

CHAPTER 12

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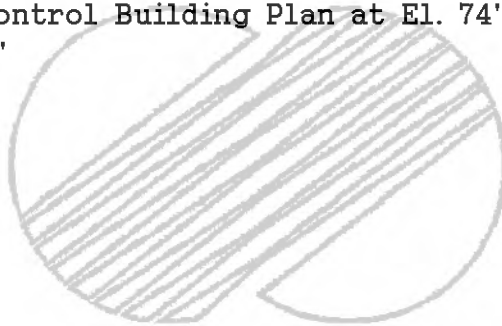
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12. RADIATION PROTECTION

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

Quantitative standards of radiation protection are established by the International Commission on Radiological Protection (ICRP), the National Council on Radiation Protection (NCRP), and the former Federal Radiation Council (FRC) which is now part of the Environmental Protection Agency (EPA). The recommendations of these bodies are reflected in 10 CFR Part 20, "Standards for Protection Against Radiation." Specifically, this regulation establishes maximum allowable levels for occupational radiation exposure.

This chapter describes the radiation protection measure of the station design and the operating policies to ensure that internal and external radiation exposures to station personnel, contractors, and the general population due to station conditions, including anticipated operational occurrences, will be within applicable limits, and furthermore, will be as low as is reasonably achievable (ALARA).

Radiation protection measures include: separation of radioactive components into separately shielded cubicles; use of shielding designed to adequately attenuate radiation emanating from pipes and equipment which are sources of significant ionizing radiation; use of remotely operated valves or handwheel extensions; ventilation of areas by systems designed to minimize inhalation and submersion doses; installation of permanent radiation monitoring systems; control of access to the site and to restricted areas; training of personnel in radiation protection; and development and implementation of administrative policies and procedures to maintain exposures ALARA.

12.1.1 POLICY CONSIDERATIONS

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. These reviews were consistent with the recommendations of USNRC Regulatory Guides 8.8 and 8.19.

The plant design was reviewed, updated, and modified as necessary during the design and construction phases. The plant layout r shielding, ventilation, and monitoring designs were integrated with traffic control, security, access control, and health physics aspects to ensure that the overall design resulted

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in a plant that will enable the lowest practicable exposures to be achieved.

The design ensures that all piping containing radioactive fluids is adequately shielded and properly routed to minimize exposure to personnel. 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 12.5.3, will be conducted to ensure that there are no defects in the shielding that might seriously affect personnel exposures during normal operation and maintenance of the plant.

The information, resulting from the study of operating plant experience and designs, is used as criteria for facility design to ensure operator exposures are ALARA. Design reviews ensure conformance with the criteria. Suggested improvements are presented as requirements or recommendations. Subsequent reviews of the revised designs are checked for conformance with these requirements.

Subsections 12.1.2 and 12.1.3 describe the criteria and methods used by the plant architect engineer in the design of the shielding, ventilation, and radiation monitoring systems, including equipment and plant arrangements and access control provisions, to keep occupational exposures ALARA. The design of these radiation protection features of the power block is reviewed by qualified engineers as a normal part of the plant architect engineer design activities.

12.1.1.2 Operation Policies

The management of Korea Electric Power Corporation (KEPCO) is committed to keeping occupational exposure to ionizing radiation ALARA. Accordingly, the Health Physics Program is prepared and conducted in conformance with the recommendations contained in Regulatory Guides 8.8 and 8.10. Title 10 Code of Federal Regulations, Part 20, provides the regulatory framework under which the ALARA philosophy is implemented.

The manager, Radiological Control Section, is responsible for the radiation protection program for YGN 1 & 2. The program consists of written health physics procedures, intensive training of radiation protection fundamentals, and periodic reviews by plant management. Section 13.1 presents the Organization of the Radiological Control Section for YGN 1 & 2 (figure 13.1-3).

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The manager, Radiological Control Section, who is responsible for the initial radiation protection program formulation, reports directly to the plant manager, and exercises supervisory control over the Health Physics group. The health physicist reports to the manager, Radiological Control Section, and exercises supervisory control over the health physics technicians and instrument repairman. The health physics technicians are responsible for radiation surveys, contamination surveys, and air sampling.

Plant personnel are either trained or indoctrinated in radiation protection by the manager, Radiological Control section. The radiation protection functions of the Health Physics group includes:

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- A. Assuring that dose limits established in accordance with the approved radiation protection program are not exceeded by plant personnel or visitors and that any dose received is ALARA.
- B. Controlling radiation exposure by:
 - 1. Evaluating radiological conditions and taking precautionary measures
 - 2. Controlling personnel-and equipment movement into and out of controlled areas.
 - 3. Ensuring proper use and care of special protective clothing and equipment.
 - 4. Conspicuously posting each area within the controlled area with appropriate caution signs.
 - 5. Administering and controlling conditions of radiation work permits for work in areas having high radiation and/or contamination levels in accordance with approved procedures.
- C. Determining requirements for and extent of the use of personnel monitoring devices and maintaining records of personnel exposure.
- D. Controlling and accounting for all radioactive material entering or leaving the plant site.
- E. Establishing procedures for dealing with potential or actual emergency conditions.
- F. Training the plant staff and visitors in radiation protection policy and procedures, as required.

