handling Area Vent Exhaust	4	Geiger-Muller Tube	Exhaust duct upstream of exhaust ventilation isolation valve	0.01 mr/hr to 100 mr/hr	Technical Specification	Technical Specification	4 dec. log	Isolate building & initiate standby gas treatment	Xe-133, Xe-135, Kr-85, Kr-87, 88	1.7x10-2mr/hr
	1	Scintillation Detector	Sample Line	$10-10^6$ cpm	Technical Specification	Not Applicable	5 dec. log	Audit discharge to environs	I-131	140 cpm
	1	Scintillation Detector	Sample Line	$10-10^6$ cpm	Technical Specification	Not Applicable	5 dec. log	Audit discharge to environs	Cs-137	260 cpm

TABLE 12.3-5: Airborne Radioactivity Monitors (Continued)

					Upscale Set Point					
Monitored Process	**No of Channels**	**Detector Type**	**of Detector Channel Location Range**	**Channel Range**	**Warning Alarm**	**Trip**	**Scale**	**Purpose or Measurement**	**Principal Radionuclides Detected**	**Radiation level for 10 (MPC)= hours**
	1	Scintillation Detector	Sample Line	10^{-7} to 6×10^{-2} µCi/cc	Technical Specification	Not Applicable	5 dec. log	Audit discharge to environs	Xe-133, Kr-85	25 cpm
	1	Geiger-Muller Tube	Sample Line	10^{-7} to 6×10^{-2} µCi/cc	Technical Specification	Not Applicable	4 dec. log	Audit discharge to environs	Xe-133, Kr-85	-
	1	Geiger-Muller Tube	Sample Line	2×10^{-2} to 4 x 10µCi/cc	Technical Specification	Not Applicable	5 dec. log	Audit discharge to environs	Xe-133, Kr-85	-
	1	Geiger-Muller Tube	Sample Line	10^{-4} to 10^{1} µCi/cc	Technical Specification	Not Applicable	4 dec. log	Audit discharge to environs	Xe-133, Kr-85	-
Auxiliary Building and Fuel Handling Area Pool Sweep Exhaust	4	Geiger-Muller Tube	Exhaust duct upstream of exhaust ventilation isolation valve	0.01 mr/hr to 100 mr/hr	Technical Specification	Not Applicable	4 dec. log	Isolate building & initiate standby gas treatment	Xe-133, Xe-135, Kr-85, Kr-87, 88, I-131	4.2 mr/hr
Control Room Ventilation	4	Geiger-Muller Tube	Exhaust duct upstream of exhaust ventilation isolation valve	0.01 mr/hr to 100 mr/hr	Technical Specification	Technical Specification	4 dec. log	Isolate control room & initiate emergency ventilation	Xe-133, Kr-85, I-31, Cs-137	4.4

Note:

1. Since there are no filters in this exhaust stream (see figure 9.4-2), the effluent monitors listed in Table 11.5-1 will also perform the function of worker protection and have sufficient sensitivity to measure 10 MPC-hours.
2. Twice the normal drywell radioactivity levels at full reactor power.

TABLE 12.3-6: IN-CONTAINMENT AREA RADIATION MONITORS

Location	Elevation (ft)	Range (R/hr)
Containment - north	208	1 to 10^7
Containment - south	208	1 to 10^7
Drywell - east	166	1 to 10^7
Drywell-west	166	1 to 10^7

TABLE 12.3-7: IN-CONTAINMENT AREA RADIATION MONITORING EQUIPMENT CHARACTERISTICS

Parameter	Characteristics
Detector assembly	
Type	Gamma (Nitrogen-filled ion chamber)
Quantity	2-Containment 2-Drywell
Mounting	Wall
Range	Range 1.0 to 10^7 R/hr
Energy Dependence	±20 percent from 0.1 MeV to 3.0 MeV
Control Room Alarm Unit	
High radiation	Alarm indicating light on monitor
Failure	Alarm indicating light on monitor
Indicator Trip Unit	
Range	1.0 to 10^7 R/hr
Indicator	4-inch scale panel meter
Indicator Accuracy	±5 percent around the midpoint of each decade
Overall Channel Accuracy	±36 percent

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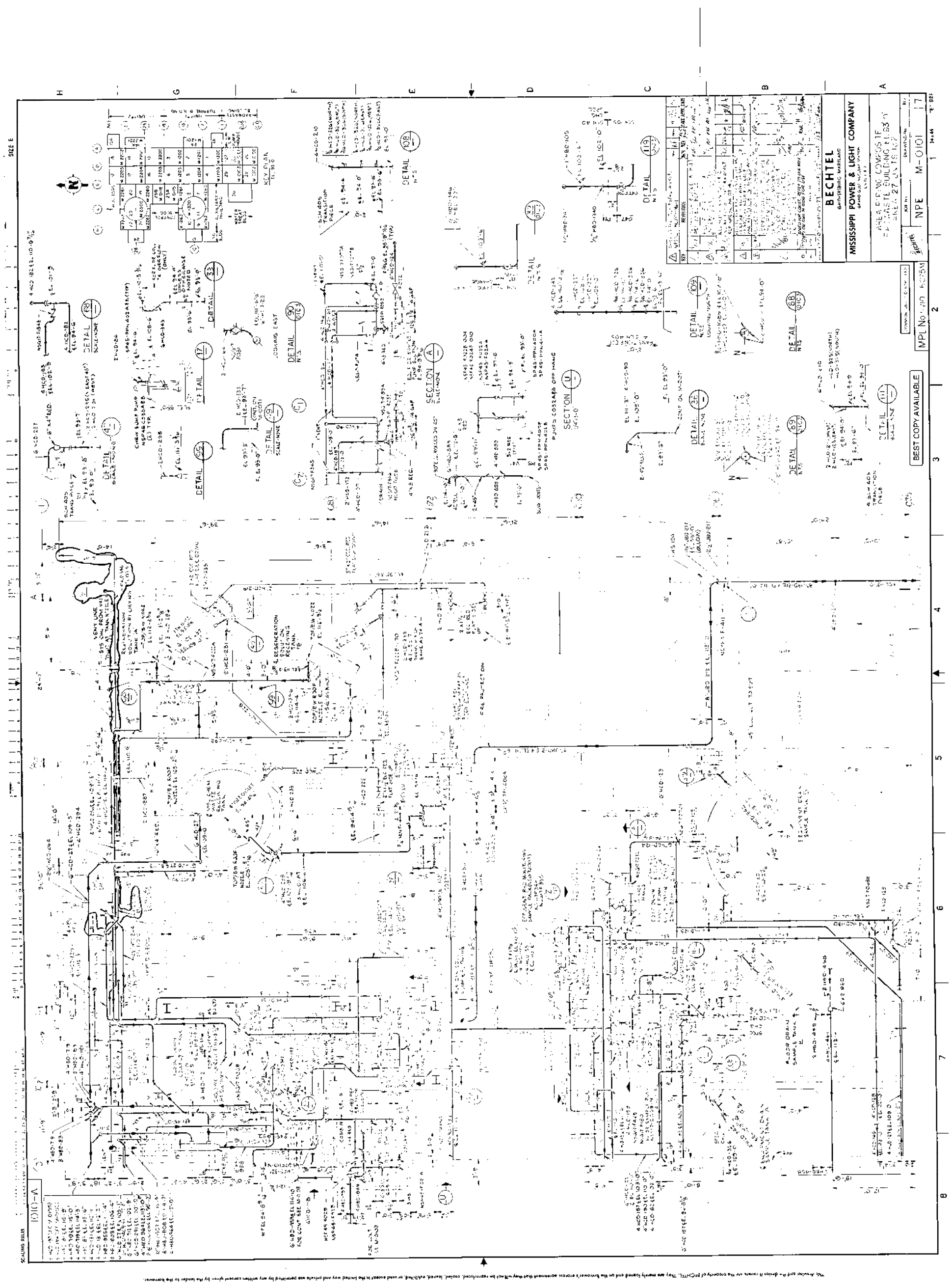

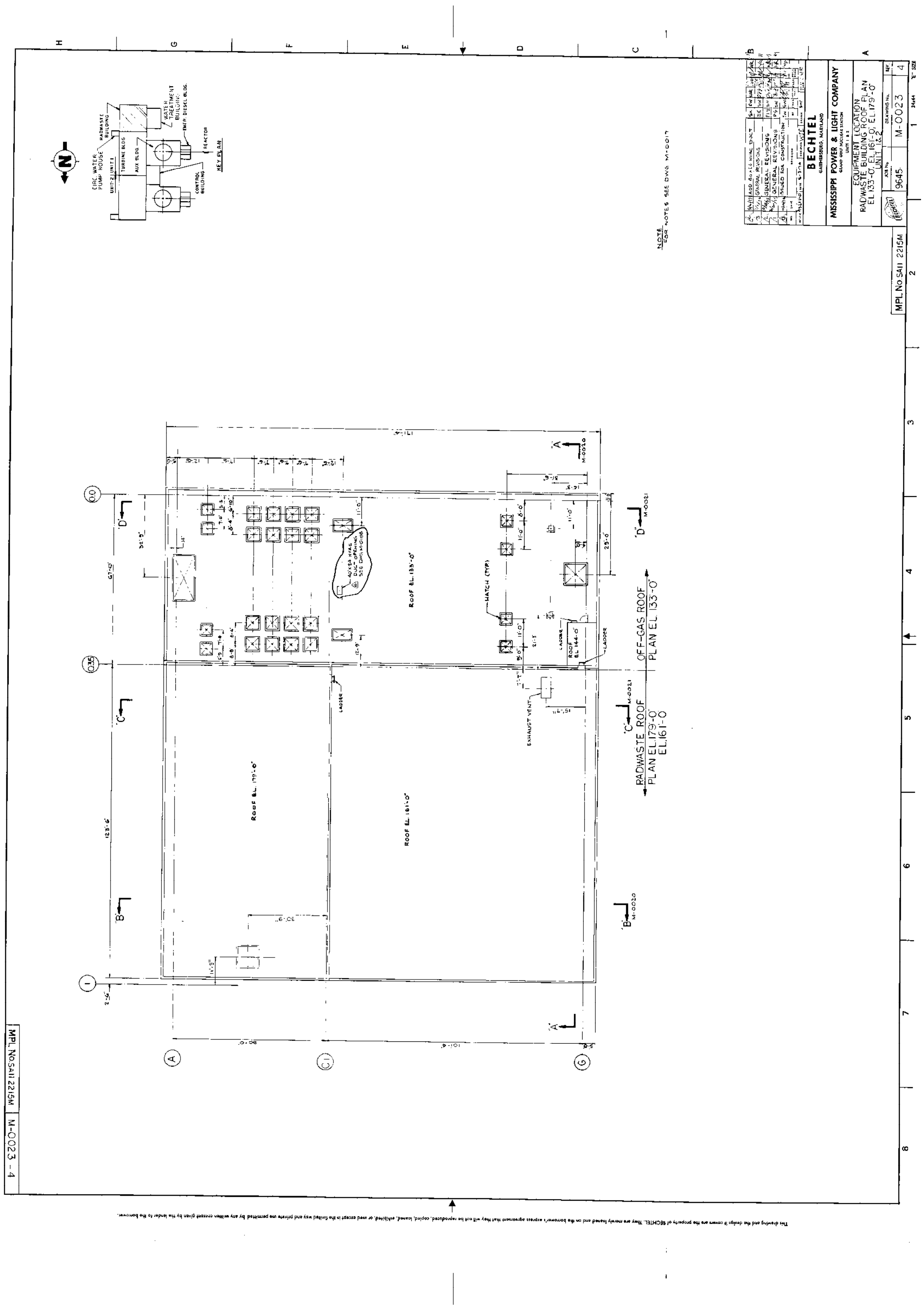
OFF-GAS ROOF
PLAN EL 133'-0"
RADWASTE ROOF
PLAN EL.179'-0"
EL.161-0
ROOF EL. 133'-0"
ROOF EL. 179'-0"
ROOF EL. 161'-0"
KEY PLAN
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
EQUIPMENT LOCATION
RADWASTE BUILDING ROOF PLAN
EL.133-0, EL.161-0, EL.179'-0"
UNIT 1&2
9645
M-0023
MPL No SA11 2215M
M-0023-4

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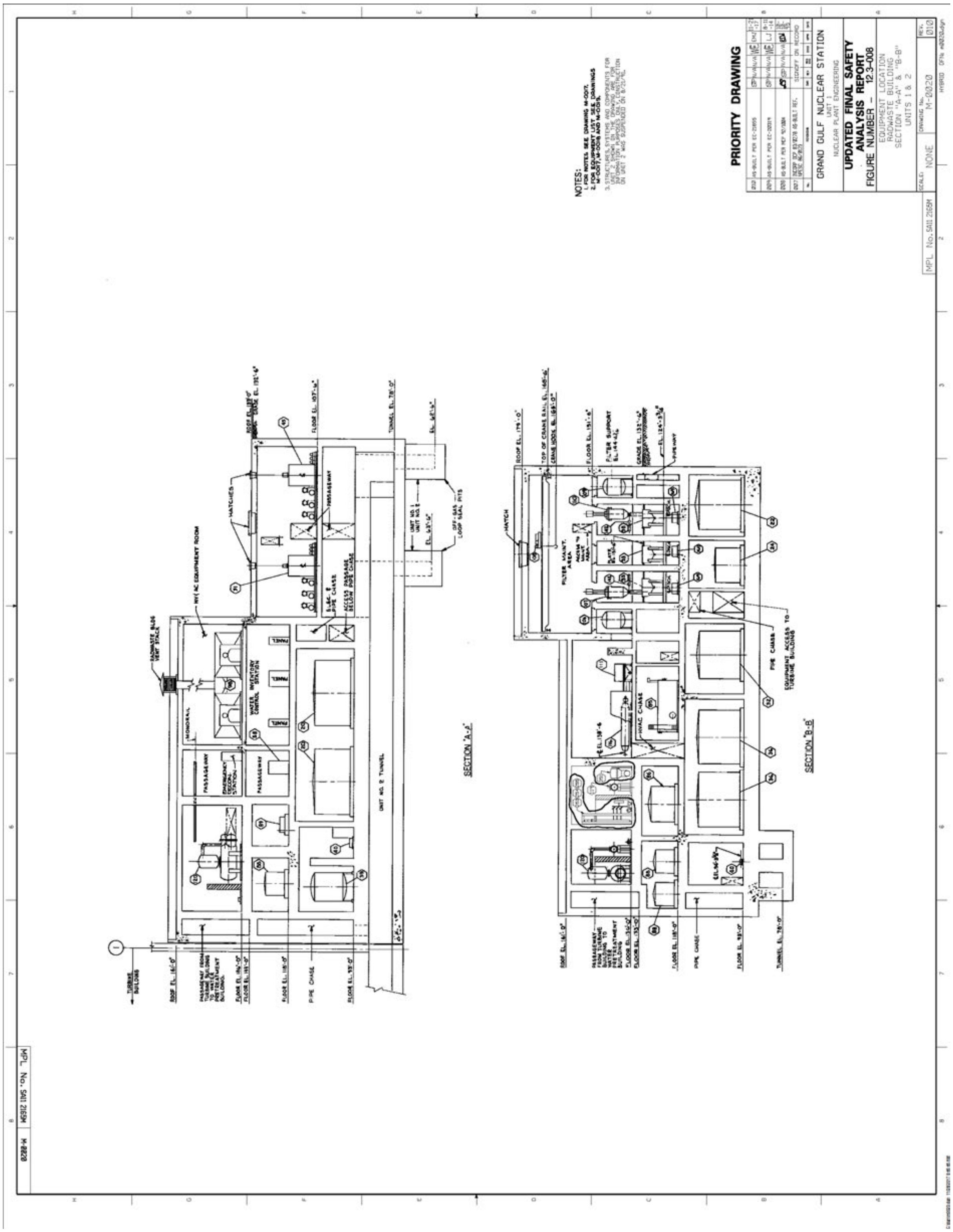
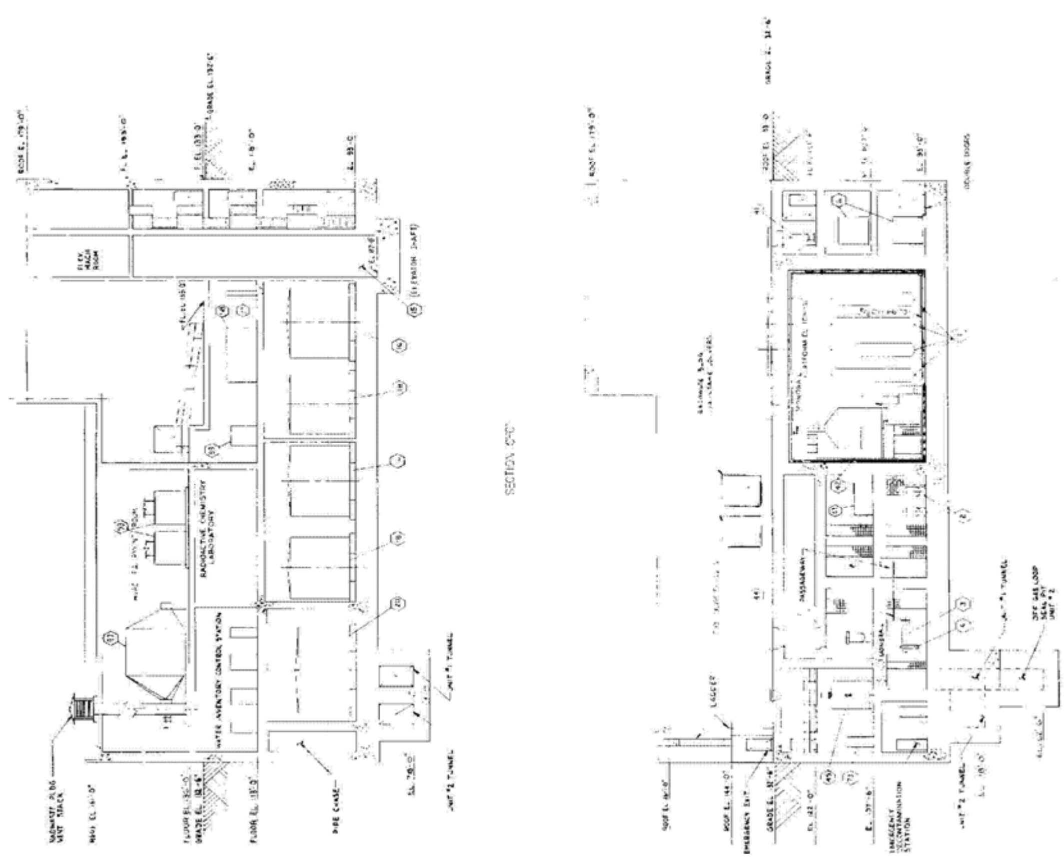
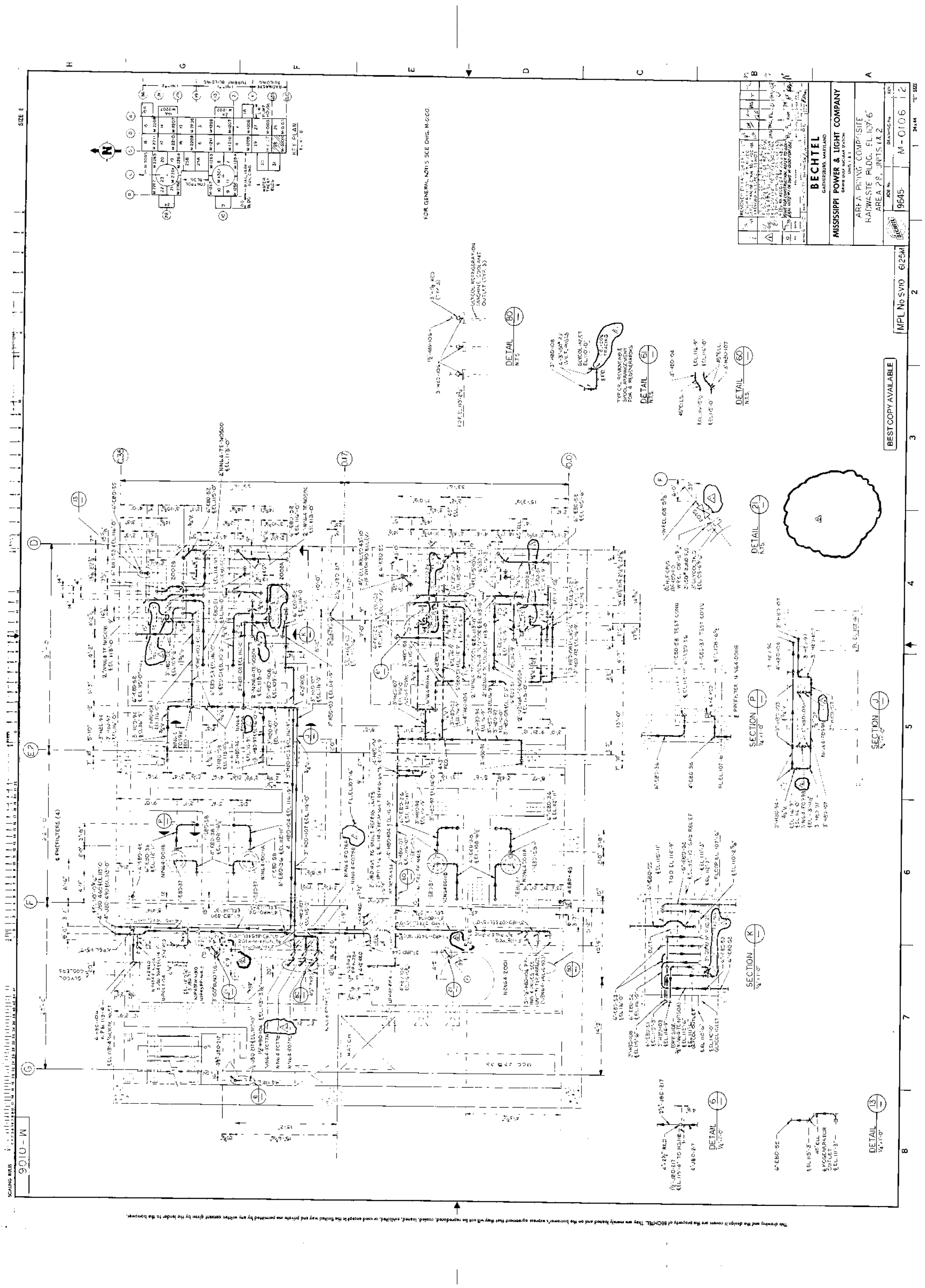
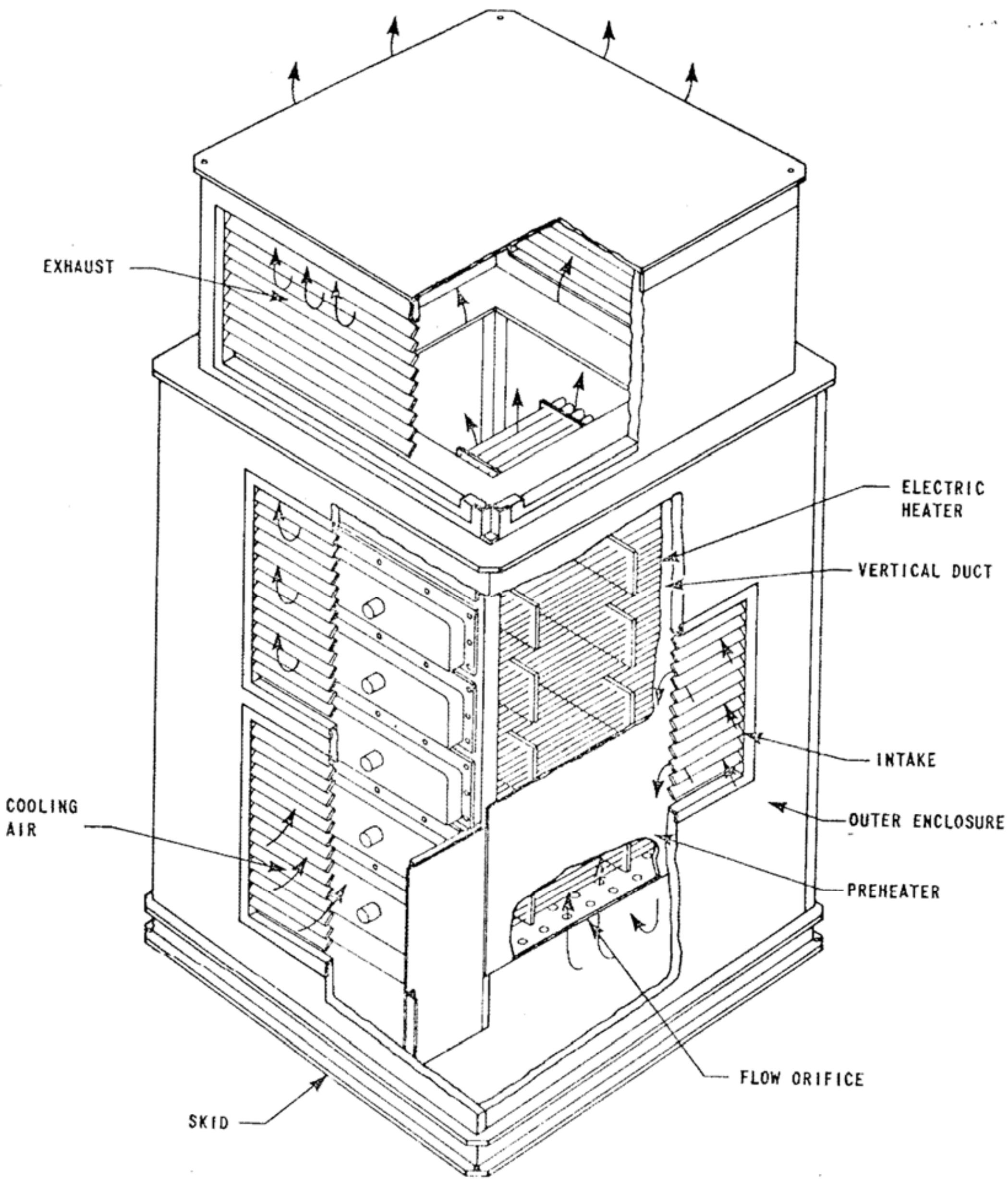

PRIORITY DRAWING
GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER – 12.3-008
EQUIPMENT LOCATION
RADWASTE BUILDING
SECTION "A-A" & "B-B"
UNITS 1 & 2
SCALE: NONE
DRAWING No. M-0020
REV. 010
MPL No. SA11 2165M
NOTES:
SECTION "A-A"
SECTION "B-B"



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

EQUIPMENT LOCATION
RADWASTE BUILDING
SECTIONS C-C AND D-D
FIGURE 12.3-9

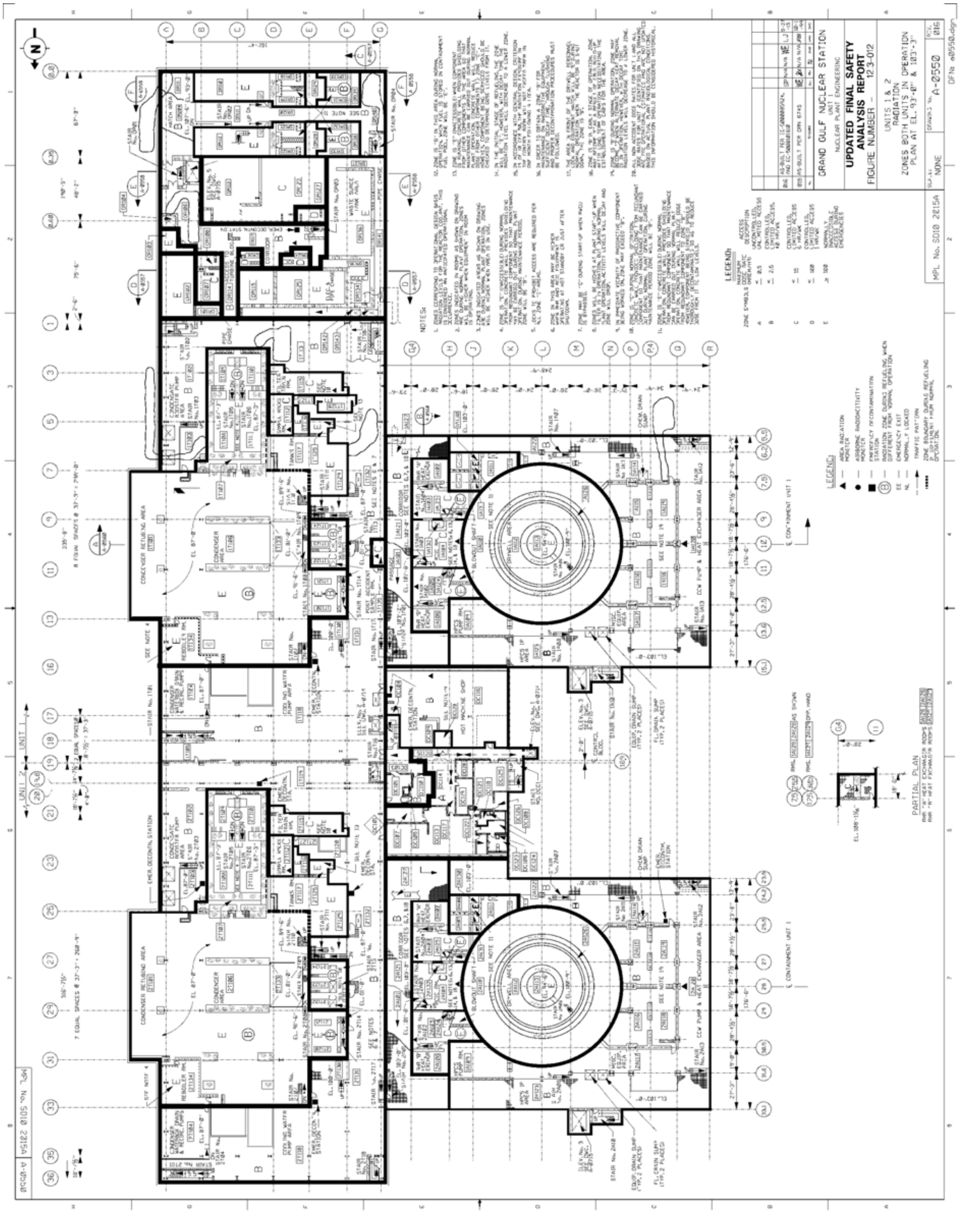

BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
BEST COPY AVAILABLE

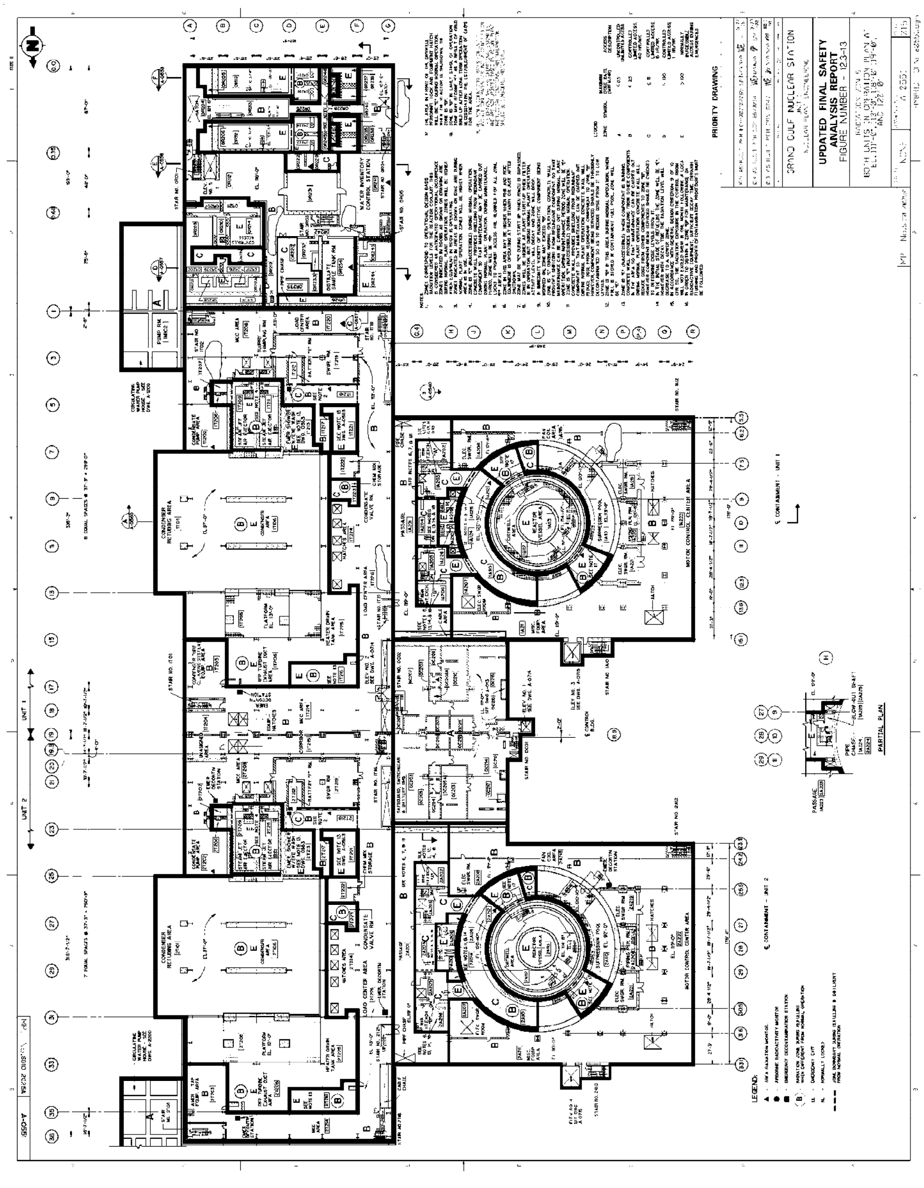


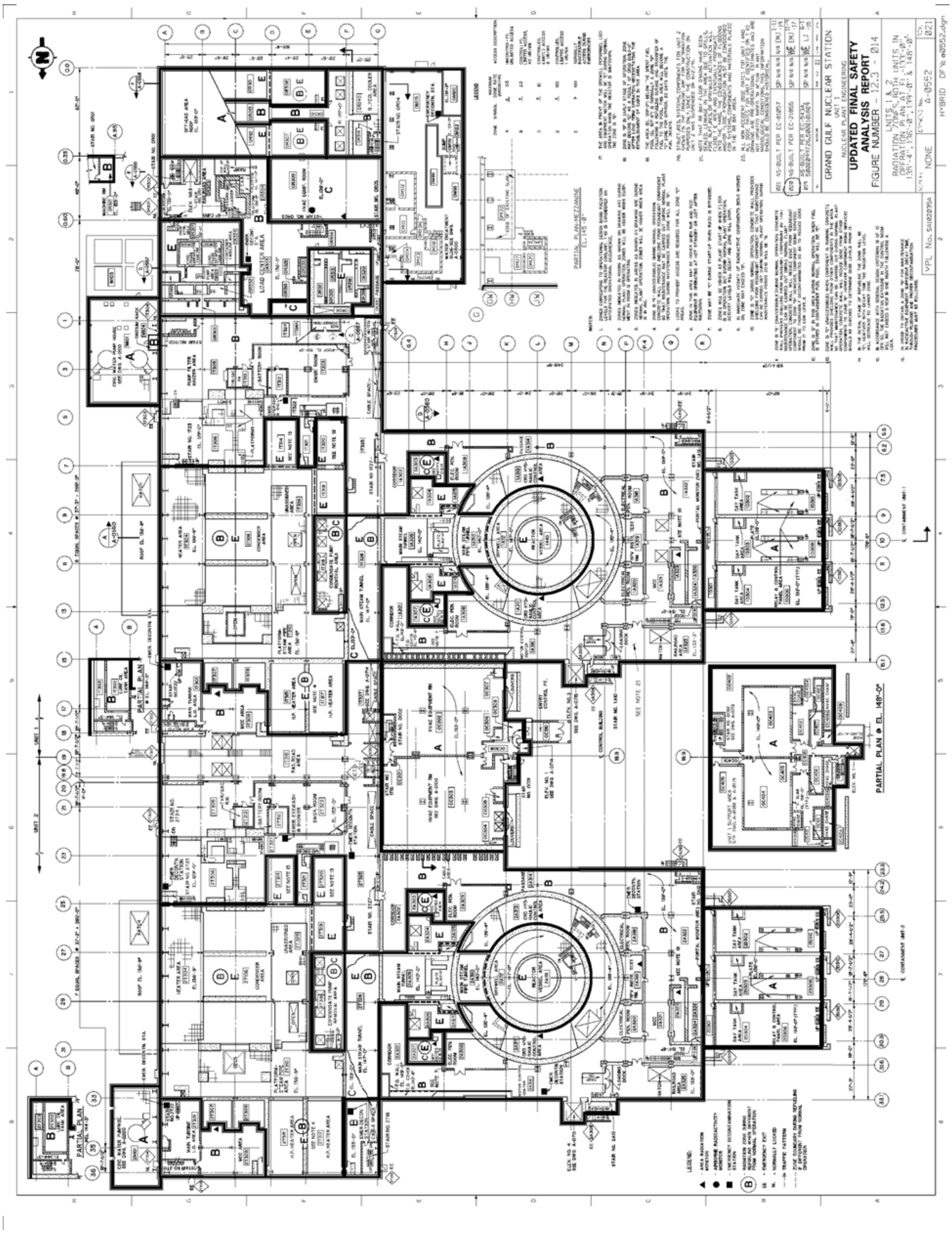
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

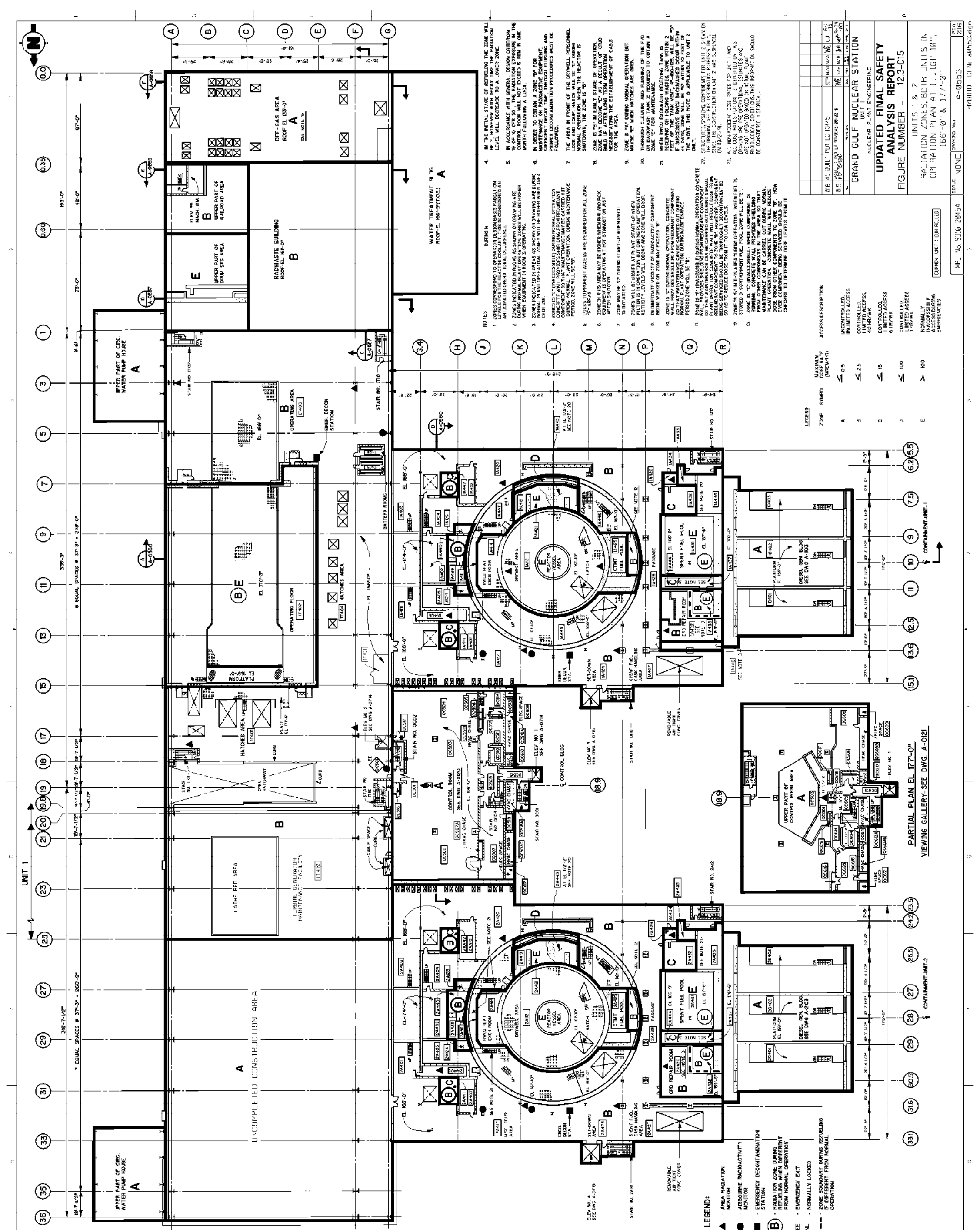
CUTAWAY VIEW OF
ELECTRIC HYDROGEN RECOMBINER

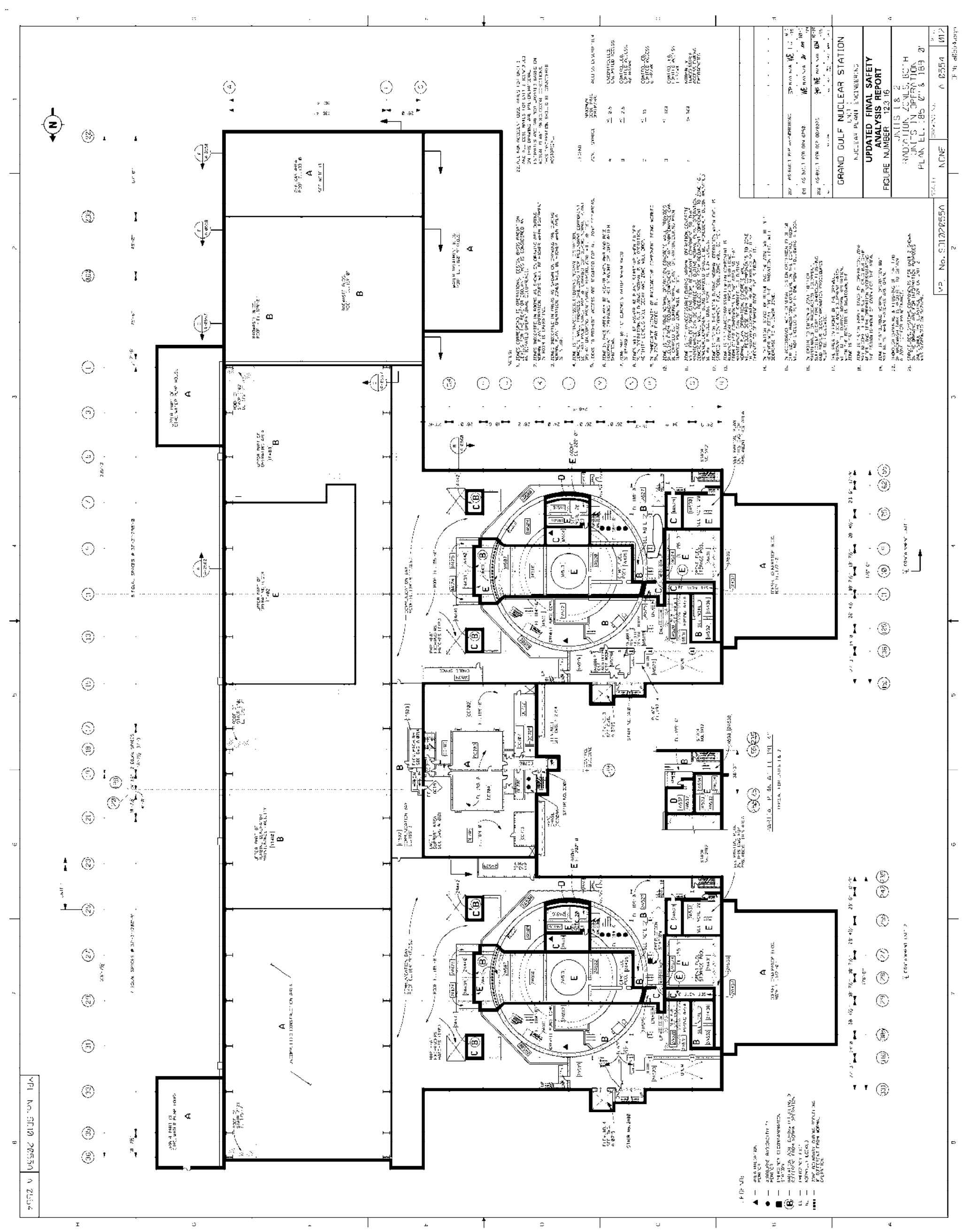
FIGURE 12.3-11


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-012
UNITS 1 & 2
RADIATION
ZONES BOTH UNITS IN OPERATION
A-0550


PRIORITY DRAWING
GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-13


GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3 - 014
UNITS 1 & 2
RADIATION ZONES, BOTH UNITS IN OPERATION PLAN AT EL. 133'-0", 135'-4", 136'-0", 139'-0" & 148'-0"
A-0552
VPL No. SA1020354
HYBRID DFN: a0552.dgn


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-015
PARTIAL PLAN EL 177'-0"
VIEWING GALLERY-SEE DWG. A-0121
UNIT 1
UNCOMPLETED CONSTRUCTION AREA
LEGEND:



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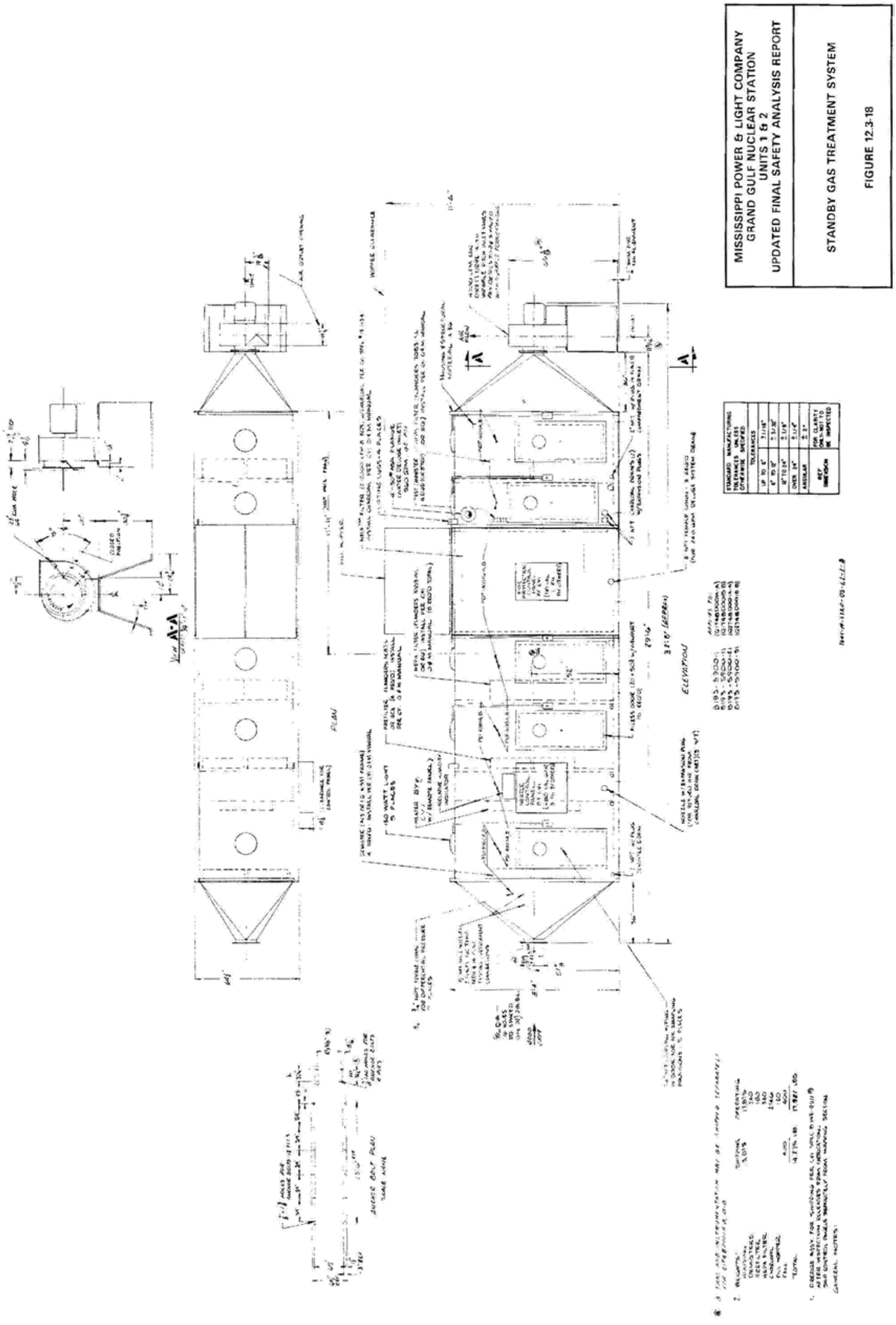
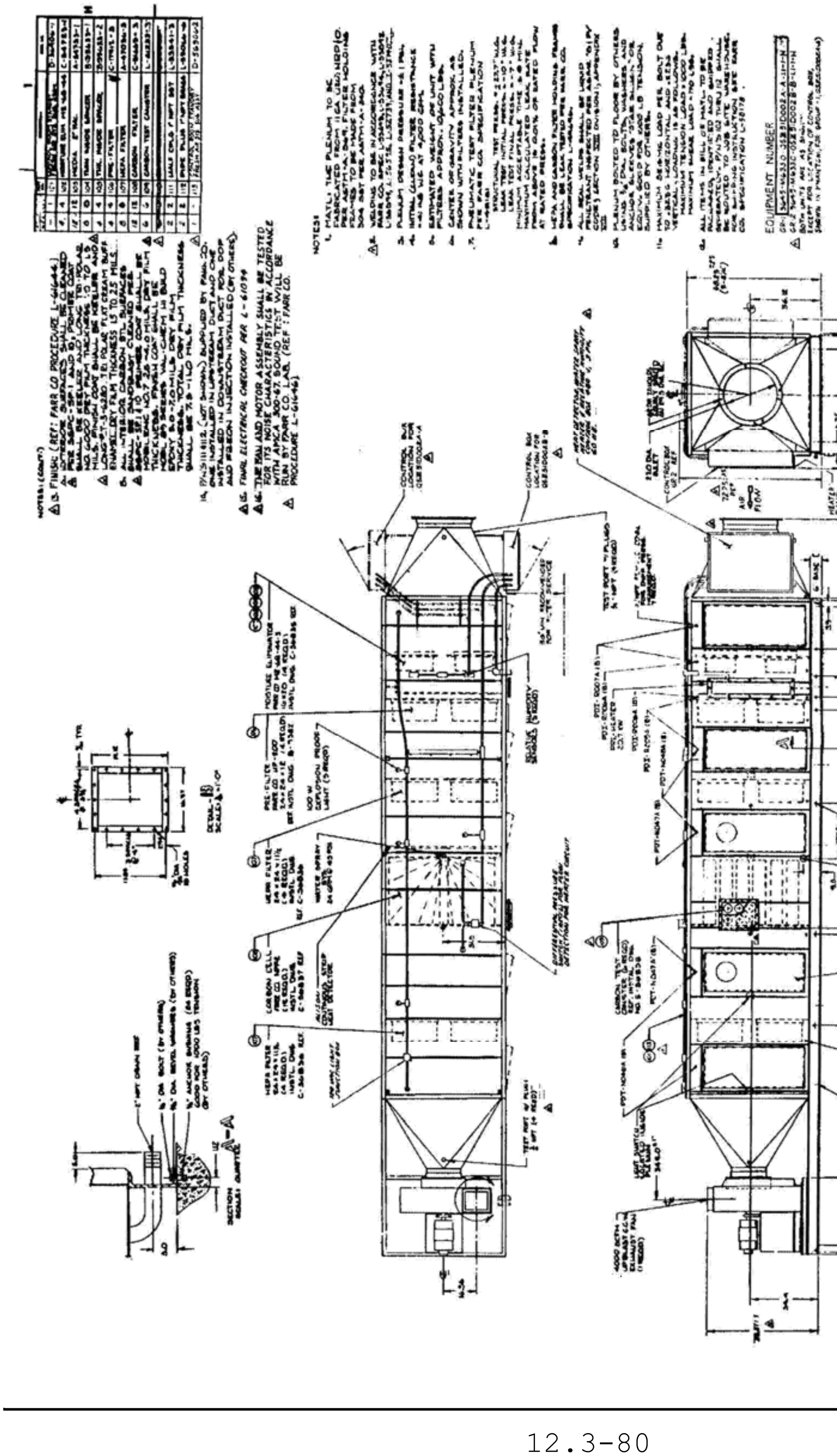
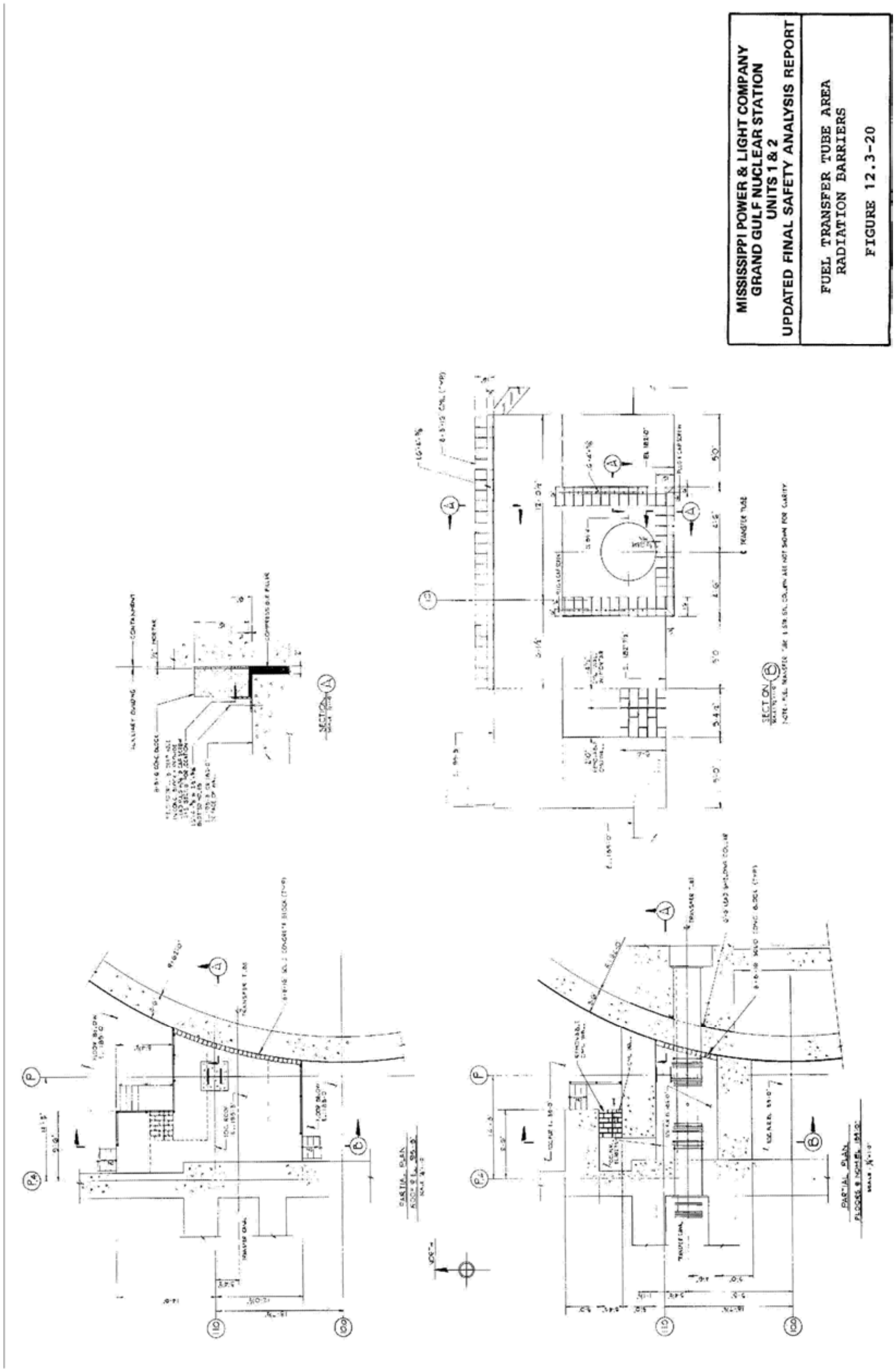

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
STANDBY GAS TREATMENT SYSTEM
FIGURE 12.3-18

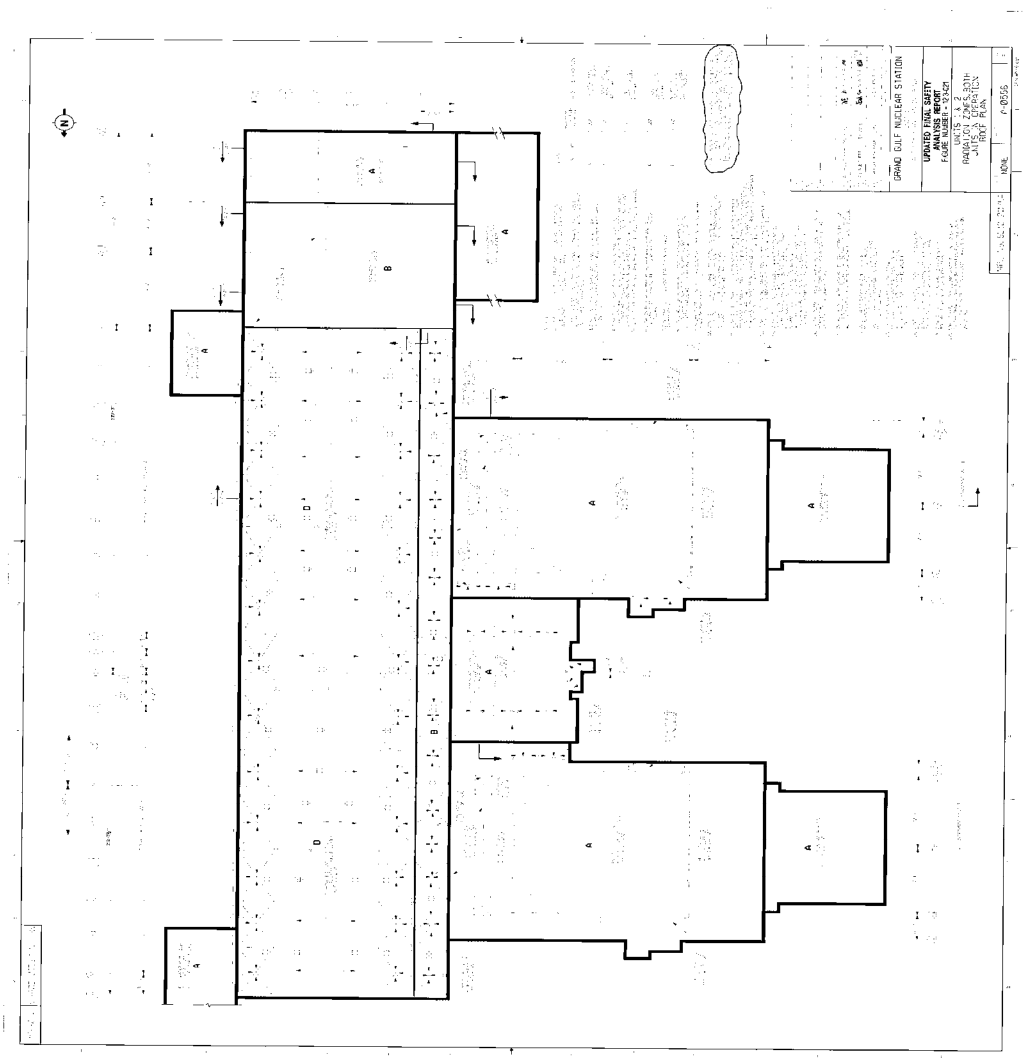


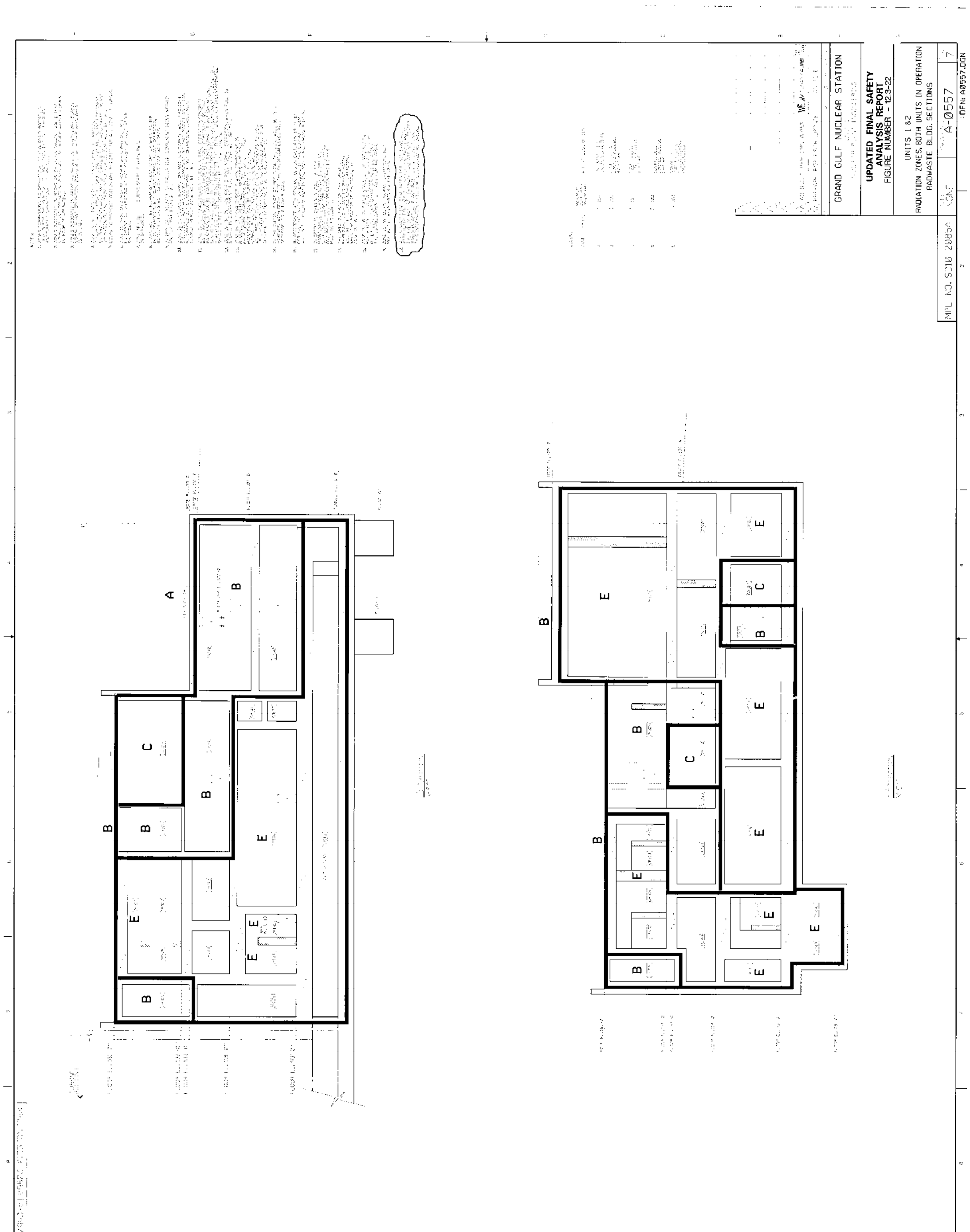
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