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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.

12.1.2.1 General Design Considerations for ALARA Exposures

General design considerations and methods employed to maintain in-plant radiation exposures ALARA in accordance with 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 and in the vicinity of plant equipment expected to require personnel attention.

In designing radioactive systems, many factors must be considered. Important considerations are to reduce the need for equipment maintenance, to minimize necessary maintenance times, and to lower the radiation levels in which maintenance and other operational activity are performed.

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 anticipated operational occurrences.

12.1.2.1.1 Equipment General Design Considerations for ALARA

- A. Equipment general design considerations 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.

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3. Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment.
 4. Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when no feasible method is available to reduce radiation levels.
- B. Equipment general design considerations directed toward minimizing radiation levels proximate to equipment or components requiring personnel attention include:
1. Provision for draining, flushing, or, if necessary, remote cleaning of equipment containing radioactive material.
 2. Design of equipment, piping, connections, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps.
 3. Provisions for minimizing the spread of contamination into equipment service areas including direct drain connections.
 4. Provisions for isolating equipment from radioactive process fluids.
 5. Provision for a spent fuel pool and refueling cleanup system to maintain the radiation level of the fuel pool area within the Zone 2 limit. See section 12.5 for the description of radiation zones.
 6. Heat exchangers have been provided with corrosion-resistant tubes with tube-to-tube sheet joints fabricated to minimize leakage. Impact baffles are provided and process fluid velocities are limited as necessary to minimize erosive effects. Provisions are made for removal of the tubes for maintenance.
 7. Pumps in radioactive service have been provided with mechanical seals to reduce seal servicing time. Smaller pumps are provided with flanged connections for ease in removal.

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8. Water is used to fluidize tanks from which resin is transported. Resin tanks incorporate integral self-cleaning screens in overflow connections to retain resins within the tank. Overflow connections for radioactive tanks are piped to the liquid radwaste system (LRS) to facilitate radwaste processing.
9. Filters are supplied with the means to perform cartridge replacement with remote tools.
10. Demineralizers are designed to remotely remove spent resins hydraulically and replace with new resins from a remote location. Resin strainers have been designed for full system pressure drop.
11. Evaporators are provided with chemical addition connections to allow the use of chemicals for descaling operations.
12. The reactor head laydown area has been designed to substantially reduce both the dose to those changing O-ring seals and personnel working in adjacent areas. An interior concrete ring shield wall separates the O-ring changeout from the radiation field under the hemispherical head.
13. Frequently operated valves of highly radioactive systems are designed for remote operation. Motor operators, air operators, and reach rods are provided where necessary. The criteria for selecting valve operators are as follows:
 - a. In radioactive areas Zones 4, 5, and 6, valves which are operated frequently; for example, on a weekly basis or more often, are equipped with a remote actuator such as a simple reach rod, electric motor actuator, or pneumatic actuator and position indicator. These valves are controlled from the applicable control station or operating aisle.
 - b. Valves which are operated occasionally; for example, between one and twelve times a year, are classified as follows:
 - (1) Valves which are located in radiation areas of 100 mR/h (Zone 5) or less during operation may be manually operated directly.

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- (2) Valves which are located in higher radiation areas (Zone 6) are generally equipped with a simple reach rod, or, if a simple reach rod cannot be installed, an electric motor actuator or pneumatic actuator and a position indicator.
- c. Valves which are operated infrequently during normal plant operations; for example, less than once a year, are manually operated directly unless their operation would result in excessive personnel exposure.
- d. Simple Reach Rod

A reach rod arrangement is considered "simple" if it has a maximum of one directional change, less than 10 feet of rod length, and one simple wall penetration with positioning tolerances in the range of inches. Rod lengths greater than 10 feet frequently require supplementary supports which are undesirable since they further restrict maintenance access to the area and increase maintenance time and radiation exposure. Simple reach rods are used to the extent practical.
- e. Valves which are operated after an accident are provided with remote operation capability in cases where the operation of such a valve may lead to exposures in excess of 3 rem.
- f. Radiation tolerant materials are used in valves in accordance with their radioactive service.
- 14. Chemical seals are provided on instrument sensing lines for process piping which may contain highly radioactive solids to reduce the servicing time required to keep the lines free from solids. Primary instrument devices, which for functional reasons are located in high radiation zones, have been designed for uncomplicated removal to a lower radiation zone for calibration or servicing.
- 15. In addition to adequate shielding, the sample laboratory is equipped with a fume hood. Shielded sample bombs are provided to sample pressurized

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primary coolant. The sample stations are designed so that sample lines may be purged to the LRS or chemical and volume control system prior to sampling. Also, sample lines incorporate the capability of being flushed.