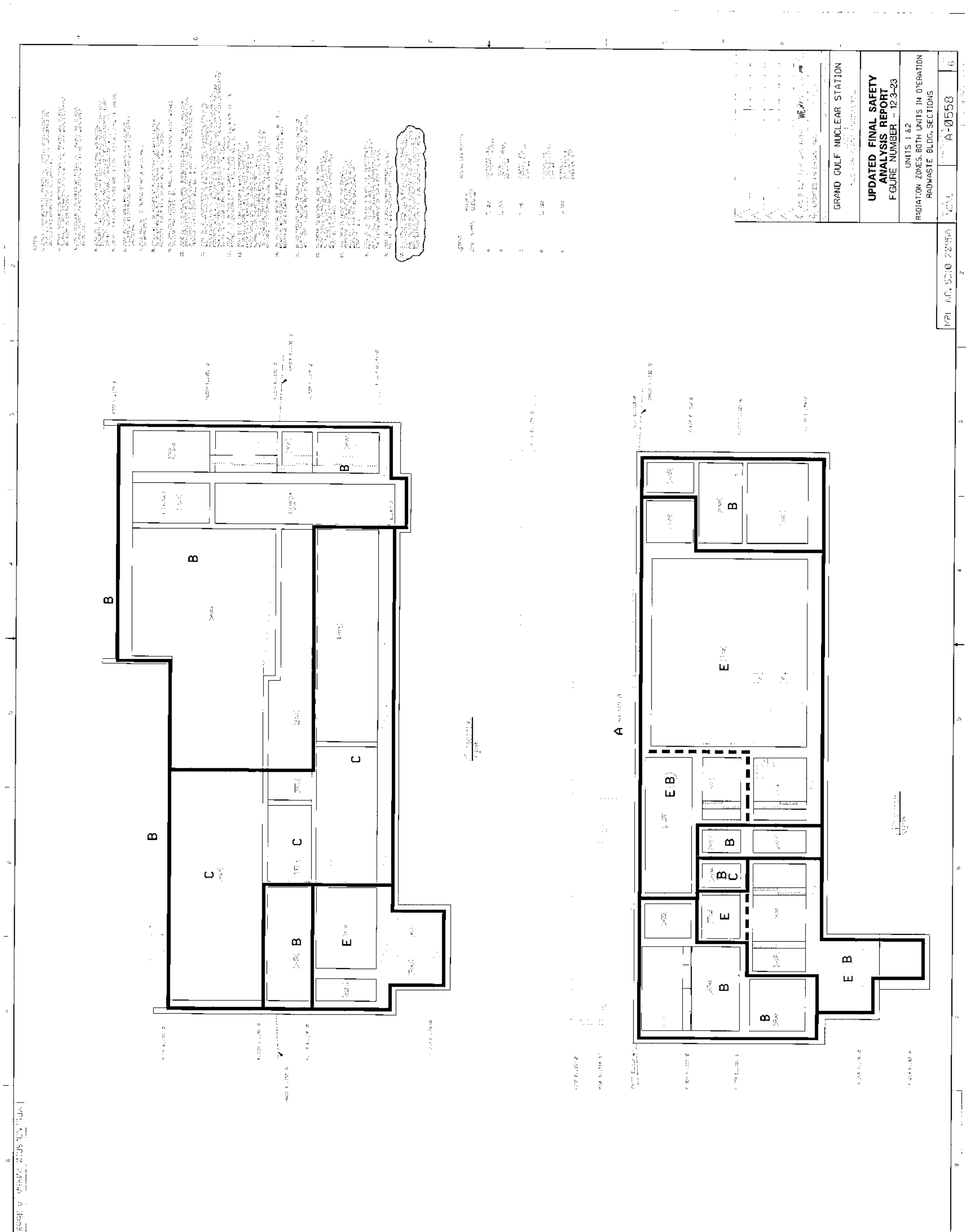
CONTROL ROOM
STANDBY FRESH AIR SYSTEM

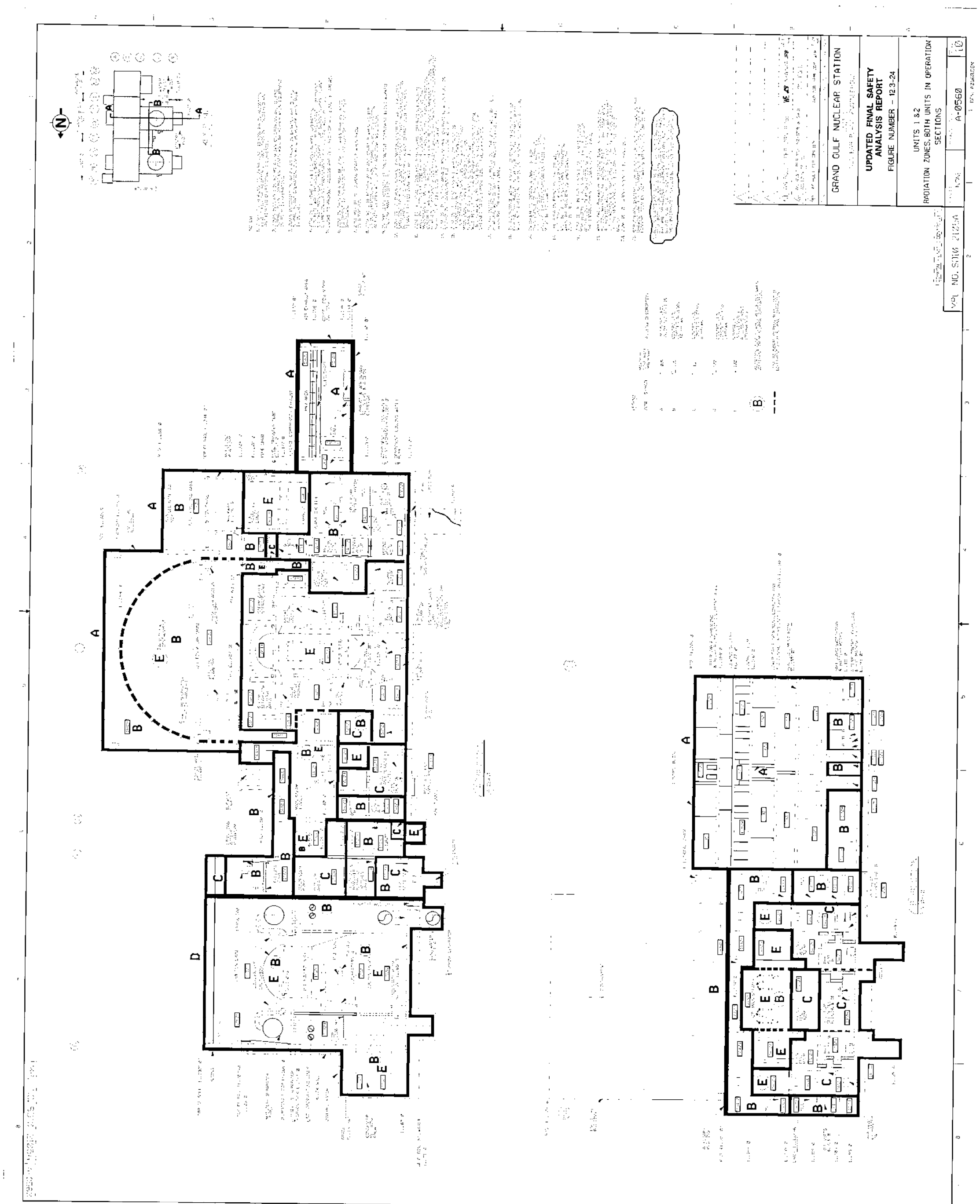
FIGURE 12.3-19

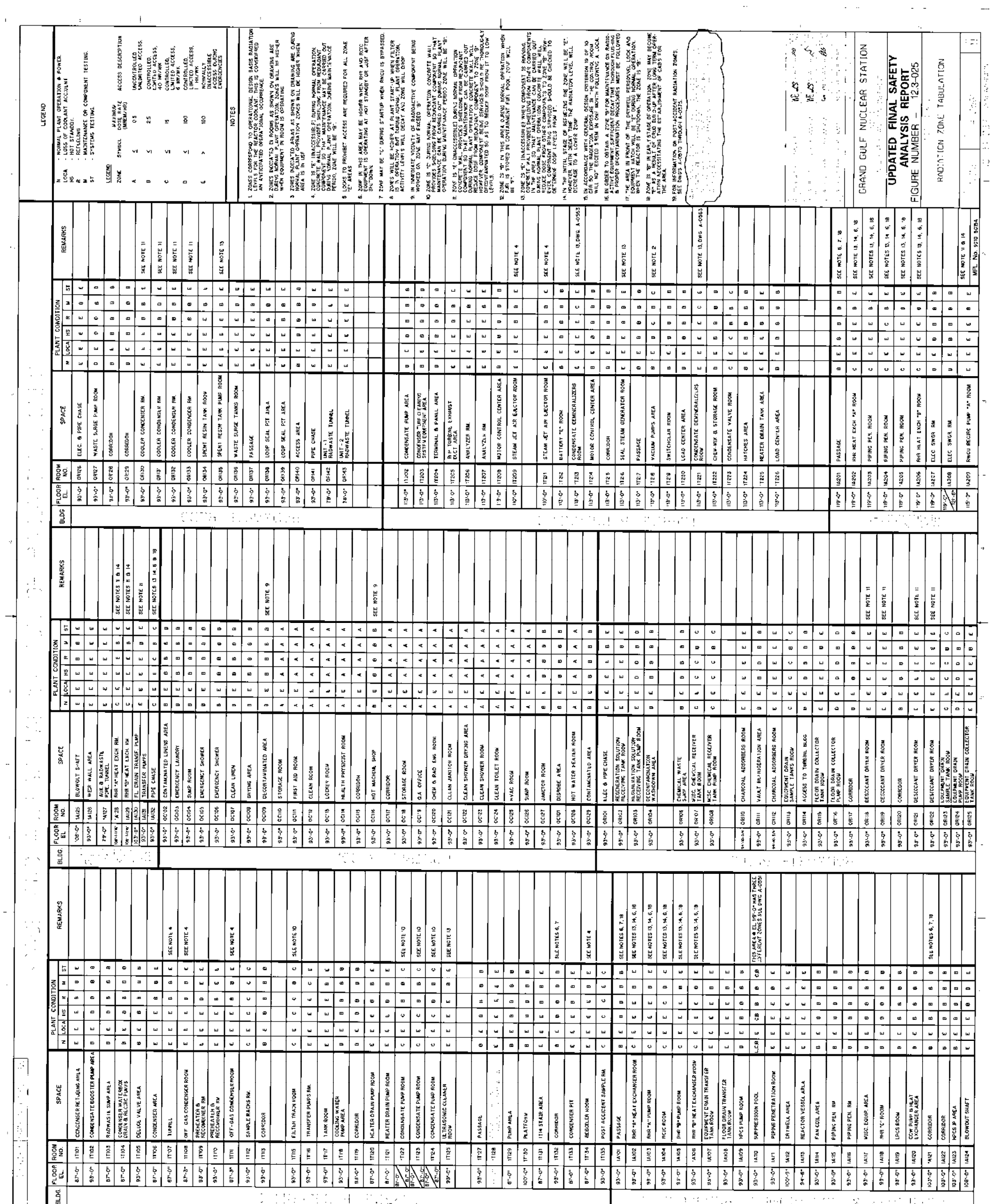

MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
FUEL TRANSFER TUBE AREA
RADIATION BARRIERS
FIGURE 12.3-20


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-021
UNITS 1 & 2
RADIATION ZONES, BOTH UNITS IN OPERATION
ROOF PLAN
NONE
A-0556


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-22
UNITS 1 & 2
RADIATION ZONES, BOTH UNITS IN OPERATION
RADWASTE BLDG. SECTIONS
A-0557
A
B
C
E


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-23
UNITS 1 & 2
RADIATION ZONES, BOTH UNITS IN OPERATION
RADWASTE BLDG. SECTIONS
A-0558


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-24
UNITS 1 & 2
RADIATION ZONES, BOTH UNITS IN OPERATION
SECTIONS
A-0560


GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER 12.3-025
RADIATION ZONE TABULATION

MPL NO. S010 5025A | A-0562

SPACE AND ACCESS TABULATION OF RADIATION ZONES

BLDG.	FLOOR EL.	ROOM NO.	SPACE	N	LOCA	HS	R	M	ST	REMARKS
AUXILIARY & CONTAINMENT BLDG - UNIT 1	119'-0"	1A210	HW.U RECIRC PUMP "D" AREA	E	F	E	E	B	E	SEE NOTES 11 & 14
	119'-0"	1A211	MISC. EQUIP. AREA	H	E	B	B	B	H	
	120'-10"	1A212	CONCRETE PLATFORM	C	E	C	B	B	B	
	119'-0"	1A215	FAN COIL AREA	B	E	B	B	B	B	
	119'-0"	1A219	ELEC. SWGR ROOM	B	E	B	B	B	B	
	173'-6"	1A220	PIPING PEN. ROOM	B	E	B	B	B	B	
	119'-0"	1A221	ELEC. SWGR ROOM	B	C	B	B	B	B	
	119'-0"	1A222	MOTOR CONTROL CENTER AREA	B	E	B	B	D	D	
	120'-0"	1A223	PASSAGE	E	E	F	E	B	E	SEE NOTE 13
	120'-0"	1A224	PIPE CHASE	E	E	E	E	B	E	SEE NOTE 13
	120'-0"	1A225	BLOWOUT SHAFT	E	E	C	E	B	E	SEE NOTE 13
	135'-0"	1A226	PIPE CHASE	E	E	C	E	B	E	SEE NOTE 13
CONTROL BLDG.	111'-0"	0C201	STAIR NO. 0C02	A	E	A	A	A	A	
	111'-0"	0C202	DIV. I SWITCHGEAR AREA	A	E	A	A	A	A	
	111'-0"	0C203	DIV. II SWITCHGEAR AREA	A	E	A	A	A	A	
	111'-0"	0C204	DIV. III BATTERY ROOM	A	E	A	A	A	A	
	111'-0"	0C205 / 0C205A	EMER. HOT SHUT-DN. RM.	A	E	A	A	A	A	
	111'-0"	0C206	DIV. II BATTERY ROOM	A	F	A	A	A	A	
	111'-0"	0C207	DIV. I BATTERY ROOM	A	F	A	A	A	A	
	111'-0"	0C208 / 0C208A	EMER. HOT SHUT-DN. RM.	A	F	A	A	A	A	
	111'-0"	0C209	DIV. III BATTERY ROOM	A	C	A	A	A	A	
	111'-0"	0C210	DIV. III SWITCHGEAR ROOM	A	E	A	A	A	A	
	111'-0"	0C211	DIV. II BATTERY ROOM	A	E	A	A	A	A	
	111'-0"	0C212	DIV. I BATTERY ROOM	A	E	A	A	A	A	
	111'-0"	0C213	DIV. III SWITCHGEAR ROOM	A	E	A	A	A	A	
	111'-0"	0C214	DIV. I SWITCHGEAR AREA	A	E	A	A	A	A	
	111'-0"	0C215	DIV. II SWITCHGEAR AREA	A	E	A	A	A	A	
	111'-0"	0C216	CORRIDOR	A	E	A	A	A	A	
	111'-0"	0C217	HVAC CHASE	A	E	A	A	A	A	
	111'-0"	0C218	PIPE CHASE	A	E	A	A	A	A	
	111'-0"	0C219	PIPE CHASE	A	E	A	A	A	A	
RADWASTE BLDG.	118'-0"	0R201	INSTRUMENT RACK AREA	E	E	E	C	C	E	
	118'-0"	0R201A	SLURRY SAMPLING RM.	F	E	E	C	C	E	
	118'-0"	0R202	DECON. PICK-UP AREA	E	E	E	E	C	E	
	118'-0"	0R203	DECAY TANK RM. "A"	E	E	E	E	C	C	
	107'-6" / 125'-6"	0R204	HATCH AREA	B	E	B	B	D	B	
	125'-6"	0R205	AFTER FILTER RM.	E	E	E	E	E	E	
	125'-6"	0R206	AFTER FILTER RM.	E	E	E	E	E	E	
	118'-0"	0R207	DISCHARGE PUMP RM.	E	E	E	E	C	E	
	118'-0"	0R208	WASTE TANK RM.	E	E	E	E	C	E	
	118'-0"	0R209	WASTE TANK RM.	E	E	E	F	C	E	
	118'-0"	0R210	VALVE & PIPE RM.	E	F	E	E	C	E	
	118'-0"	0R211	DECANT PUMP RM.	E	F	E	E	C	E	
	118'-0"	0R212	LOAD CENTER & TANK RM.	B	E	B	B	B	B	
	107'-6"	0R213	PLATFORM	D	E	B	B	B	B	
	118'-0"	0R214	PHASE SEPARATOR DECAY TANK RM. "B"	E	E	E	E	C	E	
	118'-0"	0R215	CORRIDOR	C	E	C	C	C	C	
	118'-0"	0R216	WASTE TANK RM.	E	E	E	E	C	E	
	118'-0"	0R217	DECON. PICK-UP AREA	E	F	E	E	C	E	
	118'-0"	0R219	ACCESS TO TURBINE BLDG.	B	E	B	B	B	B	
	118'-0"	0R220	MCC AREA	C	E	C	C	C	C	

BLDG.	FLOOR EL.	ROOM NO.	SPACE	N	LOCA	HS	R	M	ST	REMARKS
RADWASTE BLDG.	118'-3"	0R222	OFF GAS VAL SAMPLE RACK RM.	C	E	C	C	B	C	
	107'-6"	0R223	REGENERATOR RM.	E	E	E	B	B	E	SEE NOTE 11
	107'-6"	0R224	REGENERATOR RM.	E	E	E	B	B	E	SEE NOTE 11
	107'-0"	0R225	CORRIDOR	B	E	A	B	H	A	
	107'-6"	0R226	REGENERATOR RM.	E	E	E	H	H	E	SEE NOTE 11
	107'-0"	0R227	REGENERATOR RM.	E	E	E	B	D	C	SEE NOTE 11
	107'-6"	0R228	PASSAGE	E	E	D	B	B	B	
	118'-0"	0R229	SAMPLE STATION	E	E	C	C	B	C	
	118'-0"	0R231	HOT LABORATORY	B	C	B	B	B	B	SEE NOTE 3
	118'-0"	0R232	DISTILLATE SAMPLE TANK RM.	B	E	B	B	B	D	SEE NOTE 14
	118'-0"	0R233	HVAC DUCT CHASE	B	E	H	B	B	B	
	118'-0"	0R234	DISTILLATE SAMPLE TANK RM.	B	E	H	B	B	B	
	127'-6"	0R235	PREFILTER ROOM	E	E	E	E	E	E	SEE NOTE 11
	127'-6"	0R236	PREFILTER ROOM	E	E	E	E	E	E	SEE NOTE 11
	127'-6"	0R237	PREFILTER ROOM	E	E	E	E	E	E	SEE NOTE 11
	127'-6"	0R238	PREFILTER ROOM	E	E	E	E	E	E	SEE NOTE 11
	118'-0"	0R239	OIL SEPARATOR AREA	C	E	B	B	B	B	
	107'-6"	0R240	GLYCOL COOLER AREA	B	E	B	B	D	D	
	118'-0"	0R241	WATER INVENTORY CONTROL STATION	B	C	B	B	B	B	
	118'-0"	0R242	CORRIDOR	B	C	B	B	B	B	
	118'-0"	0R243	EVAPOR. BOT. PUMP RM.	E	E	E	E	E	E	
	118'-0"	0R244	EVAPOR. BOT. STG TANK RM.	C	E	F	F	C	F	
	118'-0"	0R245	PASSAGE	C	E	C	H	B	C	
	118'-0"	0R246	PASSAGE	C	E	C	H	C	C	
	118'-0"	0R247	PASSAGE	B	E	B	B	B	B	
TURBINE BLDG. UNIT - 1	133'-0"	1T301	MAIN TURBINE L. O. AREA	B	E	B	B	B	B	
	144'-3"	1T302	LUBE OIL TANK AREA	B	E	B	B	B	B	
	144'-3"	1T303	ELECTRICAL AREA	B	E	H	H	B	B	
	150'-5"	1T304	HEATER AREA	E	E	E	E	E	E	SEE NOTE 4
	133'-0"	1T305	PUMP FILTER DEIONIZER AREA	H	C	B	B	B	B	
	129'-0"	1T306	DETECTOR DRYER CAB. AREA	B	C	B	B	B	H	
	133'-0"	1T307	RFP TURBINE ROOM	E	E	E	B	B	C	
	133'-0"	1T308	MOTOR CONTROL CENTER AREA	B	E	B	B	D	D	
	133'-0"	1T309	H-P TURBINE ROOM	E	E	E	D	B	E	SEE NOTE 4
	136'-9"	1T310	PLATFORM - STEAM PIPE AREA	E	E	E	B	H	E	
	133'-0"	1T312	BATTERY "D" ROOM	B	E	B	B	B	B	
	133'-0"	1T313	TVR / GAC ROOM	B	E	B	B	B	H	
	133'-6"	1T314	CONDENSATE DEMIN. ROOM	H	C	E	E	H	E	SEE NOTE 13
	133'-0"	1T315	H.P. HEATER ROOM "A" TRAIN	E	E	E	B	E	E	SEE NOTE 4
	133'-0"	1T316	RAILROAD AREA	B	E	B	B	B	B	
	133'-0"	1T317	H.P. HEATER ROOM "B" TRAIN	E	E	E	B	E	E	SEE NOTE 4
	133'-0"	1T318	PUMP REMOVAL AREA	C	C	C	B	B	C	
	133'-0"	1T319	PRECOAT FILTER & PUMP ROOM	E	E	E	E	B	E	SEE NOTE 8
	139'-6"	1T320	CONDENSATE DEMIN. ROOM	E	E	E	E	B	E	SEE NOTE 13
	139'-6"	1T321	PASSAGE	B	C	B	B	B	B	
	133'-0"	1T322	GENERATOR AREA	B	E	B	B	B	B	
	133'-0"	1T323	SWITCHGEAR ROOM	B	E	B	B	B	D	
	133'-0"	1T324	MOTOR CONTROL CENTER AREA	B	E	B	B	B	B	
	133'-0"	1T325	FILTER INSTR. RACK AREA	B	E	B	B	B	H	
	147'-0"	1T326	MAIN STEAM TUNNEL	E	E	E	E	E	E	
	133'-0"	1T327	STEAM INSTR. RACK AREA	B	E	B	E	H	B	
	129'-6"	1T328	UNASSIGNED AREA	E	E	E	B	B	E	
AUX. BLDG. UNIT - 1	139'-0"	1A301	CORRIDOR	B	E	B	B	B	B	
	139'-0"	1A302	CORRIDOR	B	E	B	B	H	B	
	139'-0"	1A303	RHR HEAT EXCH. "A" RM.	C	E	E	E	E	E	SEE NOTES 13 & 14
	139'-0"	1A304	PIPE PENETRATION ROOM	E	E	E	E	B	E	
	148'-0"	1A305	MAIN STEAM AREA	E	E	E	B	B	E	

BLDG.	FLOOR EL.	ROOM NO.	SPACE	N	LOCA	HS	R	M	ST	REMARKS
AUXILIARY BLDG. UNIT - 1	139'-0"	1A306	PIPE PENETRATION ROOM	F	E	E	E	B	F	
	139'-0"	1A307	RHR HEAT EXCHANGER "B" RM.	C	E	E	E	B	C	SEE NOTES 13 & 14
	139'-0"	1A308	ELECTRICAL PEN. ROOM	B	E	B	B	B	B	
	139'-0"	1A309	ELECTRICAL PEN. ROOM	D	E	B	B	B	B	
	140'-0"	1A310	MAIN STEAM PIPE TUNNEL	E	E	F	R	R	E	
	135'-4"	1A311	CRD HYDRAULIC CONTROL AREA	B	E	H	H	H	B	
	135'-4"	1A313	CRD HYDRAULIC CONTROL UNIT AREA	B	E	B	D	D	B	
	139'-0"	1A314	PASSAGE	B	F	B	B	B	B	
	139'-0"	1A316	MOTOR CONTROL CENTER AREA	B	E	B	B	B	B	
	139'-0"	1A318	ELECTRICAL PEN. ROOM	B	F	B	B	B	B	
	139'-0"	1A319	RPV INSTR. TEST RM.	B	E	B	B	B	B	
	139'-0"	1A320	ELECTRICAL PEN. ROOM	B	E	B	B	B	B	
	139'-0"	1A321	MCC AREA	D	E	B	B	B	B	
	139'-0"	1A322	CENTRIFUGAL CHILLER AREA	B	E	B	B	B	B	SEE NOTE 19, DWG. A-0552
	139'-0"	1A323	STAND BY GAS TREATMENT ROOM	B	E	B	B	B	B	
	154'-0"	1A324	ROOF	B	E	B	D	B	D	
	133'-0"	1A325	RAILROAD AREA	B	E	B	D	B	B	
	139'-0"	1A326	STAND BY GAS TREATMENT ROOM	B	E	B	B	B	B	
DIESEL BLDG. UNIT - 1	133'-0"	1D301	CORRIDOR	A	C	A	A	A	A	
	133'-0"	1D302	DAY TANK ROOM	A	E	A	A	A	A	
	133'-0"	1D303	DAY TANK ROOM	A	F	A	A	A	A	
	133'-0"	1D304	DAY TANK ROOM	A	E	A	A	A	A	
	133'-0"	1D305	RELAY & CONTROL PANEL AREA	A	E	A	A	A	A	
	133'-0"	1D306	RELAY & CONTROL PANEL AREA	A	E	A	A	A	A	
	133'-0"	1D307	RELAY & CONTROL PANEL AREA	A	E	A	A	A	A	
WTR. TRTMT. BLDG.	133'-0"	0M110	CONFERENCE ROOM	A	E	A	A	A	A	
	133'-0"	0M111	COUNT ROOM	A	E	A	A	A	A	
	133'-0"	0M112	CHEMISTRY ROOM	A	E	A	A	A	A	
	133'-0"	0M113	OFFICE	A	E	A	A	A	A	
	133'-0"	0M114	STORAGE ROOM	A	E	A	A	A	A	
	133'-0"	0M113A	CORRIDOR	A	E	A	A	A	A	
	145'-0"	0M117	OFFICE	A	C	A	A	A	A	
	145'-0"	0M121	CHEMICAL STORAGE AREA	A	E	A	A	A	A	
	145'-0"	0M122	LAB ROOF	A	F	A	A	A	A	
	145'-0"	0M124	INSOLUBLES ROOM	A	E	A	A	A	A	
	133'-0"	0M125	CORRIDOR	A	E	A	A	A	A	
CONTROL BLDG.	133'-0"	0C301	PASSAGE	A	E	A	A	A	A	
	133'-0"	0C302	HVAC EQUIP. ROOM	A	E	A	A	A	A	
	133'-0"	0C303	HVAC EQUIP. ROOM	A	E	A	A	A	A	
	133'-0"	0C304	ELEC. SPACE	A	E	A	A	A	A	
	133'-0"	0C305	ELEC. SPACE	A	F	A	A	A	A	
	133'-0"	0C306	ELEC. SPACE	A	E	A	A	A	A	
	133'-0"	0C307	ELEC. SPACE	A	E	A	A	A	A	
	133'-0"	0C308	CORRIDOR	A	E	A	A	A	A	
	133'-0"	0C309	LOBBY	A	C	A	A	A	A	
RADWASTE BLDG.	136'-0"	0R301	VALVE ROOM	E	E	E	E	C	E	
	136'-0"	0R302	FILTER AREA	E	E	E	E	C	E	
	133'-0"	0R303	RAILROAD AREA	B	E	B	B	B	B	
	130'-0"	0R304	PASSAGE	B	E	B	B	B	B	
	136'-0"	0R305	LOAD CENTER AREA	B	E	B	B	B	B	
	136'-0"	0R306	CORRIDOR	B	E	B	H	B	B	
	122'-0"	0R307	VALVE RM. UNIT 1	E	E	E	B	B	E	
	122'-0"	0R308	VALVE RM. UNIT 2	E	E	E	B	B	E	
	136'-0"	0R309	HVAC EQUIP. ROOM	C	E	C	C	C	B	

LEGEND

N - NORMAL PLANT OPERATION @ POWER
LOCA - LOSS OF COOLANT ACCIDENT
HS - HOT STANDBY
R - REFUELING
M - MAINTENANCE OR COMPONENT TESTING
ST - SYSTEMS TESTING

LEGEND

ZONE	SYMBOL	MAXIMUM DOSE RATE (MREM / HR)	ACCESS DESCRIPTION
A		≤ 0.5	UNCONTROLLED, UNLIMITED ACCESS
B		≤ 2.5	CONTROLLED, LIMITED ACCESS, 40 HR WK
C		≤ 15	CONTROLLED, LIMITED ACCESS, 6 HR WK
D		≤ 100	CONTROLLED, LIMITED ACCESS, 1 HR WK
E		> 100	NORMALLY INACCESSIBLE ACCESS DURING EMERGENCIES