16. Remotely operated equipment is provided where practical to minimize operator radiation exposure during plant operation. An automated radwaste solidification and encapsulation system is employed to minimize exposure during radwaste processing.
17. The following design provisions are incorporated in order to minimize leakage in systems which may contain radioactive materials following a postulated accident, including the residual heat removal system, the containment spray system, the post-accident sampling system, and recirculation portions of the safety injection system.
 - a. All large valves (2-1/2 inches and larger) are provided with double packing and lantern ring leakoffs. These leakoffs outside the containment are piped either to closed drain systems (such as the equipment drain tank) or are piped to engineering safety feature (ESF) pump rooms which have filtered atmospheres during post-accident conditions.
 - b. All piping high point vents and low point drains in high pressure systems (900 pound rating and higher) are provided with double valves.
 - c. Welded pipe construction is used throughout except where flanged connections are required for equipment maintenance.
 - d. Leak detecting level instrumentation is provided in sumps located in ESF pump rooms.
18. In selected areas of the Auxiliary Building, a high density concrete of 3.45 g/cm^3 is used to increase the effectiveness of the shield walls under post-accident conditions. See paragraph 12.2.1.3 for a discussion of the post-accident source terms used in the shielding analysis for the high density concrete. In addition, refer to subparagraph 12.3.2.2.3 for a detailed listing of the areas in the Auxiliary Building in which high

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density concrete is utilized in the shielding design.

12.1.2.1.2 Facility Layout General Design Considerations
for ALARA

- A. Facility general design considerations to minimize the amount of personnel time spent in a radiation area 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 highly radioactive fluids do not pass 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 the 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 to the extent practical for utilizing movable shielding for sources within the service area when required.

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7. Providing means to control contamination and to facilitate decontamination of potentially contaminated areas.
8. Providing means for decontamination of service areas.
9. Providing space for pumps and valves outside of highly radioactive areas.
10. Providing labyrinth entrances to radioactive Pump, equipment, and valve rooms.
11. Providing adequate space in labyrinth entrances for easy access.

12.1.2.1.3 Westinghouse Design Considerations

The basic philosophy embodied in Westinghouse pressurized water reactor design considerations to ensure that occupational radiation exposures are ALARA, can be expressed as:

- A. Design of systems and components to ensure increased reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components.
- B. Design of systems and components to reduce the radiation fields to ensure that operation, maintenance, and inspection activities are performed in the minimum radiation field feasible.
- C. Design of systems and components to reduce the time spent in radiation fields during operation, maintenance, and inspection.
- D. Design of systems and components to accommodate remote and semi-remote operation, maintenance, and inspection procedures.

In translating this design philosophy into practice, Westinghouse calls upon experience from past designs operating in the field and upon other relevant field experience as well as laboratory tests. Many diverse sources of relevant field experience and data are used in implementing this design philosophy including: NRC publications (reference 1), the Atomic Industrial Forum (AIF) NESP studies (references 2 and 3), Electric Power Research Institute studies (reference 4), internal Westinghouse programs (reference 5), and personal communications with plant operators. Internal Westinghouse programs, which involve measuring and

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recording exposure and radiation level data in operating plants during operation, maintenance, and inspection, as well as personal communications with operating plant staffs, have been an invaluable source of feedback from operating plants. This information is assimilated and evaluated by the staff responsible for radiation protection functions. The feedback and radiation data from operating plants is used to construct models to predict occupational radiation exposure patterns for various operation, maintenance, and inspection activities on systems and components of Westinghouse nuclear steam supply systems. These models and exposure patterns are further described in section 12.4. From these models, the potential for improvements in areas such as reliability, repair time, and operational techniques related to occupational radiation exposures can be identified for further study.

Recommended design practice and design considerations are communicated to the system and component designers in three ways within Westinghouse WRD:

1. Consultation and personal communication
2. ALARA training programs
3. Design reviews.

The first of these communication techniques is an informal process employing open communication and sound engineering judgment. The second communication means (ALARA training programs) is a formal training session administered by the cognizant radiation protection personnel. The program is to be given to all engineers within NTD and other equipment divisions whose responsibility includes design of systems and components which may contain radioactive materials. The aim of the program is to provide design engineers with criteria, design features, operational guidelines, and operating plant experience relevant to radiation protection and the minimizing of occupational radiation exposures. Thus, the designer has an awareness of the field conditions and problems imposed by a radiation environment on the operation, maintenance, and inspection of systems and components. The final communication method listed above (design reviews) is the final check by the radiation protection specialists of the system and component designs to ensure that occupational radiation exposures will be ALARA. These design reviews are conducted coincident with the safety design review on systems and components required by 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Many of the inherent design features and policy considerations to ensure that occupational radiation exposures will be ALARA throughout their operating lives are also applicable during the

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eventual decommissioning of the plants. Westinghouse has been active in studies by AIF (reference 6) and NRC (reference 7) to define the impact of decommissioning by providing relevant information to the principal investigators. Westinghouse will continue to be aware of decommissioning impacts in the design of systems and components. The policy considerations outlined above ensure that occupational radiation exposures are ALARA in compliance with Regulatory Guides 8.8 and 8.10 and 10 CFR Part 20 (section 20.1(c)).

12.1.3 OPERATIONAL CONSIDERATIONS

Radiological health and safety procedures will be developed and continually reviewed to assure that occupational radiation exposures are ALARA. These procedures will be based on experience gained from operating plants. Improvements suggested during operation will be incorporated and implemented to continually update the program to maximize protection of the workers.

The ALARA operational plans and procedures will influence the design of the facility. Wherever possible, structures, equipment, and shielding minimize the duration of work in a radiation area and reduce the radiation levels in areas requiring normal access. Design considerations are based on data and reports from operating plants as discussed in subsection 12.1.2.

The manager, Radiological Control Section, will execute and supervise the ALARA procedures. Plant management will plan, direct, and conduct operations so that workers are properly safeguarded at all times. All personnel assigned to a specific task will have the necessary training required to safely execute their specific plant assignments.

This organizational and design structure promotes feedback to, and timely response from, the ALARA program. As such, it satisfies the intent of Regulatory Guides 8.8 and 8.10.

12.1.3.1 General ALARA Techniques

Described below are several general ALARA techniques. Further information on ALARA techniques incorporated into procedures is given in section 12.5.

- A. Permanent shielding is used, where possible, with workers behind walls or in low-level radiation areas when not actively working in high radiation areas. Temporary shielding, such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment, is used in some areas. Temporary

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shielding is used only if the total exposure, which includes exposure received during installation and removal will be effectively reduced.

- B. Systems and equipment which are subject to crud buildup, such as chemical and volume control 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 whole body exposure rates in excess of 100 mrem/hr or removable contamination levels in excess of 10^6 dpm/100 cm² will be preplanned so the job can be performed safely with a minimum of personnel exposure.
- D. On complex jobs or jobs with exceptionally high radiation levels, dry-run training will be used, and in some cases mock-ups will be 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.
- E. The work, as much as possible, is performed outside of the radiation areas. This includes reading instruction manuals or maintenance procedures, adjusting tools or jigs, repairing valve internals and prefabricating components.
- F. For repair jobs of long duration, consideration will be given to setting up a communications network such as sound powered telephones or closed circuit television to assist supervising personnel in checking on work progress from a lower radiation area.
- G. Special tools or jigs will be used when their use would permit the job to be performed more efficiently or 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 radiation source to the worker, thereby reducing the exposure received. These special tools will be used

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only if the total exposure, including that received during installation and removal, is significantly reduced.

- H. Access control points will be established in low-level radiation areas because personnel may spend a significant amount of time in these areas changing protective clothing and respiratory equipment. These access points are set up to limit the spread of contamination to as small an area as possible.
- I. Protective clothing and respiratory equipment are selected to minimize the discomfort of workers and increase efficiency so that less time is spent in radiation areas. The protective clothing is prescribed by health physics commensurate with the hazards involved and the requirements cannot be modified by other personnel.
- J. Contamination containments, i.e., glove bags, poly bottles, tents, etc., are used where practicable to allow personnel to work on highly contaminated equipment while minimizing the spread of contamination.
- K. Individuals will be instructed to remain in low-level radiation areas as much as possible, consistent with performing their assigned jobs. On certain jobs, detailed maps will be provided with the Radiation Work Permit to clearly delineate areas of high radiation levels to prevent inadvertent entry into such areas 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 the radiation levels are unusually high, a timekeeper will monitor the total exposure time using a stopwatch or similar device. This will ensure that personnel do not exceed the limits on time spent in a radiation field and thereby exceed applicable dose limits.
- N. On major maintenance jobs in high-level radiation areas, the job preplanning will include man-rem exposure estimates for the job. At the completion of the work, a debriefing session will be held 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 man-rem expended, will be

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compiled and filed for future reference. All radiation aspects, i.e., radiation, contamination, airborne radio-activity, and personnel contamination (external and internal), will be compiled and filed for future reference during preplanning of similar work situations.

12.1.3.2 Specific ALARA Considerations for Steam Generator Repair

The techniques in paragraph 12.1.3.1 are generally used when inspecting and plugging steam generator tubes.

After access to the steam generator primary side is obtained, covers are installed, if practical, over the hot and cold leg nozzle openings with a layer of material (such as canvas or plastic) to prevent tools and debris from entering the pipes, thus reducing exposure by aiding in the cleanup. If personnel are required to work inside the steam generator primary side for a significant amount of time, consideration is given to adding temporary shielding. Normally, however, this is not worthwhile since at least half the exposure is due to shine from crud inside the steam generator tubes which cannot be easily shielded because the tube sheet area must normally be kept clear for inspection or tube plugging. A remotely operated eddy current probe positioning device can be used to help locate tubes which should be plugged. Tubes are then plugged using explosive welding techniques rather than conventional tungsten inert gas welding techniques, thus reducing time inside the channel heads.