NOTES

1. ZONES CORRESPOND TO OPERATIONAL DESIGN BASIS RADIATION LEVELS FOR THE REACTOR COOLANT. THIS IS CONSIDERED AN ANTICIPATED OPERATIONAL OCCURRENCE.
2. ZONES INDICATED IN ROOMS AS SHOWN ON DRAWING ARE DURING NORMAL PLANT OPERATION. ZONES WILL BE HIGHER WHEN EQUIPMENT IN ROOM OPERATING.
3. ZONES INDICATED AREAS AS SHOWN ON DRAWING ARE DURING NORMAL PLANT OPERATION. ZONES WILL BE HIGHER WHEN AREA IS IN USE.
4. ZONE IS "E" (INACCESSIBLE) DURING NORMAL OPERATION. CONCRETE WALL PROVIDES SHIELDING FROM REDUNDANT COMPONENT SO THAT MAINTENANCE MAY BE CARRIED OUT DURING NORMAL PLANT OPERATION. DURING MAINTENANCE PERIOD, ZONE WILL BE "B".
5. LOCKS TO PROHIBIT ACCESS ARE REQUIRED FOR ALL ZONE "E" AREAS.
6. ZONE IN THIS AREA MAY BE HIGHER WHEN RHR AND RCIC EQUIPMENT IS OPERATING AT HOT STANDBY OR JUST AFTER SHUTDOWN.
7. ZONE MAY BE "C" DURING STARTUP WHEN MAIN IS BYPASSED.
8. ZONES WILL BE HIGHER AT PLANT STARTUP WHEN FILTER IS IN OPERATION, BUT DURING NORMAL PLANT OPERATION, ACTIVITY LEVELS WILL DECAY AND ZONE WILL DROP.
9. IN IMMEDIATE VICINITY OF RADIOACTIVE COMPONENT BEING WORKED ON, ZONE MAY EXCEED "A".
10. ZONE IS "C" DURING NORMAL OPERATION. CONCRETE WALL PROVIDES SHIELDING FROM REDUNDANT COMPONENT SO THAT MAINTENANCE CAN BE CARRIED OUT DURING NORMAL PLANT OPERATION. DURING MAINTENANCE PERIOD ZONE WILL BE "B".
11. ZONE IS "E" (INACCESSIBLE) DURING NORMAL OPERATION. CONCRETE WALL PROVIDES SHIELDING FROM REDUNDANT COMPONENT SO THAT MAINTENANCE CAN BE CARRIED OUT DURING NORMAL PLANT OPERATION. CONCRETE WALL WILL REDUCE DOSE FROM REDUNDANT COMPONENT TO ZONE "B". HOWEVER, COMPONENT BEING SERVICED SHOULD BE THOROUGHLY DECONTAMINATED SO AS TO REDUCE DOSE FROM IT TO LOW LEVELS.
12. ZONE IS "A" IN THIS AREA DURING NORMAL OPERATION. WHEN FUEL IS STORED IN CONTAINMENT FUEL POOL, ZONE WILL BE "E".
13. ZONE IS "E" (INACCESSIBLE) WHEN COMPONENT IS RUNNING. CONCRETE WALL PROVIDES SHIELDING FROM OTHER COMPONENTS IN THE AREA SO THAT MAINTENANCE CAN BE CARRIED OUT DURING NORMAL PLANT OPERATION. CONCRETE WALL WILL REDUCE DOSE FROM OTHER COMPONENTS TO ZONE "B". HOWEVER COMPONENT BEING SERVICED SHOULD BE CHECKED TO DETERMINE DOSE LEVELS FROM IT.
14. IN THE INITIAL STAGE OF ACCIDENT THE ZONE WILL BE "E". HOWEVER WITH DECAY TIME THE RADIATION LEVEL WILL DECREASE TO A LOWER ZONE.
15. IN ACCORDANCE WITH GENERAL DESIGN CRITERION 19 OF 10 CFR 50, THE RADIATION EXPOSURE IN THE CONTROL ROOM WILL NOT EXCEED 5 REM IN ONE MONTH FOLLOWING A LOCA.
16. IN ORDER TO OBTAIN A ZONE "B" FOR MAINTENANCE ON RADIOACTIVE EQUIPMENT, SUFFICIENT DECAY TIME, THOROUGH FLUSHING & PROPER DECONTAMINATION PROCEDURES MUST BE FOLLOWED.
17. THE AREA IN FRONT OF THE DRYWELL PERSONNEL LOCK AND EQUIPMENT HATCH WILL BE "E" DURING NORMAL OPERATION. WHEN THE REACTOR IS SHUTDOWN, THE ZONE IS "B".
18. FOR INFORMATION ON POST-ACCIDENT RADIATION ZONES, SEE DWGS. A-0570 THROUGH A-0575.
19. ALL NON-ACCIDENT DOSE RATES IDENTIFIED ON THIS DRAWING ARE PRE-OPERATIONAL ESTIMATES AND ARE NOT UPDATED BASED ON ACTUAL PLANT RADIOLOGICAL CONDITIONS. THIS INFORMATION SHOULD BE CONSIDERED HISTORICAL.

PRIORITY DRAWING

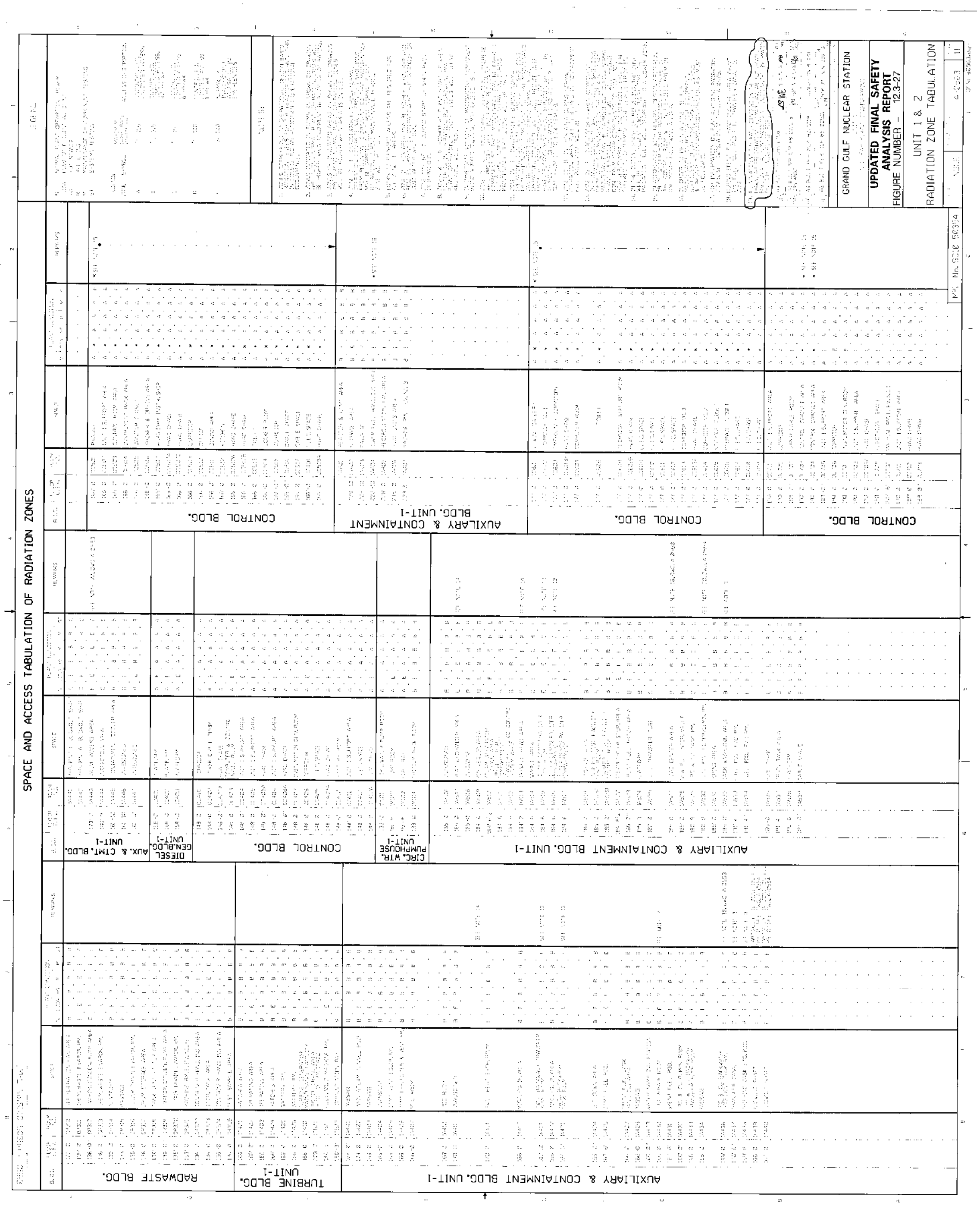
AS-BUILT DCP 82/0754
AS BUILT PER DRN 6757
REDRAWN IN NEW FORMAT. AS-BUILT PER DCP 89/0001
SIGN-OFF ON RECORD

GRAND GULF NUCLEAR STATION
UNIT 1
NUCLEAR PLANT ENGINEERING

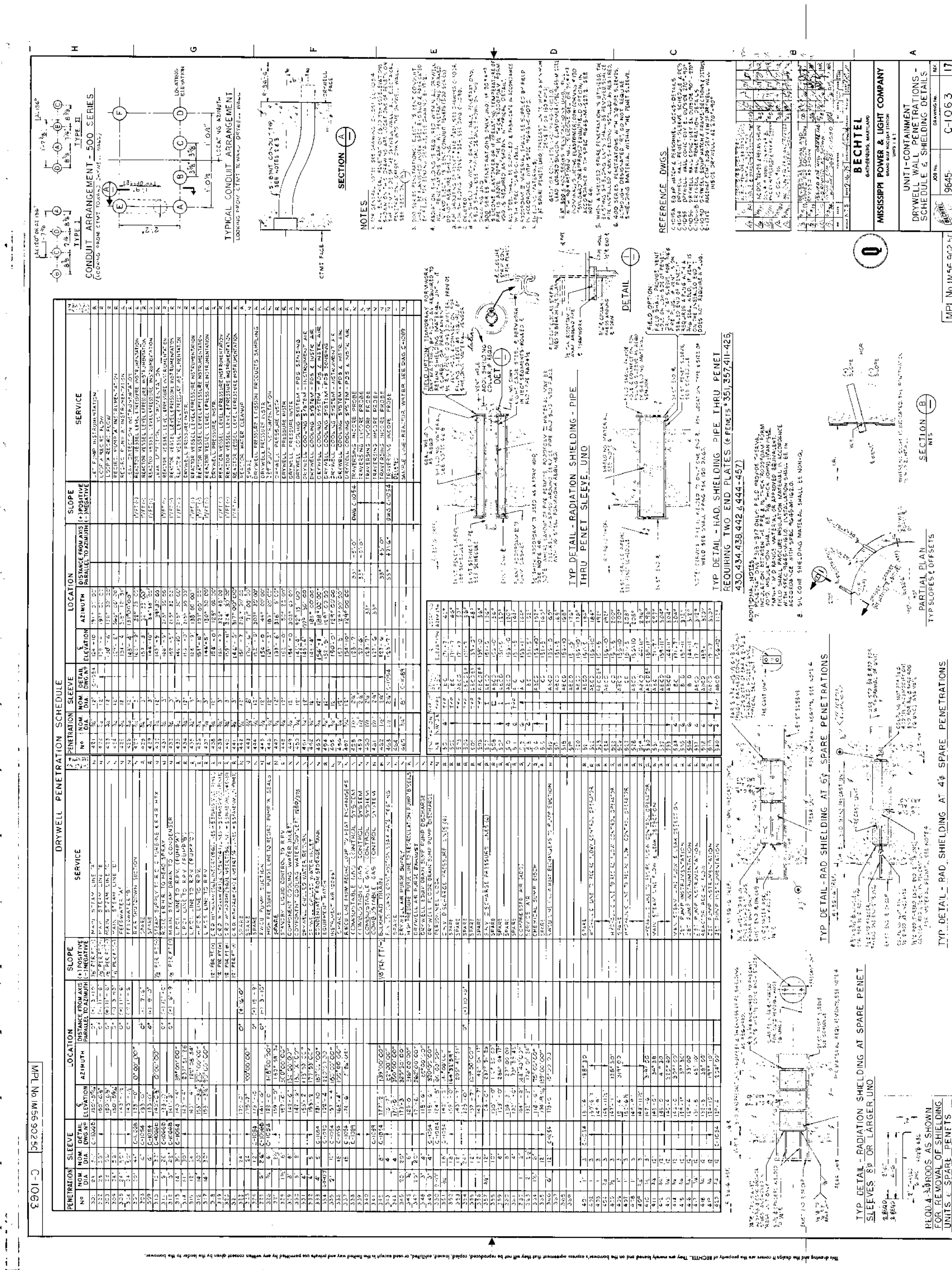
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-026

RADIATION ZONE TABULATION

COMMON - UNIT 1 CONTROLLED
MPL NO. S010 5025A
SCALE: NONE
DRAWING NO. A-0562
REV. 12
LDC 89-043
DPN NO. A0562.dgn



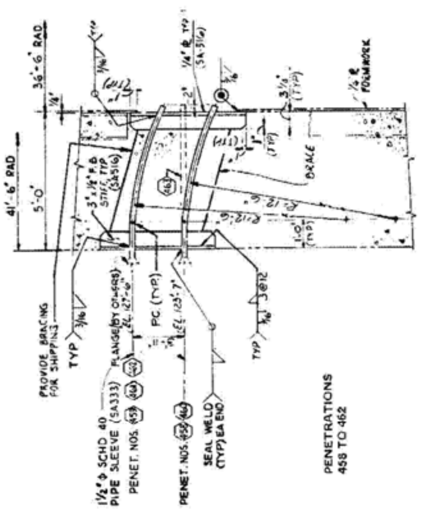
SPACE AND ACCESS TABULATION OF RADIATION ZONES
CONTROL BLDG.
AUXILIARY & CONTAINMENT BLDG. UNIT-1
AUX. & CTMT. BLDG. UNIT-1
DIESEL GEN.BLDG. UNIT-1
CIRC. WTR. PUMPHOUSE UNIT-1
RADWASTE BLDG.
TURBINE BLDG. UNIT-1
GRAND GULF NUCLEAR STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE NUMBER - 12.3-27
UNIT 1 & 2
RADIATION ZONE TABULATION

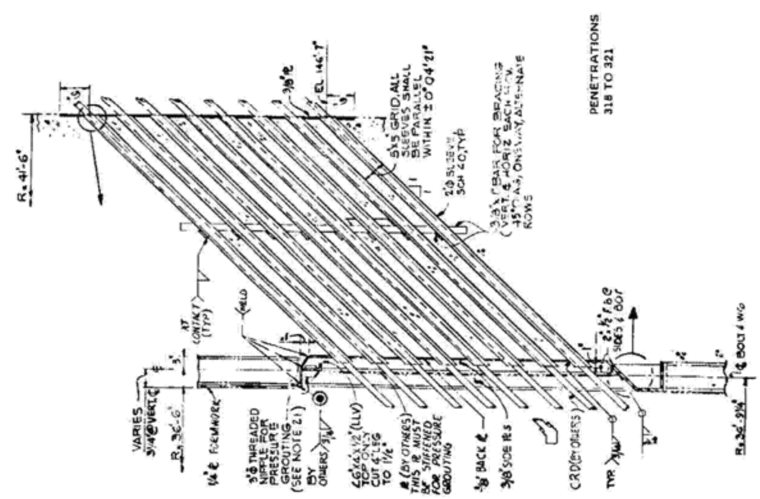

DRYWELL PENETRATION SCHEDULE
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
C-1063

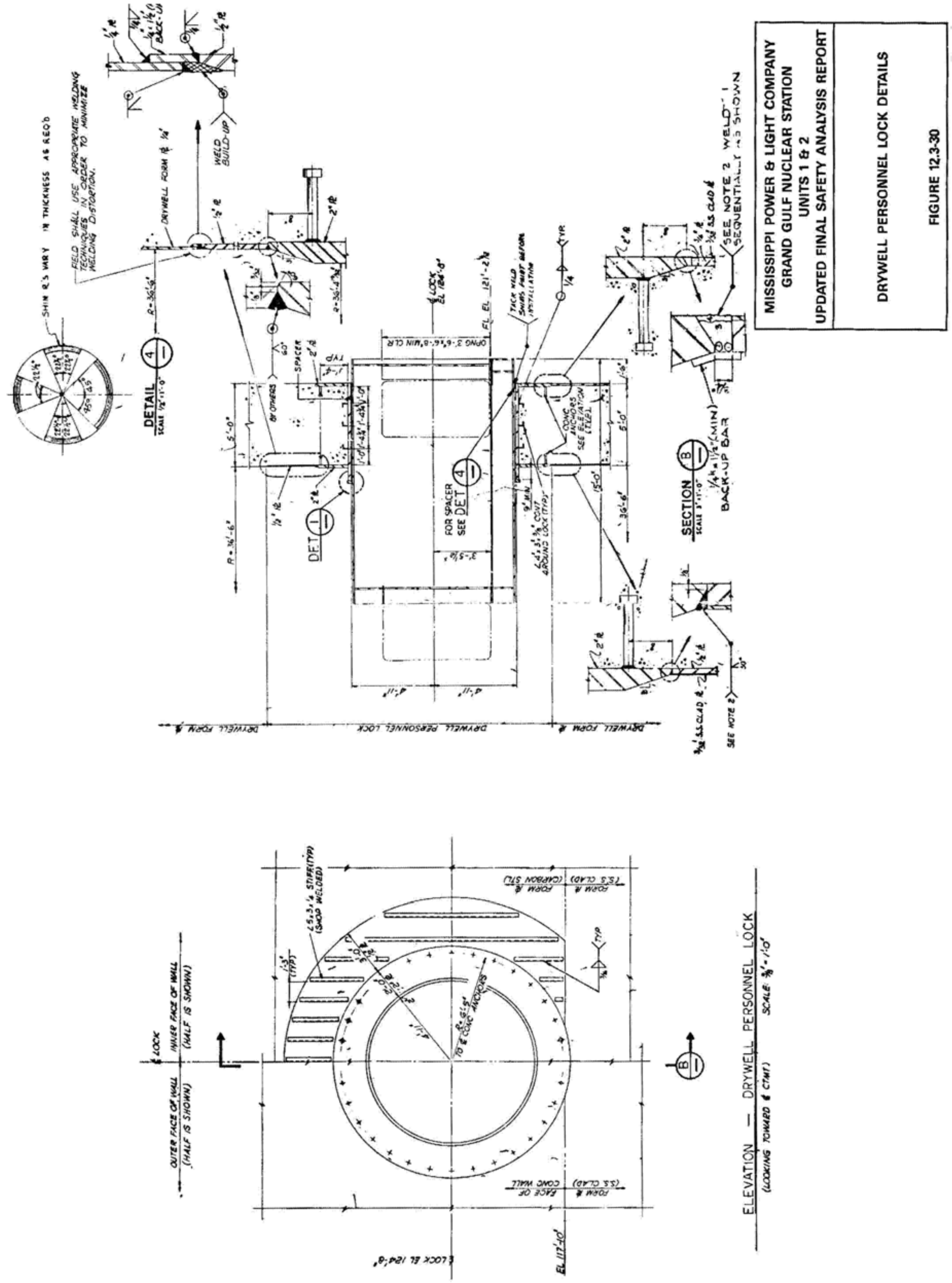
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT

DRYWELL WALL PENETRATION DETAILS

FIGURE 12.3-29






MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
UPDATED FINAL SAFETY ANALYSIS REPORT
DRYWELL PERSONNEL LOCK DETAILS
FIGURE 12.3-30
SECTION B
DETAIL 4
ELEVATION — DRYWELL PERSONNEL LOCK

12.4 DOSE ASSESSMENT

As noted in UFSAR Section 1.1.1, after Unit 1 had received its Commercial Operating License, Entergy Operations, Inc. formally requested the NRC to return the Construction Permit and officially cancel the second unit at Grand Gulf Nuclear Station. However, some references to unit 2 still appear in UFSAR Section 12.4 and associated tables to reflect the fact that Section 12.4 text and analyses were prepared assuming that two Grand Gulf units would be in operation. Per Reference 4, these analyses will be updated during future plant modifications, if required.

Radiation exposures in the plant are primarily due to direct radiation from components and equipment containing radioactive fluids. In addition, in some plant radiation areas, there can be radiation exposure to personnel due to the presence of airborne radionuclides. In-plant radiation exposures during normal operation and anticipated operational occurrences are discussed in subsection 12.4.1. Radiation exposures at other onsite locations outside the plant which arise from onsite radioactive sources, the presence of N-16 in the plant, and radioactive gaseous effluents, are discussed in subsection 12.4.2.

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

12.4.1 Exposures Within the Plant

12.4.1.1 Direct Radiation Dose Estimates

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

a. [HISTORICAL INFORMATION] [Radiation exposure data from operating BWRs are given in Tables 12.4-1 and 12.4-2.]

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

c. Expected occupancy times for various work function personnel in the different plant radiation areas are listed in Table 12.4-3.

d. Anticipated personnel requirements for the Grand Gulf Nuclear Station (per unit basis) are given in Table 12.4-3.

The maximum and expected average dose rates in plant radiation areas are given below:

Zone	Maximum Dose Rate (mrem/hr)	Expected Average Dose Rate (mrem/hr)
A	0.5	0.2
B	2.5	1
C	15	6
D	100	40
E	>100	100

Maximum dose rates have been determined by shielding calculations based on conservative assumptions regarding the source terms, self-shielding of the sources, and location of the occupational worker vis-a-vis the radioactive source, etc. For example, the calculations were based on maximum coolant and steam N-16 concentrations of 4.8E+01 and 2.5E+02 µCi/gm respectively, steam noble gas concentrations corresponding to an offgas rate of 100,000 µCi/sec after a 30-minute delay, and iodine coolant concentrations corresponding to a release rate of 700 µCi/sec for I-131 from the fuel. Only under such extreme conditions would the dose rates possibly correspond to the maximum dose rates.

An EPU evaluation of expected average dose rates in specific plant radiation areas (Zones B, C and E) provided calculated anticipated dose rates. These specific calculated dose rates are provided on Table 12.4-12. Although the Radiations Zones maximum dose rates are not exceeded, the EPU calculation provides clarification of the impact of increasing thermal power levels.

The expected dose rates have been estimated using the following bases:

a. Realistic specific activities of radioisotopes in the coolant and steam are provided in Table 2-2 of Ref. 2 after appropriate adjustments for plant parameters.

b. Adoption of stringent procedures for water chemistry control and design improvements that will minimize crud buildup.

[HISTORICAL INFORMATION] [The expected average dose rates have been used in estimating doses for all routine functions with one exception: for personnel involved in the performance of inservice inspection (ISI), an expected average dose rate of 200 mR/hr in the E Zone is used since these personnel will generally be working exclusively on reactor coolant system components.

Direct radiation exposures to plant personnel can also result from the performance of special maintenance functions. In view of the radiation protection design features described in Section 12.3 and the radiation protection program outlined in Section 12.5, it is expected that exposures due to special maintenance will be minimal. However, an exposure of 150 person-rem/unit is realistically assumed based on experience at operating BWRs.

Exposure to plant personnel from direct radiation during the performance of routine functions is estimated to be approximately 300 person-rem/yr/unit. Details of the person-rem estimates are given in Table 12.4-3. A breakdown of the exposures (including the special maintenance category) by work functions is provided in Table 12.4-4.]

12.4.1.2 Exposures Due to Airborne Radioactivity

Generally, localized leakage of liquids and gases resulting in airborne radioactivity is expected only in D and E Zones; these zones contain equipment and components that handle highly radioactive fluids, and are normally inaccessible. Also, the ventilation systems in the auxiliary, turbine, and radwaste buildings are designed in such a manner that airborne contamination from high-radiation zones will not generally spread into low-radiation zones. Furthermore, operating procedures described in subsections 12.3.3, 12.3.4, and 12.5.3 are intended to minimize exposure to airborne activity. Consequently, negligible airborne radiation exposures are expected in those areas of these buildings which are accessible.

The existing radiation zoning design (e.g., the maximum designed dose rates for each area of the plant, Table 12.4-12) will not change as a result of the increased dose rates associated with the EPU increased power levels. The EPU radiation levels in various areas of the plant were estimated by multiplying the previously existing measured dose rates by the expected increase in radiation levels following implementation of the EPU increased power levels for the respective plant areas as shown on Table 12.4-12.