12.1.3.3 Specific ALARA Considerations for Reactor Head Removal and Installation

Techniques described in paragraph 12.1.3.1 are normally used when removing and installing the reactor head. Quick disconnect electrical cables and reactor head ventilation ducts which can be quickly removed are used to reduce time spent in radiation areas. Temporary shielding can be installed around the outer control rod drive mechanisms to reduce exposure if crud collects in the control rod drive housing. A temporary shield may be provided in the refueling cavity, if necessary, for personnel shielding when they are not actively working. Communications are provided up to the refueling floor so that personnel can communicate with supervision, thus reducing lost time and exposure to workers and supervisors.

12.1.3.4 Specific ALARA Considerations for Inservice Inspections

Techniques in paragraph 12.1.3.1 are normally used when performing inservice inspections. Remote testing devices will be

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used in the conduct of the examinations, where applicable. Written and possibly photographic or videotape records will be made of preservice inspection operations that have potential for future significant radiation exposure to personnel. By training the examiners and alerting them to the specific problems that they can expect to encounter, less time will be spent in radiation areas.

12.1.3.5 Specific ALARA Considerations for Other Operations Involving Radiation Exposure

Other operations such as refueling, radwaste handling, spent fuel handling, loading and shipping, routine maintenance, sampling, and calibration are discussed in section 12.5.

12.1.4 REFERENCES

1. "Occupational Radiation Exposure at Light Water Cooled Power Reactors," U.S. Nuclear Regulatory Commission, NUREG-0323, March 1978.
2. "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum, AIF/NESP-005, September 1974.
3. "Potential Benefits of Reducing Occupational Radiation Exposure," Atomic Industrial Forum, AIF/NESP-010, May 1978.
4. "Primary System Shutdown Radiation Levels at Nuclear Power Generating Stations," Electric Power Research Institute, EPRI-404-2, December 1975.
5. Brown, W.S., "Westinghouse Electric Corporation Efforts to Maintain Occupational Radiation Exposures ALARA," presented at ACRS Environmental Subcommittee Meeting, January 25, 1978.
6. "An Engineering Evaluation of Nuclear Power Plant Decommissioning Alternatives," Atomic Industrial Forum, AIF/NESP-009, November 1976.
7. "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor," Battelle Pacific Northwest Laboratory, NUREG/CR-0130, June 1978.

12.2 RADIATION SOURCES

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

12.2.1 CONTAINED SOURCES

The shielding design source terms are based upon the three general plant conditions of normal full power operation, shutdown, and design basis events.

12.2.1.1 Sources for Normal Full Power Operation

The main sources of activity during normal full power operation are nitrogen-16 (N-16) from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products. All shielding has its design basis as the maximum case of clad defects in fuel rods producing 1.0 percent of core thermal power.

Each plant system is shielded according to the amount of activity present, adjacent zoning and access criteria. These systems include:

- Reactor coolant system (RCS)
- Main steam supply system
- Chemical and volume control system (CVCS)
- Boron recycle system (BRS)
- Radwaste processing system
- Steam generator blowdown system (SGBS)
- Spent fuel pool cooling and cleanup system (SFPCCS).

For all systems transporting or containing radioactive materials, conservative allowance is made for decay in transit or storage and daughter product formation is considered wherever experience has shown this contribution to be significant.

The design sources in this chapter are presented by building location and system. General location of the equipment discussed in this chapter is shown in figures provided in section 12.3. Equipment list is shown in figure 1.2-11.

12.2.1.1.1 Containment

12.2.1.1.1.1 Reactor Core. The primary radiation within the containment during normal operation are neutrons and gamma rays emanating from the reactor core. Tables 12.2-1 and 12.2-2 list neutron and gamma multigroup fluxes at several locations outside reactor vessel during full power operation.

Full power radiation sources in the reactor cavity are used to design the primary shield. Typical data based on an 8-inch reactor vessel - primary concrete annulus include:

- A. Neutron particle fluxes at the inside surface of the primary shield concrete at the core midplane (shown in figure 12.2-1)
- B. Gamma ray energy fluxes at the inside surface of the primary shield concrete at the core midplane (shown in figure 12.2-2)
- C. Gamma ray dose rates at the inside surface of the primary shield concrete at the core midplane (shown in figure 12.2-3).

12.2.1.1.1.2 Reactor Coolant System. Sources of radiation in the RCS are fission products released from fuel and activation and corrosion products which are circulated in the reactor coolant.

The activation product N-16 is the predominant activity in the reactor coolant pumps, steam generators, and reactor coolant piping. The N-16 activity in each of the components depends upon the total transit time to the component and is listed in table 12.2-3 for a number of locations in the RCS.

The reactor coolant equilibrium fission and corrosion product activities are discussed in section 11.1 and are listed in table 11.1-2. The pressurizer activity for the liquid and vapor volumes are given in table 12.2-4.

The isotopic composition and specific activity of typical out-of-core crud deposits are given in table 12.2-5. Typically, 1 milligram of deposited crud material is found on one square centimeter of a relatively 'smooth surface. This may be as much as 50 times higher in crud trap areas. Crud trap areas are generally locations of high turbulence, areas of high momentum change, gravitational sedimentation areas, high affinity material areas, and possibly thin boundary layer regions.

12.2.1.1.1.3 Main Steam Supply System. Radioactivity in the main steam supply system is based on a steam generator tube leakage rate of 100 lbm/d concurrent with 1.0 percent fuel cladding defects. Partition factors for activity into the steam system from the reactor coolant are 100 percent for noble gases, 1 percent by weight for halogens, and 0.1 percent by weight for particulates. Activity concentrations for the steam generator liquid and steam are presented in table 11.1-8. The model used to calculate activities in the secondary system is given in paragraph 11.1.1.5.

12.2.1.1.1.4 Processing Systems

12.2.1.1.1.4.1 Steam Generator Blowdown Processing System. Radiation sources in the steam generator blowdown processing system (subsection 10.4.8) are based on the assumptions stated in subparagraph 12.2.1.1.1.3. The steam generator secondary side liquid can be processed through the steam generator blowdown processing system at a maximum rate of 3.78×10^5 lbm/hr total for the three steam generators. Design blowdown liquid concentrations are presented in table 11.1-8.

12.2.1.1.1.4.2 Chemical and Volume Control System. Radiation sources in the CVCS consist of those radioisotopes carried in the reactor coolant, discussed in subparagraph 12.2.1.1.1.2. Nitrogen-16 is the predominant radiation source in the excess letdown and regenerative heat exchangers. The design of the letdown system ensures that most of the N-16 has decayed before the letdown stream leaves the containment. The reactor coolant will be delayed at least 60 seconds for the normal letdown path at 120 gal/min or 48 seconds for the excess letdown path before exiting the containment. All CVCS heat exchangers other than the excess letdown and regenerative heat exchangers are located in the auxiliary building.

The activity in each component of the letdown system depends upon detailed equipment and system design. The shielding design is based on the maximum expected activity in each component.

12.2.1.1.2 Auxiliary Building/Control Building

12.2.1.1.2.1 Chemical and Volume Control System. The CVCS source activity is the reactor coolant inventory with 1 percent defective fuel cladding. The N-16 coolant activity decays before the letdown line exits the containment and therefore is not significant in determining shielding requirements for the CVCS equipment outside the containment.

The major CVCS equipment items include the regenerative and letdown heat exchangers, mixed bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The boron thermal regeneration (BTR) subsystem contains the three BTR heat exchangers and the BTR demineralizers. The seal water subsystem for the reactor coolant pumps includes the injection and return filters, and the seal water heat exchanger. The reactor coolant purity control subsystem includes the concentrated boric acid polishing demineralizer and filter, the recycle evaporator sample pre-conditioning demineralizer, and the concentrated boric acid sample cooler. From a radiation shielding standpoint, only the polishing demineralizer and its filter will concentrate or process significant amounts of radioactive material. The design source strengths for CVCS fluids and components are listed in tables 12.2-7 through 12.2-10.

The letdown heat exchanger provides second-stage cooling for the reactor coolant prior to entering the demineralizers. The activity at this point is identical to the letdown coolant source outside the containment.

The mixed bed retains the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation bed can be used intermittently to remove lithium for pH control, and supplements the mixed bed in removing Y, Cs, Mo, and the crud metals.

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

The BTR heat exchangers include the moderating, chiller, and letdown reheat units. The radiation sources in this equipment are modified to account for activity removed by the demineralizers upstream of the units.

The seal water heat exchanger cools the water from the reactor coolant pump seals.

The function of the reactor coolant purity control subsystem is to control the ingress of impurities (Mainly aluminum, calcium, magnesium, and silica) into the reactor coolant to lessen the severity of crud deposits on fuel cladding surfaces. The demineralizer is used to clean up the boric acid storage tanks initially and then to process recycle evaporator concentrates for the ten year service life of the demineralizer resin. The concentrated boric acid polishing filter is located downstream of the demineralizer and serves to maintain boric acid within specification for suspended solids and to trap resin fines from the demineralizer.

12.2.1.1.2.2 Control Building. There are no radioactive sources in the control building. The shielding requirements for the control room are dictated by the post-loss-of-coolant accident (LOCA) dose to control room personnel from direct gamma radiation from the containment, the external radioactive cloud, and control room airborne activity. The source terms are discussed in section 15.6.