There are, however, a few plant radiation areas where there can be exposures due to airborne contamination. These are: a) the containment, b) spent fuel pool area in the auxiliary building, and c) the liquid radwaste pump rooms in the radwaste building. Airborne radioactivity can arise from coolant and steam leakage into the containment and also from the escape of radioactivity from the suppression pool (radioactive steam is assumed to leak continuously from the safety/relief (S/R) valves to the suppression pool) during normal operation. During refueling, radioactivity (mostly iodine isotopes) can escape from the refueling pool. The airborne concentration during refueling is, however, expected to be lower than it would be during normal power operation due to lower releases and a higher purge rate. Also, the refueling pool will be continually cleaned by the fuel pool cooling and cleanup system. Airborne radioactivity in the spent fuel pool (SFP) area of the auxiliary building during refueling is expected to be low, since there will be purging of the atmosphere and continuous cleanup of the SFP water. Most of the pump rooms in the radwaste building are D Zones and there will be some leakage of radioactivity from pump seals, etc. Occupancy in these regions, however, is expected to be very limited. Realistic estimates of airborne concentrations of radionuclides in the containment, refueling area of the auxiliary building, and the pump rooms of the radwaste building are provided in Table 12.2-18. Annual occupancy (person-hours), dose rates, and person-rem due to airborne radioactivity in these areas are given in Table 12.4-5.[HISTORICAL INFORMATION] [Annual occupancy for various locations in the containment are given in Table 12.4-11 based on GEH estimates derived from experience of previous Mark III containment’s.

Exposure to airborne radioactivity can also result from actuation of safety/relief valves either at full power (Type 1) or after an isolation scram (Type 2). These anticipated occurrences will result in steam releases to the suppression pool and escape of

radioactivity from the pool to the containment atmosphere. Type 1 and Type 2 actuations are estimated to occur twice and once a year, respectively. The Type 2 occurrence is expected to be limiting in terms of its radiological consequences because of the length of time steam is discharged to the suppression pool. Type 1 is associated with momentary pressure relief.

Operator doses for the more severe Type 2 occurrence have been estimated using the following bases:

a. Source terms for noble gases, iodine, and other radioisotopes given in Tables 12.4-6 and 12.4-7.

b. Four minutes egress time for an operator to get out of the contaminated area. This includes one minute to open the airlock door. A clean-air shower provided in the region of the airlock is expected to reduce the egress doses due to iodine inhalation and betas by roughly 40 percent; however, no dose reduction factor is applied in the calculations.

c. Homogeneous mixing of airborne contaminants in the containment annulus (CRD System area, TIP drive area, etc.) surrounding the drywell (approximately 300,000 cubic feet) which is approximately one-fifth the entire containment free volume.

d. Containment occupancy times during normal operations are provided in Table 12.4-11 based on GEH derived experience with previous Mark III containments.

e. Containment airborne concentrations are not corrected for plate out on walls, as it will be negligible in the first few minutes.

f. An average radiohalogen carry-over of 2.0 percent by weight.

Estimates of exposures resulting from a Type 2 occurrence are provided in Table 12.4-8. Using only the annulus volume in conjunction with the TIP station location directly above the suppression pool maximizes the whole body dose. The whole body dose effects are determined by dividing the annular area into a very large number of differential volume elements and approximating the activity in the volume element by a point source. The total dose effect is, therefore, the sum of the

individual dose effects and takes into consideration the energy spectrum associated with the released activity, as well as attenuation between the point source and the dose recipient. The thyroid and lung dose rate via the inhalation pathway was calculated using the model and dose factors given in References 4 and 7. These doses are primarily dependent upon the activity in the immediate vicinity of the dose recipient and, therefore, are inversely proportional to the assumed mixing volume. For example, if the entire containment free volume was assumed available for mixing, these doses would be decreased by almost a factor of five. Conversely, if a smaller mixing volume is assumed, the skin and thyroid doses would be increased. The whole body annual submersion dose rates have been estimated using the whole body dose factors for radioisotopes (Refs. 4 and 5) and a finite cloud model.] The skin and thyroid doses are based on those models specified in Regulatory Guide 1.3.

12.4.1.3 Illustrative Examples of Dose Assessment

Dose assessments for various operations were based on actual operating plant data. A number of typical examples are provided below. Note that the maximum/minimum values for number of personnel do not correspond to the maximum/minimum number of days required or dose rates so that the person-rem totals are not simple multiplications of the maximum/minimum factors.

[HISTORICAL INFORMATION] [The dose assessments in Tables 12.4-3 and 12.4-4 were derived from the average number of personnel, average length of time, and average dose rate to perform each particular operation.

a. Operation and Surveillance

   The dose rates in the corridors and other normally occupied areas are expected to be much lower than the maximum radiation levels for each zone shown in the radiation zone drawings] (Figures 12.3-12 through 12.3-17.) [HISTORICAL INFORMATION] [The expected radiation levels are provided in subsection 12.4.1.1. Based on these expected dose rates and typical time periods for operation and surveillance, exposures were calculated.

b. Normal Maintenance

An example of normal maintenance would be work on control rod drives. From operating plant data this work has taken from 10 to 45 days, using 7 to 12 workers accumulating 50-190 mrem/day. Total exposures have been 17 to 80 person-rem. Main steam isolation valve repairs have taken from 10 to 65 days using 6 to 9 workers at about 100 mrem/day. Total exposures have ranged from 7 to 60 person-rem.

c. Radwaste Handling

Annual exposures for radwaste handling were determined as a result of the system evaluations described in subsection 12.1.2.4.b.1. Average expected dose rates and personnel stay times were used.

d. Refueling

Based on operating plant data, refueling operations have required 15 to 30 days, using 8 to 30 personnel who receive from 20 to 40 mrem/day. Person-rem totals have ranged from 10 to 32.

e. Inservice Inspection

Inservice inspections at operating BWRs have taken from 10 to 90 days, requiring from 3 to 12 personnel at 4 to 190 mrem/day. Total exposures have ranged from 10 to 100 person-rem.

f. Special Maintenance

Special Maintenance is generally of a non-recurring nature and not readily predictable. It includes implementation of design changes and unexpected repair or replacement of equipment and components.

Designs are continually being improved so that the newer plants should not experience all of the problems that have occurred on operating plants. Some special maintenance has created greater than 150 person-rems of exposure but the frequency of occurrence is irregular. An estimated annual average exposure from special maintenance work is represented by 150 person-rems, but this amount may be exceeded in any 1 year.

One example of special maintenance at BWR operating plants is core spray sparger replacement. Typically, this has taken one month with an average of 40 to 70 personnel/day at a dose rate of 230 to 360 mrem/day. Total exposure has been in the range of 280 to 660 person-rem. An activity such as this might occur only once in the life of a plant.

Other special maintenance such as valve operator or pump impeller replacement might occur several times in the life of the plant but exposures would be much less.]

12.4.2 Exposures at Locations Outside the Plant Structures

Radiation exposures at the nearest site boundary will arise from: a) onsite radioactive sources outside plant buildings, b) sky shine due to the presence of N-16 in the plant buildings, and c) release of gaseous effluents from the plant. The sky shine dose due to N-16 is expected to be the predominant contributor. Direct shine associated with EPU increased power levels was evaluated and contributed to the direct radiation (including skyshine) from contained radioactive sources within the facility. Table 12.4-13 contains the EPU estimated doses to members of the public due to normal operation at EPU increased power levels for both gaseous and liquid radwaste effluents. Table 12.4-9 provides the estimated skyshine doses at EPU increased power levels.

12.4.2.1 Direct Radiation Dose Estimates

The only onsite source that is significant in terms of direct radiation is the condensate water storage tank (CWST). Exposures due to direct radiation from this tank have been evaluated using the following assumptions and parameters:

a. The Unit 1 CWST is shielded from the nearest site boundary by the radwaste and water treatment buildings (Figure 2.1-1). The planned site of the Unit 2 CWST (per UFSAR Section 1.1.1, Unit 2 has been canceled) was approximately 2,780 feet from the nearest site boundary on the east side of the plant. Even though this boundary may have been well shielded by the bluff on the east side of the plant, the earth shielding was conservatively ignored.

b. The CWST capacity is 300,000 gallons; all of this volume is conservatively assumed to be at a concentration of 1.8 x 10^{-4} µCi/cc.

c. A tank is considered as a point source emitting gamma energy of 0.8 MeV/disintegration.

d. Only shielding by air is considered (shielding by the 3/8 inch thick tank wall and self-attenuation by water in the CWST are conservatively ignored). Buildup in the air has been included.

e. Annual occupancy of 4,380 hours is assumed at the nearest site boundary location based on a yearly occupancy fraction of 0.5.

The dose at the nearest site boundary location has been estimated to be 5.04 x 10^{-3} mr/yr.

12.4.2.2 N-16 Skyshine Doses

Skyshine doses are due to air scattering of the high-energy gammas emitted by decaying N-16 present in reactor steam in the main steam lines, turbines, and moisture separators. These doses have been evaluated by using the program SKYSHINE (Ref. 6). The layout of the turbine building walls and floors used in the dose evaluation is given in the general arrangement drawings, Figures 1.2-9 A, B, C. Data regarding the source term and the major shields used in the dose evaluation are listed below:

a. Specific activity of N-16 in the reactor steam is 2.5E+02 µCi/gm (see Section 11.1 and Table 11.1-4).

b. A 2-foot thick concrete floor is located above the moisture separator-reheaters (El. 203'-0").

c. There are 4-1/2-foot-thick concrete walls below the above floors.

d. A 3-inch-thick steel plate is located between the turbine and generator.

An operating load factor of 80 percent was used to estimate the doses at various outside locations given in Table 12.4-9. The dose point distances are measured from the intersection of the turbine-generator axis and the wall between the Unit 1 and Unit 2 turbine buildings.

12.4.2.3 Exposures Due to Airborne Activity

Doses at the site boundary due to released airborne radioactivity are given in Section 11.3.

12.4.3 Deleted

12.4.4 References

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

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

3. NUREG-0016, Calculation of Releases of Radioactive Materials In Gaseous And Liquid Effluents From Boiling Water Reactors, April 1976.

4. NRC Regulatory Guide 1.109, Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents For the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I, March 1976.

5. ORNL-5114, Nuclear Decay Data for Selected Radionuclides, March 1976.

6. Collins, D. G., et al, "Skyshine, A Computer Procedure for Evaluating Effect of Structure Design on N-16 Gamma Ray Dose Rates."

7. NRC Regulatory Guide 1.3, The Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.

8. Letter from W. T. Cottle to NRC Document Control Desk, GNRO-91/00148, August 15, 1991, Subject: Schedule for UFSAR Changes Reflecting Termination of Construction Permit No. CPPR-119 for GGNS Unit 2.

TABLE 12.4-1: IN-PLANT EXPOSURE DATA FROM OPERATING BOILING WATER REACTORS

[HISTORICAL INFORMATION]

Year	No. of Units	Name of Reactor	Design Power Level (Mwe)	Megawatt year	Annual Exposures (person-rem/yr Unit)
1971	4	Dresden 1,2	200,809	395	715
		Nine Mile Point	610	347	195
		Oyster Creek	650	449	240
Average per Unit					288
1972	7	Dresden 1,2,3	200, 809 809	1244	728
		Millstone Point 1	690	378	596
		Monticello	545	424	61
		Nine Mile Point	610	382	285
		Oyster Creek	650	515	582
Average per Unit					322
1973	7	Dresden 1,2,3	200,809 809	1112	909
		Millstone Point 1	690	225	620
		Monticello	545	390	154
		Nine Mile Point	610	411	517
		Oyster Creek	650	425	1236
Average per Unit					491

TABLE 12.4-1: IN-PLANT EXPOSURE DATA FROM OPERATING BOILING WATER REACTORS (Continued)

[HISTORICAL INFORMATION]

Year	No. of Units	Name of Reactor	Design Power Level (Mwe)	Megawatt year	Annual Exposures (person-rem/yr Unit)
1974	11	Dresden 1,2,3	200,809 809	843	1662
		Millstone Point 1	690	430	1430
		Monticello	545	349	349
		Nine Mile Point	610	386	824
		Oyster Creek	650	435	984
		Pilgrim	655	234	415
		Quad Cities 1 & 2	809,809	958	4385
		Vermont Yankee	514	304	216
Average per Unit					578
1974	11	Dresden 1,2,3	200,809 809	708	3209
		Millstone Point 1	690	465	2022
		Monticello	545	345	1353
		Nine Mile Point	610	359	681
		Oyster Creek	650	374	1132
		Pilgrim	655	308	744
		Quad Cities 1 & 2	809,809	834	1385
		Vermont Yankee	514	429	139

TABLE 12.4-1: IN-PLANT EXPOSURE DATA FROM OPERATING BOILING WATER REACTORS (Continued)

[HISTORICAL INFORMATION]

Average per Unit	970
Overall average for the period 1971-1975	597

Notes:

a. The data are taken from Reference 1. Only BWRs of power levels greater than 500 MWe have been considered with one exception; Dresden Unit 1 (200 MWe) has been included since no exclusive data for Dresden Units 2 and 3 have been provided in Reference 1.

b. Only reactors that had been in commercial operation for at least 18 months at the end of the reporting year have been considered with one exception; Dresden Unit 3 has been included in the data for 1972 because only a plant total was available for Dresden.

TABLE 12.4-2: DISTRIBUTION OF PERSON-REM ACCORDING TO WORK FUNCTIONS

[HISTORICAL INFORMATION]

Work Function	1975 Percentage (Ref.1)	1974 Percentage (Ref.2)
Reactor operation	10.8	14
Normal maintenance	52.5	45
Radwaste handling	6.9	2.5
Refueling	7.8	14
Inservice inspection	3.0	2.7
Special maintenance	19.0	20

Notes: a. Percentages for the Reference 1 case are based on approximately 50 percent of the total exposures reported in 1975 for all light water reactors.

b. Percentages for the Reference 2 case are based on approximately 39 percent of the total exposures reported in 1974 for all light water reactors (note that this total does not add up to 100 percent; however, these numbers are in strict accordance with the percentages quoted in Reference 2).

TABLE 12.4-3: ESTIMATES OF OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA DOSES TO PLANT PERSONNEL

[HISTORICAL INFORMATION]

Operation	Zone	Percentage of Occupancy	Hrs/hr	Hourly Dose Rate (Rems/Hr)	Yearly Dose Rate (Rems/yr)	No. of Personnel	Annual Exposures (Person-Rem/yr Single Unit)
Operation & Surveillance	A	75	1,560	2×10^{-4}	0.312	30	9.4
	B	21	437	1×10^{-3}	0.437	30	13.1
	C	3	62	6×10^{-3}	0.372	30	11.2
	D	0.9	19	4×10^{-2}	0.760	30	22.8
	E	0.1	2	1×10^{-1}	0.200	30	6.0
Total		100	2,080				62.5
Normal Maintenance	A	75	1,560	2×10^{-4}	0.312	40	12.5
	B	11		1×10^{-3}	0.229	40	9.2
	C	12.5		6×10^{-3}	1.560	40	62.4
	D	1.0		4×10^{-2}	0.840	40	33.6
	E	0.5		1×10^{-1}	1.000	40	40.0
Total		100	2,080				157.7
Radwaste Handling	A	70	1,456	2×10^{-4}	0.2912	8	2.3
	B	20	416	1×10^{-3}	0.4160	8	3.3
	C	9	187	6×10^{-3}	1.1220	8	9.0
	D	0.5	11	4×10^{-2}	0.44	8	3.5

TABLE 12.4-3: ESTIMATES OF OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA DOSES TO PLANT PERSONNEL (Continued)

[HISTORICAL INFORMATION]

Operation	Zone	Percentage of Occupancy	Hrs/hr	Hourly Dose Rate (Rems/Hr)	Yearly Dose Rate (Rems/yr)	No. of Personnel	Annual Exposures (Person-Rem/yr Single Unit)
	E	0.5	10	1×10^{-1}	1.00	8	8.0
Total		100	2,080				26.1
Refueling	A	30	48	2×10^{-4}	0.0096	40	0.4
	B	40		1×10^{-3}	0.064	40	2.6
	C	24		6×10^{-3}	0.234	40	9.4
	D	4		4×10^{-2}	0.240	40	9.6
	E	2		1×10^{-1}	0.300	40	12.0
Total		100	160				34.0
Inservice Inspection (ISI)	A	55	132	2×10^{-4}	0.264	20	0.5
	B			1×10^{-3}	0.085	20	1.7
	C			6×10^{-3}	0.102	20	2.0
	D			4×10^{-2}	0.096	20	1.9
	E			1×10^{-1}	0.720	20	14.4
Total		100	240				20.5
Grand total for all operations						138	301

TABLE 12.4-4: DISTRIBUTION OF DIRECT RAIDATION PERSON-REM DOSES ACCORDING TO WORK FUNCTIONS FOR GRAND GULF NUCLEAR STATION

[HISTORICAL INFORMATION]

Operation	Annual Exposures (Person-rem/year Single Unit)	Percentage
Operation and surveillance	62.5	14
Maintenance*	307.7	68
Radwaste handling	26.1	6
Refueling	34.0	7.5
Inservice inspection	20.5	4.5
Total	451	100

*Includes an assumed exposure of 150 person-rem/yr unit during the performance of special maintenance functions.

TABLE 12.4-5: DOSES TO PLANT PERSONNEL CAUSED BY AIRBORNE RADIOACTIVITY

[HISTORICAL INFORMATION]

	Dose Rate (rem/hr)			Annual Occupancy (Person-Hr/Yr)						Annual Exposures (Person-rem/yr-Single Unit)		
Location	Thyroid	Lung	Total Body	Operation & Surveillance	Maintenance	Rad-waste Handling	Refueling	Inservice Inspection	Total	Thyroid	Lung	Total Body
Containment (power)	2×10^{-3}	1.12×10^{-4}	3.28×10^{-4}	5472	2249	---	---	---	7721	15.40	0.86	2.53
Containment (Refueling)	3.0×10^{-4}	N	N	---	2400	---	2400	4800	9600	2.88	N	N
Auxiliary Bldg. (Refueling Area)	8.4×10^{-6}	N	N	---	---	---	1600	---	1600	0.01	N	N
Pump rooms in the Radwaste Bldg*	3.4×10^{-3}	1.82×10^{-3}	N	---	300	---	---	---	300	1.02	0.55	N
									Total	19.31	1.41	2.53

* Dose rates are based on condensate demineralizer regeneration solution receiving pump room air concentrations.
N: Negligible

TABLE 12.4-6: SUMMARY OF ACTIVITY IN THE CONTAINMENT BUILDING WITH TIME AFTER ISOLATION SCRAM

[HISTORICAL INFORMATION]

	Activity in Curies			
	Time after Isolation Scram			
Nuclide	**4 min.**	**4 hr.**	**8 hr.**	**16 hr.**
Kr-83m	6.9 + 0	5.9 + 1	1.4 + 1	7.3 - 1
Kr-85m	1.5 + 1	1.4 + 2	7.7 + 1	2.3 + 1
Kr-85	2.7 - 2	5.3 - 1	5.4 - 1	5.7 - 1
Kr-87	2.5 + 1	5.0 + 1	5.7 + 0	7.5 - 2
Kr-88	4.2 + 1	2.7 + 2	1.0 + 2	1.5 + 1
Rb-88	2.3 + 0	2.8 + 2	1.1 + 2	1.6 + 1
Kr-89	5.0 + 0	---	---	---
Rb-89	4.2 - 1	---	---	---
Kr-90	1.7 - 2	---	---	---
Rb-90	3.2 - 2	---	---	---
1-131	4.8 - 5	7.0 - 2	7.3 - 2	7.1 - 2
Xe-131m	2.1 - 1	3.7 + 0	3.7 + 0	3.8 + 0
1-132	8.8 - 5	3.7 - 1	1.9 - 1	1.1 - 1
1-133	8.7 - 5	7.2 - 2	6.5 - 2	5.0 - 2
Xe-133m	1.8 + 0	3.1 + 1	3.0 + 1	2.8 + 1
Xe-133	8.1 + 1	1.4 + 3	1.4 + 3	1.4 + 3
1-134	4.5 - 4	4.1 - 2	1.8 - 3	3.3 - 6
1-135	1.5 - 4	5.0 - 2	3.4 - 2	1.5 - 2
Xe-135m	1.3 + 1	3.2 + 1	2.3 - 2	9.7 - 3
Xe-135	3.2 + 1	6.4 + 2	4.8 + 2	2.8 + 2
Xe-137	9.0 + 0	---	---	---
Xe-138	3.3 + 1	5.4 - 3	---	---
Cs-138	1.1 + 0	7.5 - 1	4.3 - 3	---
Xe-139	5.8 - 2	---	---	---
Total	2.7 + 2	2.9 + 3	2.2 + 3	1.8 + 3

Notes

1. This is a selected list of significant nuclides only.
2. The pressure relief transient is described as follows:

TABLE 12.4-6: SUMMARY OF ACTIVITY IN THE CONTAINMENT BUILDING WITH TIME AFTER ISOLATION SCRAM (Continued)

[HISTORICAL INFORMATION]

Time	Event
0 sec.	Turbine trip with 10 percent bypass following extended operation at fuel failure corresponding to an offgas release rate of 100,000 μCi/sec at 30 minutes reference time and halogen concentrations in reactor water being that shown in Table 11.1-2. Relief of reactor pressure to main condenser via 10 percent bypass and to the suppression pool via relief valves.
45 sec.	Main steam isolation valves close. Relief of reactor pressure to suppression pool via low low set relief valves only, following closure of others. Reactor in hot standby mode.
1 hr.	Begin reactor cooldown and depressurization via relief valve release to suppression pool.
4 hr.	Reactor depressurization complete.

TABLE 12.4-7: SOURCE STRENGTH IN CONTAINMENT BUILDING VENTILATION AIR WITH TIME AFTER ISOLATION SCRAM SUMMATION OF GAMMA ENERGIES

[HISTORICAL INFORMATION]

(MeV/sec)

Time After Scram	0.0-0.75 MeV	0.75-1.5 MeV	1.5-2.5 MeV	2.5-3.5 MeV	3.5-4.5 MeV	4.5-5.5 MeV	Total
1 min.	1.3 + 11	4.7 + 10	2.9 + 11	4.1 + 10	8.7 + 9	3.3 + 8	5.2 + 11
2 min.	5.3 + 11	1.8 + 11	1.2 + 12	1.6 + 11	2.6 + 10	1.1 + 9	2.1 + 12
4 min.	1.6 + 12	5.6 + 11	3.7 + 12	4.8 + 11	5.0 + 10	3.3 + 9	6.4 + 12
10 min.	3.7 + 12	1.6 + 12	9.2 + 12	1.2 + 12	3.8 + 10	1.8 + 10	1.6 + 13
20 min.	4.4 + 11	2.7 + 12	1.1 + 13	1.7 + 12	9.0 + 9	5.2 + 10	2.0 + 13
1 hr.	4.8 + 12	3.6 + 12	1.2 + 13	2.1 + 12	7.0 + 8	1.4 + 11	2.3 + 13
4 hr.	1.1 + 13	3.8 + 12	1.7 + 13	2.0 + 12	1.4 + 6	2.6 + 11	3.4 + 13
16 hr.	5.2 + 12	2.1 + 11	9.3 + 11	7.5 + 10	---	1.5 + 10	6.4 + 12

Note: See Table 12.4-6 for description of relief transient.

TABLE 12.4-8: EXPOSURES DUE TO SAFETY/RELIEF VALVE DISCHARGES (Type 2)
[HISTORICAL INFORMATION]

Organ	Dose (mrem)
Whole body	39
Skin	327
Thyroid	0.7

Table 12.4-9: Direct Shine Annual Dose to Members of the Public

	EPU (mrem)	40 CFR 190 Limit (mrem)
Direct Radiation	4.38	25

TABLE 12.4-10 DELETED

a.

TABLE 12.4-11: CONTAINMENT OCCUPANCY STAY TIMES DURING NORMAL OPERATIONS*

[HISTORICAL INFORMATION]

Station	**Occupancy, Person-hr/yr**
Sample Station	1250
RWCU	850
CRD	1591
Refueling Equipment	236
Standby Liquid Control	497
Sumps	274
Recirc Flow Control	314
Containment Cooling	618
TIP	159
C & I Panels	1013
ECCS & Process Sampling	1092
Containment personnel Lock	803
Miscellaneous	480

*Values are based on General Electric estimations and were derived from experience with previous plants with consideration of the equipment locations and activities characteristic of the Mark III containment.

Table 12.4-12: Current and Anticipated Measured Radiation Fields in Selected Areas

Building and Area Description	Radiation Zone (mrem/hr)	Anticipated Survey Results (EPU) (mrem/hr)
Turbine Building El. 166' (open areas)	B (≤ 2.5)	2.3
Turbine Building El. 133' (open areas)	B (≤ 2.5)	0.1-2.2
Turbine Building El. 113' (open areas)	B (≤ 2.5)	0.12
Turbine Building (near LP turbine)	E (> 100)	560 - 3920
Turbine Building (near condenser bays)	E (> 100)	10 - 3200
Turbine Building (near SJAE)	E (> 100)	0.6 – 2320
Turbine Building (near offgas condensers)	E (> 100)	315 – 5720
Turbine Building (near moisture separators)	E (> 100)	565 – 3960
Turbine Building (near condensate demineralizers)	E (> 100)	3 – 135
Turbine Building (near FW heaters)	E (> 100)	240 – 14,300
Turbine Building (near ventilation charcoal filters)	C (≤ 15)	< 0.1 – 0.12
Turbine Building (near ventilation HEPA filters)	C (≤ 15)	< 0.4 – 0.4
Auxiliary Building (open areas)	B (≤ 2.5)	2.3
Auxiliary Building (areas adjacent to SFP during normal operations)	B (≤ 2.5)	0.6
Auxiliary Building (areas adjacent to SFP during refueling)	B (≤ 2.5)	0.1 – 1.7
Containment Building (open areas)	B (≤ 2.5)	1.2
Radwaste Building (general areas)	B (≤ 2.5)	0.1 – 0.7
Refuel Floor (Aux Bldg El. 245' open area during normal operations)	B (≤ 2.5)	< 0.1 – 1.2
Refuel Floor (Aux Bldg El. 245' above SFP during refueling)	B (≤ 2.5)	< 0.1 – 1.2

Table 12.4-13: Estimated Annual Doses to Members of the Public Due to Normal Operations Gaseous and Liquid Radwaste Effluents

Type of Dose	10 CFR 50 Appendix I Design Objectives	EPU Doses
Liquid Effluents		
Dose to total body from all pathways	3 mrem/yr	0.401 mrem/yr
Dose to any organ from all pathways[1]	10 mrem/yr	0.929 mrem/yr (GI-Lli)
Gaseous Effluents		
Gamma dose in air[2]	10 mrad/yr	0.410 mrad/yr
Beta dose in air[2]	20 mrad/yr	0.224 mrad/yr
Dose to total body of an individual[2]	5 mrem/yr	1.22 mrem/yr
Dose to skin of an individual[2]	15 mrem/yr	1.97 mrem/yr
Radioiodines and Particulates Released to the Atmosphere		
Dose to any organ from all pathways[3]	15 mrem/yr	0.390 mrem/yr (Thyroid)

[1] With respect to dose due to the liquid effluent pathway, the GI-Lli is the dominant organ.
[2] The gamma and beta dose in air, including the dose to the total body and skin of an individual due to gaseous effluents is based on noble gases.
[3] With respect to dose due to iodines and particulates (including tritium) in the gaseous effluent pathway, the thyroid is the dominant organ.

12.5 RADIATION PROTECTION PROGRAM

12.5.1 Organization

12.5.1.1 Program and Staff Organization

The basic objective of the plant radiation protection program is to protect individuals from exposure to radiation and radioactive materials. Specifically, the radiation protection program consists of rules, practices, instructions, guidelines, and procedures (described in subsection 12.1.3 and Section 13.5) which will keep doses to individuals in restricted/controlled areas of the plant to levels that are as low as reasonably achievable (ALARA) and within the limits set forth in 10CFR20. Guidelines cover special nuclear, source, and by-product materials. The program is prepared and executed utilizing the recommendations given in Regulatory Guides 1.8, 1.39, 8.2, 8.4, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.34, 8.35 and 8.36 except as delineated in various sections of Chapter 12 and Appendix 3A. The program ensures that the radiation protection and training requirements of 10CFR19 and 10CFR20 are met.

The program ensures that radiation protection training is provided to workers; that personnel and in-plant monitoring is performed; and that records of training, exposure, and surveys are maintained. It also ensures that the GGNS commitment to maintain exposures as low as reasonably achievable is fulfilled. A more detailed discussion on ALARA management philosophy and responsibilities is provided in Section 12.1.

Responsibility for the operation of the radiation protection program and overall radiological safety at the plant rests with the Radiation Protection Manager. Responsibilities of the radiation protection staff relative to radiation protection are given below. The plant radiation protection section organization is shown on Figure 13.1-1.

The Radiation Protection Supervisors have functional control of and are responsible for establishing the radiation protection program. They have the responsibility for ensuring that the ALARA policy is implemented. The Radiation Protection Supervisors (or other designated individual qualified in accordance with Technical Specification 5.3.1 and Regulatory Guide 1.8) are designated as the backup Radiation Protection Manager.

Upon appointment to the active position, the Radiation Protection Manager will meet the position qualification requirements.

Radiation Protection Specialists are responsible for preparing reports, performing technical studies, reviewing and writing procedures and performing other special assignments.

The Radiation Protection Supervisors are responsible for the direction of the daily operation of the radiation protection program at GGNS and for the supervision of the health physicists.

The Health Physicists implement the radiation protection program by performing routine and special survey programs in accordance with plant radiation protection instructions. This level of staffing will provide adequate radiation protection coverage for plant personnel despite sickness, vacation, or employee attrition.

Radiation monitoring and control practices will be such that resultant radiation exposures and releases of radioactive materials in effluents to unrestricted areas are maintained as low as reasonably achievable. Records of surveys, radiation monitoring, and radioactive waste disposal will be maintained in accordance with the requirements of 10CFR20.

12.5.1.2 <u>Program Objectives</u>

The objectives of the radiation protection program are:

a. To provide administrative control of persons on the site to ensure that personnel exposure to radiation and radioactive materials is within the requirements of 10CFR20 and that such exposure is kept ALARA.

b. Deleted

c. To ensure that waste shipments meet guidelines established in plant procedures and instructions.

12.5.1.3 <u>Radiation Protection Program</u>

The station radiation protection program will be officially initiated when fuel is received and will be in effect continuously thereafter until the unit is decommissioned. This program consists of rules, management and worker philosophies, practices, instructions, and procedures that are used to accomplish the

objectives stated above in a practical and safe manner. The program is consistent with the recommendations of NRC Regulatory Guide 8.2.

The radiation protection program ensures that:

a. Personnel receive appropriate radiation protection training commensurate with their respective responsibilities

b. Appropriate access control techniques and protective clothing are used to limit external contamination

c. Respiratory protection equipment is used when appropriate to limit internal exposure from internal contamination

d. Radiologically controlled areas (RCA) are segregated and appropriately posted to limit exposure potential

e. Radiological instruments and equipment are provided and properly calibrated so that exposure potential may be assessed

f. Appropriate personnel dosimetry is provided

g. An internal dose assessment program (whole body counting and/or bioassay) is provided

A more detailed discussion of the procedures and instructions used to implement this program is contained in subsection 12.5.3.

12.5.2 Equipment, Instrumentation, and Facilities

12.5.2.1 Facilities

Radiation Protection and radiochemistry facilities include a Radiation Protection Supervisor's office located at El. 93'-0" in the control building, the radiation protection laboratory located at El. 93'-0" in the control building, the 133' Unit 2 Turbine building RCA Access point, the low-level counting room located in the water treatment building, and the hot radiochemistry lab located at El. 118'-0" in the radwaste building.

12.5.2.1.1 Radiation Protection Facilities

The radiation protection laboratory is located at elevation 93'-0" in the control building. Job planning and radiation work permit coordination is accomplished at the radiation protection laboratory, the 133' Unit 2 Turbine Building RCA Access Point or in the ALARA offices. Radiation Protection equipment (dose rate instruments, count rate meters, air samplers, environmental survey instruments...) and miscellaneous radiation protection supplies (survey maps, smears, air sampling media, paper, pens...) are located in the radiation protection laboratory or other designated areas.The radiation protection laboratory is equipped with filing cabinets and desks to provide work space for radiation protection personnel and storage space for records.

The locker room contains personal effects lockers for use by GGNS personnel. Adjacent to the locker room are toilet, washroom, and shower facilities. Protective clothing will be available at central locations in designated areas depending on the amount of work being performed and the conditions in the area. Change facilities will be provided for those areas where and when personnel traffic warrants. Such facilities may consist of benches and clothes racks.

Radiation Protection sample counting equipment is located in the Unit 2 Turbine Building at El. 93'-0". Equipment used for routine counting of smears and air samples such as pancake probe or end window G-M counters, alpha and beta scintillation detectors, and/or gas flow proportional counters is normally located in this area. Special samples requiring gamma isotopic analysis and/or low-level counting may be analyzed in the water treatment building low-level counting room.

Personnel decontamination facilities at the central access corridor consist of standup shower facilities and decontamination sinks. The decontamination area may also be used for decontamination of small equipment. Large or highly contaminated equipment may be decontaminated at the hot machine shop decontamination facility. Decontamination equipment may consist of ultrasonic cleaners, sand blasting cabinets, turbulators, stainless steel wash basins, and/or similar equipment. The ventilation systems for the access control area and the hot machine shop are described in subsection 9.4.10. Drainage from the decontamination facilities is held up as liquid radwaste for appropriate treatment.

The radiochemistry clean lab and counting room is located in the water treatment building at El. 133'-0". This area will be used for gamma isotopic analysis, beta and alpha low-level analysis, and other radiation protection/radiochemistry special sample counting and analysis.

12.5.2.1.2 Access and Exit of Controlled Access Areas

Routine access to and from the main controlled access area encompasing the turbine building, auxiliary building, containment, radwaste building and hot machine shop normally is through the 133' Unit 2 Turbine Building RCA Access Point. Other access points to and from the main controlled access area may be established by radiation protection management. At each of these exit points, whole body friskers and/or G-M tube friskers will be placed for personnel contamination surveys. Personnel will be required to monitor themselves prior to exiting the controlled access area.

12.5.2.2 Radiation Protection Instrumentation

12.5.2.2.1 Laboratory Instrumentation

Laboratory instrumentation located in the low-level counting room, hot lab, and radiation protection count room allow plant personnel to ascertain the radioactive material present in survey samples. Typical samples would be contamination survey smears, airborne survey filters, and charcoal cartridges, but tritium surveys and other samples may be processed also. Examples of laboratory instrumentation are listed in Table 12.5-2. Each laboratory counting system is checked and calibrated at regular intervals with standard radioactive sources traceable to a National Institute of Standards and Technology (NIST) source in accordance with approved instructions. Counting efficiency, background count rates, and high voltage settings are checked by qualified personnel in accordance with plant instructions.

12.5.2.2.2 Portable Survey Instrumentation

Portable survey instrumentation is located at the access control area and in-plant control points. This instrumentation will allow Radiation Protection and/or designated plant personnel to perform alpha, beta, gamma, and neutron surveys for radiation, airborne, and surface contamination control.

Each portable survey instrument will be calibrated annually, when in use, or prior to use, after undergoing repair work. Calibrations are normally performed using instrument calibrators (discussed in subsection 12.5.2.2.4) by qualified personnel. Instruments will be source-checked to verify proper operation in accordance with plant instructions. Sufficient quantities of each type of instrument are available to permit calibration, maintenance, and repair without causing a shortage in operational instrumentation. Examples of portable radiological survey instrumentation is listed in Table 12.5-1.

12.5.2.2.3 Personnel Monitoring Instrumentation

Personnel monitoring is provided by use of dosimeters of legal record (DLR), direct-reading dosimeters, and survey instrumentation. Personnel monitoring will be performed in accordance with 10 CFR Part 20 Section 1502.

Extremity dosimetry will be provided as necessary.

Direct-reading dosimeters will be calibrated: prior to use; if damage is suspected or repair is performed; every three months if the dosimeters are used as the primary method for compliance with 10CFR20 or every twelve months if the dosimeters are used to supplement the primary method for compliance with 10CFR20.

Personnel monitoring instrumentation will consist of, and not be limited to, G-M count rate meters (contamination friskers), portal monitors, and other instrumentation as identified in Table 12.5-1. These instruments will be calibrated annually when in use, or prior to use, after undergoing repair.

12.5.2.2.4 Radiation Protection Equipment

Portable air samplers are normally used to sample airborne radioactive material concentrations. Air samplers are calibrated for flow annually. Samples may be analyzed for radioactive particulate and radioiodine airborne concentrations.

Continuous air monitors may be used to monitor airborne concentrations at specific work locations. In such cases, local indication is provided as well as trend information. Alarm setpoints are variable; visual and audible alarms are provided.

Respiratory protective equipment is available at the access control area. Self-contained breathing apparatus for emergency use is available at the radiation protection locker and emergency equipment storage lockers. Equipment will be maintained and used in accordance with Regulatory Guide 8.15 except as noted in Appendix 3A.

Instrument calibrators will normally be used for calibrating gamma dose rate instrumentation. These may be self-contained, heavily shielded, multiple source calibrators. Beta and alpha radiation sources will also be available for instrument calibration. Calibration sources are traceable to the National Institute of Standards and Technology or equivalent.

Protective clothing will be supplied for personnel working in radiologically controlled areas, as needed. The clothing required for a particular instance will be prescribed by radiation protection personnel based on actual or potential radiological conditions.

An adequate inventory of protective clothing will be maintained on hand at the access control area or other control points, as necessary to support plant activities. This clothing will include such items as lab coats, overalls, hoods, caps, plastic oversuits, gloves (plastic, rubber, cloth), shoe covers, boots, and rubbers.

Additional contamination control supplies will be available. These include vacuum cleaners, mops, absorbent paper, plastic sheets and bags, barricade ropes, signs, and labels.

A listing of radiation protection equipment is given in Table 12.5-1.

12.5.2.2.5 Inplant Iodine Monitoring Instrumentation Under Accident Conditions

Inplant iodine monitoring will be accomplished by use of continuous air monitors and/or portable low volume samplers with subsequent laboratory analysis of filter media.

12.5.2.2.5.1 Continuous Air Monitors (CAMs)

CAMs will normally be utilized at GGNS to provide monitoring for particulate, or iodine, activity separately, and/or simultaneously. A TEDA impregnated charcoal, Silver Zeolite or other appropriate cartridge/filter paper will be used to collect radioiodine for CAMs which provide iodine monitoring.

CAMS are placed in general areas of interest to continuously monitor and record ambient airborne radioactivity levels under both normal and abnormal conditions.

12.5.2.2.5.2 Portable Air Samplers

Low volume portable air samplers will be used to collect radioiodine samples, as needed. Either TEDA-impregnated charcoal cartridges, 4 percent silver-loaded silica gel (silver gel) cartridges, or other appropriate cartridges will be used for sample collection.After sample collection, these cartridges will be carried to one of the inplant locations available for sample analysis. The normally used counting systems are located in the Water Treatment Building (Counting Room) and Radwaste Building (Hot Lab) both of which are physically separated from the Auxiliary Building. GeLi or equivalent detectors are available for analysis of the cartridges. If high levels of noble gas are present in the area sampled, silver gel or silver zeolite will be used for sampling to prevent collection of noble gas, or charcoal cartridges will be used and then purged with clean air via reverse flow through the cartridge, prior to analysis. Portable air sampling equipment is listed in Table 12.5-1. Laboratory counting equipment is listed in Table 12.5-2.

12.5.2.2.6 Other Radiation Protection Instrumentation

The area radiation monitoring system is installed in areas where it is desirable to have continuous dose rate information. Monitors indicate dose rate via control room readout and provide local audible and visual alarms upon reaching preset dose rate alarm levels. Fixed continuous airborne radioactivity monitors are also provided at strategic locations where personnel exposure to airborne radionuclides is likely. More information on these fixed instruments is given in Section 12.3.

12.5.3 Procedures

Procedures will be developed to cover necessary areas of plant operations and maintenance. Section 13.5 provides information on the procedures to be developed. In addition, this section describes certain methods that will be embodied in procedures and/or instructions to ensure that occupational radiation exposures will be ALARA. The SERI commitment to Regulatory Guides (see Appendix 3A) will be incorporated into procedures and/or instructions as appropriate.

Strict adherence to the GGNS plant administrative procedures and radiation protection procedures will ensure that personnel radiation exposures are both as low as reasonably achievable and within the limits of 10 CFR 20. Policy and operational considerations for radiation protection are set forth in subsections 12.1.1 and 12.1.3.

12.5.3.1 Radiation and Contamination Surveys

Radiation Protection personnel normally perform routine radiation and contamination surveys, the techniques of which are delineated in radiation protection procedures. Periodic surveys are taken in the containment and other accessible areas. Surveys are performed on a frequency that varies with the potential radiological hazards associated with a given area. Routine survey frequencies will be specified in radiation protection procedures. These surveys consist of radiation measurements and/or contamination surveys as appropriate for the specific area. Air samples are also routinely taken in portions of the controlled access area. Survey information is factored into exposure stay-time determinations and radiation work permit (RWP) specifications. An RWP may specify the need for additional surveys for specific operations and/or maintenance activities.

Radiation level surveys may be performed for alpha, gamma, beta, and/or neutron exposure rates. Contamination surveys are normally performed to establish gross beta-gamma contamination levels, but may be processed for specific types of radiation (beta-alpha-gamma) or specific radionuclides (via gamma spectroscopy). Air samples are normally taken to establish airborne concentrations of particulates, noble gases, and/or radioiodine, but specific nuclide information may also be obtained. Availability of current survey information will aid in keeping exposures ALARA.

12.5.3.2 Methods to Maintain Exposures ALARA

Methods to maintain exposures ALARA are not only included in plant administrative and radiation protection procedures but are also contained in applicable operating and maintenance directives and instructions. Some examples of the types of methods that will be used to maintain exposures ALARA are discussed below for the following operational categories:

12.5.3.2.1 Refueling

A portion of the water in the condensate storage tank is used to flood the volume over the reactor under the drywell head (reactor well) during refueling. This water has been purified by filter/demineralizers which reduces the amount of activity in the water that will be pumped to the reactor well. After the reactor coolant system is depressurized, it is degassed and sampled to verify that the gaseous activity is low prior to removing the reactor head. After flooding the refueling pool above the reactor, purification of the refueling pool water is continued to maintain exposures, due to activity in the water, ALARA. Movement of irradiated fuel assemblies will be accomplished with the assembly maintained underwater. The normal radiation level on the refueling bridge is expected to be less than 5 mrem/hour by following these procedures. The radiation work permit system is used to maintain positive radiological control over work in progress.

12.5.3.2.2 Inservice Inspection (ISI)

Prior to entry into radiation areas to perform inspections, personnel may study, as appropriate: blueprints, drawings, photographs, slides, videotapes, previous inspection reports, previous radiation and contamination surveys, and previous radiation work permits appropriate to the particular job. This will acquaint them with the job location, the work to be done, and radiation and contamination levels previously experienced. Surveys are performed to the extent required to determine present contamination and/or radiation levels. From this data, previous data, and past experience of personnel, a radiation work permit is issued detailing the necessary steps required to keep exposures ALARA. Equipment is checked and/or calibrated to verify it is operating properly prior to entry into the radiation area. Temporary shielding will be used, where practicable, to reduce radiation exposure.

12.5.3.2.3 Radwaste Handling

The handling of radwaste has been minimized by station design. The radwaste system is shielded and incorporates remotely operated liquid and solid radwaste systems. Filter media and spent resin can be sluiced remotely into a shipping container to allow solidification or dewatering. Evaporator bottoms can also be mixed remotely with a solidification agent and sluiced into shipping containers (which reduces exposures due to packaging and handling). Specific methods that are used to reduce exposures include such things as labeling drums prior to filling, when applicable, and using remote closure devices where applicable. An electrically operated pallet/fork truck and an overhead bridge crane may be used to remove containers from the filling area to the storage area and/or onto a conveyance for shipment.

12.5.3.2.4 Spent Fuel Handling, Loading, and Shipping

Spent fuel handling and loading of a shipping cask is performed underwater, using the fuel handling cranes and/or manual extension tools. This normally requires that a small crew work in the fuel handling area and usually involves little exposure. The radiation work permit system will be used to maintain positive radiological controls over this task.

Some of the methods used to assure this are:

a. Maintain at least 8 feet 6 inches of water above the fuel assembly active fuel to minimize direct radiation.

b. Purify fuel pool water to minimize exposure due to water activity.

c. Cool the fuel pool water to minimize inhalation doses due to airborne activity.

d. Provide continuous air sampling while moving spent fuel to evaluate airborne activity.

e. Provide an air sweep of the fuel pool surface when needed to reduce inhalation doses due to airborne activity.

f. Have emergency procedures immediately available.

After the shipping cask is loaded, it will be decontaminated to minimize loose contamination.

12.5.3.2.5 Normal Operation

The station was designed so that significant radiation sources are minimized, shielded, and/or placed in cubicles. Much instrumentation required for normal operation reads out remotely in the control room or in other low-radiation areas. Instrumentation that cannot be placed remotely or that is read infrequently is situated, where possible, so that it can be read from the entrance to the cubicle or from a low-radiation area within the cubicle. Operators are instructed to stay outside cubicles in which radiation levels are high as much as possible and are appraised of the areas inside cubicles where the radiation level is usually the lowest.

An area radiation monitoring system is available and provides indication of radiation levels and local alarms. The ventilation system is designed to minimize spread of airborne contamination.

12.5.3.2.6 Routine Maintenance

Routine maintenance falls into the categories of preventive maintenance (planned and scheduled maintenance such as lubrication, adjustments, and tests) and corrective maintenance (unscheduled maintenance such as valve packing, pump seal replacement, and stopping leaks). Instructions are written for the usual preventive maintenance jobs and for some recurring corrective maintenance jobs. These instructions specify the precautions to be taken. The instructions list the required lubricants, special tools and equipment, and the acceptance standards. This serves to minimize the time spent in the radiation area.

When a radiation work permit is issued, the radiation and/or contamination levels are listed, shielding is specified if appropriate and additional specific instructions are given to personnel. For corrective maintenance jobs in radiation areas, for which a general instructions is used, a similar approach is used.

Extension tools are used when practical. Detailed surveys are performed and the radiation work permit is issued (if required) with specific instructions. The individuals performing the work may be required to read instruction manuals or may be shown pictures or sketches to aid in understanding what is to be accomplished, how it is to be accomplished as safely and quickly

as possible, and what the acceptance criteria are. Additional requirements may be imposed to reduce exposures at the discretion of radiation protection.

Post-job debriefings may be utilized to obtain input from personnel actually performing the work in addition to supervisory and support personnel. This will assist in revising directives and instructions for ALARA considerations.

12.5.3.2.7 Sampling

Most sampling of radioactive systems is performed inside hoods in chemistry sampling stations, which protect personnel from airborne activity. Protective clothing and gloves are required when sampling radioactive systems to prevent contamination of personnel.

A survey instrument is used to check radiation levels while sampling and to determine radiation levels on the sample container when sampling systems with a significant potential for personnel exposure. If installed survey equipment is available at the sample location, it can be used to check radiation levels during sampling and on the sample container. The liquid sample container is normally washed with clean water and dried before being brought into the adjacent radiochemistry laboratory for analysis. The dose from sample bottles is minimized by grasping the bottle at the top, by using tongs, or by using a sample carrier.

12.5.3.2.8 Calibration

Calibration of portable gamma detection instruments is performed in a manner so as to maintain exposure of the persons performing calibrations ALARA. Portable sources used to calibrate fixed instruments (such as the area radiation monitoring system) are transported in shielded containers.