12.2.1.1.3 Fuel Building

12.2.1.1.3.1 Spent Fuel Storage and Transfer. The predominant radioactivity sources in the spent fuel storage and transfer areas in the fuel building are the spent fuel assemblies.

Spent fuel assembly sources are discussed in paragraph 12.2.1.2. For shielding design, the spent fuel pool is assumed filled with spent fuel assemblies at 14-inch center to center spacing. The fuel storage facility is discussed in more detail in sub-section 9.1.1.

12.2.1.1.3.2 Spent Fuel Pool Cooling and Cleanup System. Sources in the SFPCCS are a result of transfer of radioactive isotopes from the reactor coolant into the refueling pool during refueling operations. The reactor coolant activities for fission, corrosion, and activation products (table 11.1-2) are decayed for the amount of time required to remove the reactor vessel head following shutdown, are reduced by operation of the letdown system filter and 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 SFPCCS filter and demineralizer. In addition to the SFPCCS, a refueling pool filtration system has been incorporated to increase the crud removal ability. The refueling pool filtration system is a once through system which returns the water to the refueling pool after filtration. Design spent fuel pool and refueling pool water activity is presented in paragraph 11.1.1.4.

Maximum activities for fuel storage pool cooling and cleanup system filter and demineralizer are based on the following assumptions:

- A. Fifty percent of reactor coolant is drained from the reactor coolant system prior to head removal.
- B. The reactor coolant is purified in the CVCS for 66 hours prior to removal of the reactor head.
- C. The reactor coolant activities at shutdown are the 100 percent power operating reactor coolant activities with 1 percent failed fuel for the design conditions.
- D. CVCS purification flow rate is 120 gpm.

- E. The volume of the refueling cavity is 300,000 gallons.
- F. The volume of the fuel storage pool is 460,000 gallons.
- G. The fuel pool purification system purifies 150 gpm from the refueling cavity and 150 gpm from the fuel pool. The purified 300 gpm is returned to the fuel storage pool and 150 gpm flows from the fuel storage pool through the fuel transfer tube to the refueling cavity.

Activity buildup on the SFPCCS filters and demineralizer is presented in table 11.2-7.

12.2.1.1.4 Turbine Building

12.2.1.1.4.1 Main Steam Supply and Power Conversion Systems.

Potential radioactivity in the main steam supply and power conversion systems is a result of steam generator tube leaks and fuel cladding defects. This radioactivity is sufficiently low that no radiation shielding for steam or condensate lines is required. However, provisions are made for adding temporary shielding for the condensate polishing demineralizers if operating experience dictates the need. The polishing demineralizer activity is presented in paragraph 11.2.2.6.

The design activity for the condensate demineralizer is based on the following assumptions:

- A. 4100 gpm condensate flow rate through each condensate demineralizer.
- B. The condensate activity is the same as the main steam activity, except noble gases.
- C. The steam generator blowdown rate is 248 gpm.
- D. 100 lb/day primary to secondary leakage.
- E. Steam generator blowdown filter decontamination factor (DF)

DF = 1000 for crud
DF = 1 for others.
- F. The activity of the condensate is based on 1 percent failed fuel.
- G. Steam generator partition factor (PF)

PF = 0.001 for nonhalogen
PF = 0.01 for halogen
PF = 1 for inert gas.
- H. 28 days between regenerations.

12.2.1.1.5 Radwaste Building

12.2.1.1.5.1 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 concentrations of radionuclides present in the process fluids at various locations in the radwaste systems such as pipes, tanks, filters, demineralizers, and evaporators are discussed in section 11.2. These nuclide concentrations will be used in the final shielding design. Shielding for each component of the radwaste systems is based on conservative activity conditions as given in sections 11.1, 11.2, and 11.4.

12.2.1.1.5.2 Gaseous Radwaste System. Radiation sources for each component of the waste gas system are based on operation under the maximum activity conditions as given in sections 11.1 and 11.3. Tabulation of the maximum activities is shown in table 11.3-4.

The assumptions used in the calculation of the component inventories are as follows:

- A. Normal flow rate through GRS is mainly from the volume control tank (VCT) hydrogen purge at a rate of 0.7 std cfm per plant.
- B. GRS inputs from other sources result in an additional average flow rate of 0.1 std cfm per plant.
- C. Input gaseous activity concentration is VCT vapor activity listed in table 11.3-2.
- D. The carrier gas is at ambient temperature during passage through the charcoal beds.
- E. There are two shutdown degasifications per year.

12.2.1.1.5.3 Boron Recycle System. The radiation sources in the BRS are listed in tables 12.2-12 through 12.2-15. The major equipment items included in this system are the recycle holdup tanks and the recycle evaporator with its associated equipment, i.e., feed demineralizers and filter, condensate demineralizer and filter, and concentrates filter. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction. The evaporator feed demineralizers are located upstream of the holdup tanks and contain mixed bed resins which remove non-gaseous activity from the reactor coolant directed to the

holdup tanks. A decontamination factor of 10 across these beds is taken for all particulate activity.