Where possible, fixed instruments requiring routine calibration are situated so that the necessary test signals can be provided from a low-radiation area with the instruments in place.

12.5.3.3 Controlling Access and Stay Time

The protected area includes all areas within the security fence which surround the station. This area is further broken down into radiologically controlled areas (for radiation protection

purposes). Radiologically controlled areas are further categorized as controlled access areas, radiation areas, high radiation areas, locked high radiation areas, very high radiation areas, airborne radioactivity areas, contamination areas, high contamination areas, and radioactive materials areas to comply with 10 CFR 20 and plant procedures and instructions.

Personnel entering the radiologically controlled areas of the station must be trained in radiation protection and emergency procedures and instructions as specified in subsection 12.5.3.8, or must be escorted. Entrance to the controlled access area is discussed in subsection 12.5.2. A radiation work permit may be issued, if required, for the areas to be entered. Stay time will be specified on RWPs as appropriate based on radiological survey information. Additional personnel monitoring devices, protective clothing, and respiratory equipment may also be issued, if required.

Access to high radiation areas is provided in accordance with GGNS Technical Specifications.

During major outages, such as refueling, RWP numbers may also be used to record personnel exposures and to correlate these to specific jobs. By using these RWP numbers, as described in the radiation protection procedures, exposure information relating to specific jobs can be accumulated for preplanning sessions in the future. In addition, on complex or new jobs involving significant exposure, a review is performed in accordance with administrative procedures in an attempt to improve methods and keep exposures ALARA. This technique has previously been discussed in subsection 12.1.3

12.5.3.4 Contamination Control

Contamination limits for personnel, equipment, and areas are listed in the station administrative procedures. Surveys are performed routinely, as discussed in subsection 12.5.3.1, to determine contamination levels. Additional surveys may be performed after maintenance work or after an operation that may have increased contamination levels. Any area found contaminated is identified and access is controlled in accordance with approved procedures. Contamination areas are decontaminated as soon as practical.

Tools and equipment used in contaminated areas are monitored and/or bagged (or wrapped) prior to being removed from the work area to prevent the spread of contamination. All tools and equipment being removed from the controlled access area are surveyed for contamination by radiation protection personnel (or other qualified personnel as specified in plant administrative procedures and directives), to ensure that they meet clean area limits. If the tools and equipment do not meet the limits they are decontaminated to meet the limits prior to release from the controlled access area. If decontamination is not practical or if after decontamination the tools and equipment do not meet the clean area limits, they are retained in the controlled access area. Decontamination facilities were discussed previously in subsection 12.5.2.

Control of personnel contamination (external and internal) is provided by using protective clothing and respiratory equipment. Each individual is responsible for surveying himself and his clothing when he exits a contamination area or controlled access area. If contamination is found, the individual is decontaminated, using the facilities previously described in subsection 12.5.2, under the direction of radiation protection or other qualified personnel.

In many areas special coatings are applied to walls and floors of areas containing radioactive fluids which will aid in decontaminating these areas if it should become necessary.

In addition, equipment vents and drains normally are piped directly to sumps (or other collection devices) to prevent radioactive fluids from flowing freely to the nearest drain.

12.5.3.5 Respiratory Protection

If personnel entry is required into areas where the sources of airborne radioactivity cannot be removed or controlled, either occupancy will be controlled and/or respiratory protection equipment will be provided to maintain exposures within the limits of 10 CFR 20 Subpart C. When airborne radioactivity is detected which is greater than 30 percent of the limits specified in 10 CFR 20, the area is isolated and posted as an airborne radioactivity area and access is controlled. Entry into these areas requires the issuance of a radiation work permit. The radiation work permit system (subsection 12.5.3.7) provides radiation exposure control by controlling and recording

conditions under which work in airborne radioactivity areas is performed. Air sampling techniques are used to ensure that appropriate respiratory protective equipment is specified on the radiation work permit. Selection of the appropriate type of respiratory equipment is then determined. The respiratory protection program is organized to conform to 10 CFR 20 Subpart H and Regulatory Guide 8.15 with clarifications as described in Appendix 3A.

Normally respiratory equipment is available in various areas of the plant and is issued at the 133' Unit 2 Turbine Building RCA Access Point. In some cases, such as emergencies or outages, respiratory equipment can be obtained from the control room, emergency kits and various radiation protection control points. The respiratory equipment that is available includes full-face masks and self-contained breathing equipment. Table 12.5-1 gives examples of respiratory equipment.

12.5.3.6 <u>Personnel Dosimetry</u>

Plant employees, visitors, and support personnel are normally required to wear a personnel dosimeter when they are in the radiologically controlled area.

Visitors and support personnel may be initially issued dosimeters at the radiation protection dosimetry facility in the Site Processing Facility. Only those individuals who have completed radiation worker training in radiation protection and emergency preparedness in accordance with plant procedures will be authorized to enter radiologically controlled areas unescorted once the proper security clearance has been obtained. When visitors and other persons who have not completed this training enter a radiologically controlled area, they must be escorted by personnel who have completed the required training.

DLRs will be processed in accordance with a NVLAP accepted program as required by 10CFR20.1501. Personnel dosimeters will be read daily for each individual who has entered a radiologically controlled area. This will provide a daily estimate of personnel exposure. DLR badges may be processed more frequently during outages, refueling, or when an individual's exposure status is in doubt. Dosimeter readings will normally be recorded and/or maintained by plant personnel.

Each member of the plant operating staff will be monitored for intakes of radioactive material if required, per 10CFR20.1502(b). A whole-body counter will be located at GGNS for use as necessary.

Additional bioassays will be performed on an individual basis as necessary. These special individual measurements will be initiated when the results of monitoring in the workplace indicate that significant intakes may have occurred or when workers have been associated with known incidents possibly involving significant intakes of radioactivity. Air monitoring and surface contamination surveys in the workplace and tests of skin contamination, nose blows, and nasal smears will be used for determining whether special measurements will be required. Investigation levels are established in plant administrative procedures.

Exposure data of personnel will be collected and recorded on Form NRC-5, or the equivalent. Occupational exposures incurred by individuals prior to working at GGNS will be summarized on Form NRC-4, or the equivalent. These records will be maintained in accordance with 10CFR20.2106.The GGNS internal dosimetry program conforms to Regulatory Guides 8.9 Revision 1 and 8.34.

12.5.3.7 Radiation Work Permits

Radiation work permits will be issued by radiation protection personnel prior to allowing work to be performed in certain radiologically controlled areas. The Radiation Protection Manager or designee has direct management responsibility for the issuance of radiation work permits. Radiation work permits state allowable stay times (if appropriate), protective clothing and equipment required, monitoring requirements, and any special instructions or cautions pertinent to radiological hazards. These permits ensure that work is performed in a radiologically safe manner. Violations of permits which affect radiological safety will be reported by condition reports. The employee will be carefully instructed on avoiding violation in the future and consequences of such violations. Repeated or willful violation will be cause for disciplinary action.

12.5.3.8 Radiation Protection Training

Each member of the permanent operating organization whose duties entail entering radiologically controlled areas or directing the activities of others who enter radiologically controlled areas will be instructed in the fundamentals of radiation safety. A

description of the radiation safety course is given in Section 13.2. All permanent plant personnel will be required to attend a retraining program in radiation protection every year. Personnel whose duties do not require entry into radiologically controlled areas will be made aware of the reasons for keeping out of such areas. The training program will include instruction in applicable provisions of the Nuclear Regulatory Commission's regulations for the protection of personnel from exposures to radiation or radioactive materials. These instructions will be provided pursuant to 10CFR19.

Considerable time and effort will be devoted to ensure that employees understand radiation and radiation safety as it applies to their work. Supervisors are responsible for ensuring that their employees follow proper radiation protection procedures and instructions. The amount and type of training will depend on the radiological hazards associated with the work they perform. Orientation lectures on radiation and radiation protection will be given to all new employees entering radiologically controlled areas of the plant.Training will continue with detailed discussions of the specific radiological hazards associated with work assignments. In the course of their work employees may receive additional training in radiation protection practices from supervisors, senior co-workers, and radiation protection personnel. The necessity for each employee to recognize radiological conditions and minimize his/her exposure will be stressed in training sessions.

12.5.3.8.1 Respiratory Protection Training Program

Individuals requiring access to areas where respiratory protection will be utilized will complete the respiratory protection training program. The instructor will be a qualified individual with a thorough knowledge and considerable experience regarding the application and use of respiratory protective equipment and the hazards associated with radioactive airborne contaminants.

Training will include lectures, demonstrations, discussions of pertinent station procedures, and actual wearing of respirators to become familiar with the various devices utilized at Grand Gulf Nuclear Station. The program will include as a minimum: discussion of the airborne contaminants against which the wearer is to be protected, including their physical properties, DACs, physiological effect, toxicity, and means of detection;

discussion of the construction, operating principles, and limitations of the respirator and the reasons the respirator is the proper type for the particular purpose; discussion of the reasons for using the respirators and an explanation of why more positive control is not immediately feasible, including recognition that every reasonable effort is being made to reduce or eliminate the need for respirators; instruction in procedures for ensuring that the respirator is in proper working condition; instruction in fitting the respirator properly and checking adequacy of fit; instruction in the proper use and maintenance of the respirator; discussion of the application of various cartridges and canisters available for air-purifying respirators; instruction in emergency action to be taken in the event of malfunction of the respiratory protective devices; review of radiation and contamination hazards, including the use of other protective equipment that may be used with respirators; classroom and practical (hands-on) training to recognize and cope with emergency situations; and other special training as needed for special use. Computer Based Training may be used in place of lectures to provide information to a trainee. Individuals will be required to don the device(s) that may be used and will perform the appropriate leak detection pressure test.

A quantitative test will normally be utilized to quantitatively measure and record leakage. If leakage exceeds the devices rated protection factor and retests confirm this, the individual will not be approved to use the device. After successful completion of training and fitting programs, appropriate records will be maintained to ensure individuals meet all required qualification criteria (medical, training, fitting) and are issued only the approved type and model of protective device(s). Individuals will receive retraining and reconfirmation of respirator fit at least every two years. Related records will be maintained as Quality Assurance records.

12.5.3.9 Radioactive Materials Safety Program

Various types and quantities of radioactive sources are employed to calibrate the process and effluent radiation monitors, the area radiation monitors, and portable and laboratory radiation detectors. Check sources that are integral to the area, process, and effluent monitors consist of small quantities of by-product material and do not require special handling, storage, or use-procedures for radiation protection purposes. The same consideration applies to solid and liquid radionuclide sources of

exempt quantities or concentrations which are used to calibrate or check the portable and laboratory radiation measurement instruments. Recognized methods for the safe handling of radioactive materials, such as those recommended by the National Council of Radiation Protection and Measurements, are incorporated in procedures and instructions to ensure proper handling of radioactive material. External doses will be minimized by a combination of time, distance, and shielding considerations. Internal doses will be minimized by the measurement and control of loose contamination.

Licensed radionuclide sources will be subjected to material controls for radiological protection. These controls will include:

a. Monitoring of shipments containing radioactive materials in accordance with 10 CFR 20.1906.

b. Monitoring of each sealed source for removable surface contamination (leakage testing) in accordance with TRM requirements.

c. Labeling of each source or source holder with the radiation symbol, stating the activity, isotope, and source identification number.

d. Storage in a locked area of each source that is not installed in an instrument or other piece of equipment.

e. Inventorying of sources every six months.

f. Maintenance of records on the results of inventories, leakage tests, use, location, condition, principal user, and the receipt and final disposition dates for all sources.

Additional details of the materials safety program are provided in the plant radiation protection instructions.

Radioactive sources that are subject to the material controls described herein will be used or handled only by or under the direction of radiation protection personnel or personnel approved by the Radiation Protection Manager or designee. Each individual using these sources will be familiar with the radiological restrictions and limitations placed on their use. These limitations protect both the user and the source. The experience,

qualifications, and training programs for personnel who are responsible for handling and monitoring radioactive materials are described in subsection 12.5.3.8 and Section 13.2.

TABLE 12.5-1: PORTABLE RADIATION PROTECTION INSTRUMENTS

Instrument	Number	Radiation Detected	Accuracy	Range	Remarks
Ion Chamber Survey Meter	2	Beta, gamma	±20% of reading	0-20,000 R/hr.	Multiple probe dose rate survey instrument
Ion Chamber Survey Meter	20	Beta, gamma	±10% of full scale	0-50 R/hr	
G-M Survey Meter	4	Beta, gamma	±10% of full scale	0.1 mR/hr.-1000 R/hr.	Extendible Dose rate instrument
G-M Survey Meter	12	Beta, gamma	±10% of full scale	0-500,000 CPM	Portable, battery-operated count rate instrument
G-M Survey Meter	6	Beta, gamma	±10% of full scale	0-200 mR/hr.	Survey instrument
G-M Survey Meter (frisker)	24	Beta, gamma	±10% of full scale	0-500,000 CPM	Count rate instrument for personnel surveys
Neutron Survey Meter	3	Neutrons	+15% (moderated CF-252)	0-5000 mrem/hr.	BF Tube within cadmium load Polyethylene sphere. Neutron Dose Rate Inst. (Rem Counter)

TABLE 12.5-1: PORTABLE RADIATION PROTECTION INSTRUMENTS (Continued)

Instrument	Number	Radiation Detected	Accuracy	Range	Remarks
Aplpha Survey Meter	4	Alpha particles	±10% of full scale	0-500,000 CPM	Alpha Scintillation (Zns) count rate instrument
Personnel Dosimeters	200	Gamma	±10%	0-200 mR	Direct reading (ion chamber)
Personnel Dosimeters	50	Gamma	±10%	0-500 mR	Direct reading (ion chamber)
Personnel Dosimeters	50	Gamma	±10%	0-2 R	Direct reading (ion chamber)
Personnel Dosimeters	50	Gamma	±10%	0-5 R	Direct reading (ion chamber)
Personnel Dosimeters	50	Gamma	±10%	0-100 R	Direct reading (ion chamber)

TABLE 12.5-1: PORTABLE RADIATION PROTECTION INSTRUMENTS (Continued)

Instrument	Number	Radiation Detected	Accuracy	Range	Remarks
Personnel Dosimeters	100	Gamma	±10%	0-200 R	Direct reading (ion chamber)
Continuous Air Monitor	2	Beta, gamma	±20% of reading	0-50,000 CPM	Adjustable alarm
DLR (Dosimeter of Legal Records)	As required	Beta, Gamma Neutron	±20% Beta-Gamma	10mR-10R	
Portable High Volume air Sampler	15	N/A	N/A	0-2.5 SCFM flow rate	Used for Grab Samples in Work Areas
Dosimeter Charger	5	-	-	-	-
Whole Body Friskers	3	Beta, gamma	-	-	-
Low Volume Air Sampler	10	N/A	N/A	0-2.5 SCFM	Continuous low volume sampling

TABLE 12.5-1: PORTABLE RADIATION PROTECTION INSTRUMENTS (Continued)

Instrument	Number	Radiation Detected	Accuracy	Range	Remarks
Instrument Calibrator	1	Gamma	-	.002-500 R/hr	Multiple source shielded self contained calibrator
Air Purifying Respirators	50	N/A	N/A	N/A	Full face-negative pressure mode
Atmosphere Supplying Respirators	10	N/A	N/A	N/A	Air line respirator continuous flow mode full face mask
Atmosphere Supplying Respirators	50	N/A	N/A	N/A	SCBA positive pressure mode

NOTE: Adequate supplies will be made available for routine operations and outage conditions.

TABLE 12.5-2: FIXED LABORATORY INSTRUMENTATION

Instrument	Number (See Note)	Radiation Detected	Efficiency	Range	Remarks
COUNTING ROOM					
Gamma spectrometer	1	Gamma	~15-20% relative to 3”x3” NaI	0-2 MeV	Computer based pulse height analysis sys. with Ge detector
Gas flow proportional counter	1	Beta, Alpha	~45% Sr-90 ~33% Po-210	0-10^{6} Counts	End window gas flow detector
Liquid Scientillation detector	1	Beta (H-3)	~40% H-3	0-10^{6} Counts	To be primarily used for liq. samp. for tritium
HOT LAB:					
Gamma Spectrometer	1	Gamma	~15-20% relative to 3”x3” NaI	0-2 MeV	Computer based pulse height analysis sys. with Ge detector

TABLE 12.5-2: FIXED LABORATORY INSTRUMENTATION (Continued)

Instrument	Number (See Note)	Radiation Detected	Efficiency	Range	Remarks
RADIATION PROTECTION COUNT ROOM					
Alpha Counter Scaler	1	Alpha	~45% Sr-90 ~33% Po-210	$0\text{-}10^{6}$ Counts	To be used for Counting Air Samples and Smears
G-M Counter Scaler	1	Beta, Gamma	10% Co-60	$0\text{-}10^{6}$ Counts	To be used for Counting Air Samples and Smears

Note: Quantity and type of instrumentation will vary based on analytical requirements.

12.6 DESIGN REVIEW OF PLANT SHIELDING

12.6.1 Introduction

NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used In Postaccident Operation," identified the requirement that a review be performed of the radiation and shielding design of the spaces around systems that may, as a result of an accident, contain highly radioactive materials.

The radiation and shielding design review was performed to identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, the review results ensure that adequate access to vital areas and protection of safety-related equipment are provided through the use of design changes, additional shielding, or administrative control changes. The issue of personnel access requirements following a postulated LOCA is also addressed in NUREG-0737. Grand Gulf will comply with these requirements.

12.6.2 Systems Identified for Shielding Review

Systems outside containment which will or may have to function during a serious transient or accident and which may contain highly radioactive materials can be classified into two categories as follows:

a. Those systems required for plant shutdown or mitigation of accident consequences which are also expected to contain highly radioactive materials.

b. Those other systems directly connected to the reactor coolant system, suppression pool or containment atmosphere which, while neither designed to nor expected to contain highly radioactive material, are postulated to contain such material.

The systems containing radioactive fluid that were identified for review of plant shielding and environmental qualifications are:

a. Portions of the RHR system during the following accident modes:

   1. LPCI mode - Portions used to inject the suppression pool water.

   2. Shutdown cooling mode - Portions to provide "normal" residual heat removal service (although not expected to be used).

   3. Suppression pool cooling mode - Portions used to recirculate the suppression pool water for cooling.

   4. Containment spray mode - Portion used to recirculate suppression pool water for containment spray.

b. Portions of the low pressure core spray (LPCS) system used to recirculate the suppression pool water.

c. Portions of the high pressure core spray (HPCS) system used to recirculate the suppression pool water.

d. Portions of the reactor core isolation cooling (RCIC) systems as follows:

   1. Steam supply piping and turbine exhaust piping.

   2. RCIC pump suction from suppression pool.

e. Feedwater leakage control system

f. MSIV leakage control system and SGTS

g. Post-accident sampling system

h. Portions of combustible gas control system associated with H_2 analyzers.

i. Portions of suppression pool makeup system associated with RHR/LPCS sealing water for suppression pool level instrumentation.

j. Portion of the drains system located in auxiliary building required to accomplish radwaste pump-back function (system under design development).

12.6.3 Shielding Review Methodology

12.6.3.1 Source Term

The fission product release data used in the shielding review analysis was obtained from GEH for a 1 MWt BWR-6 with a total burn-up of 1095 MWd. These data were then adjusted to a power level of 4025 MWt to obtain a conservative fission product inventory for the Grand Gulf core. This fission product inventory was then partitioned into three categories of sources as described below.

12.6.3.2 Release Fractions

For those systems which contain pressurized reactor coolant, 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the particulates were assumed to be present with a dilution volume equal to the pressure vessel liquid volume (11,085 ft^3). Systems containing recirculating liquid were assumed to have 50 percent of the iodine and 1 percent of the particulates diluted by the suppression pool volume plus the pressure vessel liquid volume (146,320 ft^3, total). All airborne sources were assumed to contain 100 percent of the noble gases and 25 percent of the iodines. The dilution volume for contained airborne sources was selected two ways. If the system contains drywell air, only the drywell volume was used; otherwise, the entire volume of the containment plus drywell was used for dilution. Instantaneous mixing between the drywell and containment was assumed for all containment through-wall and through-penetration doses; this assumption maximized these contributions.

12.6.3.3 Post-Accident Radiation Zone Maps

Using the sources of subsection 12.6.3.2, dose rate versus distance curves were developed for containment wall and penetration contributions using the QAD-CG code and for contained source contributions from various pipe sizes using point kernal methodology. For contained sources, infinite pipe length was conservatively assumed. The contributions from all sources were then summed to develop the post-accident radiation zone maps, Figures 12.6-1 through 12.6-6.

12.6.4 Identification of Areas Outside Containment for Review

Areas outside the containment which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident are identified in Table 12.6-1.

12.6.5 Integrated Personnel Exposures

Access to various plant locations may be required after a loss-of-coolant accident. The Emergency Procedures were reviewed to determine which plant locations may require access to either control the plant or assist in the post-accident plant recovery. Based upon these assumptions the integrated personnel doses were calculated for the locations and presented in Table 12.6-2.

TABLE 12.6-1: AREAS OUTSIDE OF CONTAINMENT FOR REVIEW OF PLANT SHIELDING

Building	Elevation	Area
Auxiliary Building	93'-0"	RHR A, B, and C equipment rooms
		RCIC equipment room
		HPCS equipment room
		LPCS equipment room
		ECCS instrument track areas
		Floor and equipment transfer tanks and pumps rooms
		Refuel water transfer pumps room
		Control rod drive pumps and filter area
Auxiliary Building	119'-0"	RHR equipment rooms
		ESF switchgear rooms
		MSIV leakage control system equipment area
Auxiliary Building	135'-0"	Standby gas treatment equipment room and sampling area
		RHR equipment rooms
		BOP computer multiplexing and isolation panels areas
		CRD RPV temperature recorder panel area
Auxiliary Building	166'-0"	Containment hydrogen sample rack areas
		Radiation monitoring rack areas
		Containment exhaust charcoal filter room
		Instrument air booster compressors area
Auxiliary Building	184'-0"	Vicinity of blowout shaft area
Auxiliary Building	208'-10"	Enclosure building fans

TABLE 12.6-1: AREAS OUTSIDE OF CONTAINMENT FOR REVIEW OF PLANT SHIELDING (Continued)

Auxiliary Building	245’-0”	HVAC units area
Control Building	93’-0”	Radiation Protection area
Control Building	111’-0”	Remote shutdown panel area
Control Building	148’-0”	Lower cable spreading room
Control Building	166’-0”	Main control room
Control Building	177’-0”	Technical Support Center
Control Building	189’-0”	Upper cable spreading room
Turbine Building	-	General access to all areas
Turbine Building	93’-0”	Post-accident sample room
Radwaste Building	118’-0”	Radioactive chemistry laboratory
Diesel Generator Building	-	General access to all areas

TABLE 12.6-2: VITAL AREAS WHERE PERSONNEL ACCESS IS REQUIRED POST-ACCIDENT DOSES FOR 30 DAYS AFTER ACCIDENT

Location	Type of Access	Integrated Personnel Dose (Rem)*	Dose Rate Avg. over 30 days (mRem/hr)***
Control Room	Continuous	0.054	0.075
Technical Support Center	Continuous	0.054	0.075
Remote Shutdown Panel	Extended	4**	
Diesel Generator Buildings	Extended	0.12	
Post-Accident Sampling Station	Extended	See Note 1	
SGTS Sampling Station	Intermittent	0.44	
Laboratories	Extended	4.38	
ADS Air Supply Makeup Connection	Intermittent	3.46	
ADS Booster Compressor Area	Single Entry	1.13	

*Per operator

**Assumes continuous occupancy after first day

***Calculated for continuous occupancy only

Note 1: Dose rate dependent upon sample taken, source term used and time after accident

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]