The evaporator condensate demineralizer is charged with anion. resin to remove any boron and iodine activity which may be carried over with the evaporator condensate.

The recycle holdup tanks are each equipped with a diaphragm. Gases which flash from the reactor coolant letdown to the holdup tanks are retained under the diaphragm until approximately 500 cubic feet of gas has accumulated; the gases are then removed to the waste gas system. The radiation sources in the holdup tanks are based on 50 percent of the gaseous activity flashing into the vapor phase.

The recycle evaporator feed filter and condensate filter are located downstream of their respective demineralizers, and serve to retain particulates and any resin fines which may escape from the demineralizers.

The sources for the feed filter correspond to a radiation level of 100 rem/h, contact. The condensate filter sources result in levels of less than 1 rem/h, contact. The maximum activity of the liquid concentrates in the recycle evaporator is 40 μ Ci/g. The resultant radiation sources on the concentrates filter correspond to an exposure rate of approximately 3 rem/h.

142 | 12.2.1.1.5.4 Stored Radioactivity. The principal sources of activity not stored inside the plant structures are the reactor makeup water storage tank (RMWST), the refueling water storage tank (RWST) and the condensate storage tank (CST). The RMWST and CST are expected to contain concentrations of radionuclides which yield a surface dose rate of 0.5 mrem/h or less. The RWST is expected to have a maximum contact dose rate of less than 10mrem/h when the water is returned from the refueling pool. Radioactivity levels in the RWST can be reduced by processing through the spent fuel pool purification filter and demineralizer. 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 the spent fuel shipping cask for transport offsite. Storage space is allocated in the radwaste building for storage of spent filter cartridges and solidified spent resins, evaporator bottoms, and chemical wastes.

Additional long term solid waste storage capacity is provided in the drum storage building located adjacent to the radwaste building. Radioactive wastes stored inside plant structures are shielded such that there is Zone 1 access outside the structure. If it becomes necessary to temporarily store radioactive wastes outside plant structures, adequate radiation protection measures are taken by the health physics staff.

12.2.1.1.5.5. Field Run Pipe Routing. All radioactive process piping of 2 inch and larger diameter is run and shielded by Bechtel. Field routing of smaller lines will be done by providing for field engineering with criteria for routing these lines.

12.2.1.2 Sources for Shutdown

In the reactor shutdown condition, the only additional significant sources requiring shielding consideration are the spent reactor fuel, the residual heat removal system (RHRS), and the incore detector system. Individual components may require shielding during shutdown due to deposited crud material. Estimates of accumulated crud are given in table 12.2-5. The radiation sources in the reactor coolant system, CVCS, BRS, are bounded by the sources given for full power operation with the exception of a short time period (i.e., less than 24 hours) following shutdown during which the fission product spiking phenomena and crud bursts can result in increased radiation sources. The spiking phenomena involves the release of a portion of the accumulated water soluble salts (e.g., iodine and cesium) and gases (e.g., xenon and krypton) from the interior cladding surface of defected fuel rods during the shutdown and coolant depressurization (reference 1). Crud bursts are the resuspension or solubilization of a portion of the accumulated deposited corrosion products into the RCS during shutdown such as during oxygenization of the reactor coolant. However, special shielding considerations to accommodate these increases should be unnecessary due to several factors including:

- A. The spike or crud burst release is of short duration (generally less than 6 hours).
- B. The CVCS is generally in operation at full reactor coolant purification capability during shutdown (120 gal/min).

The maximum gamma ray source strengths in the RHRS are given in table 12.2-16 for 4 and 8 hours after reactor shutdown. The RHRS is placed into operation at 4 hours following a shutdown at the maximum shutdown rate. The system removes decay heat from the reactor for the duration of the shutdown. The sources given are maximum values with credit for 4 and 8 hours of fission and corrosion product decay and purification. Core average gamma ray source strengths are given in table 12.2-17. These source strengths are used in the evaluation of radiation levels within and around the shutdown reactor. These sources are based on a three region core with the regions operated at 288, 575, and 863 effective full power days, respectively. Spent fuel gamma ray source strengths are given in table 12.2-18.

Shielding requirements for spent fuel transfer are based on the fission product activity present 100 hours after shutdown to conservatively take credit for the time elapsed prior to the initiation of refueling operations. Shielding calculations for a spent fuel assembly assume the radial and axial peaking factors are both 1.65 which are characteristic of the highest rated discharged assembly.

These source strengths are used in the evaluation of radiation levels for spent fuel handling, storage, and shipping. These are based on one core region operated for 900 effective full power days. All of these sources may be put on a per unit volume of homogenized core basis by multiplying by the power density of 109.2 watts per cubic centimeter. Core average and spent fuel neutron source strengths are given in table 12.2-19.

The only source materials, byproduct materials, or special nuclear materials requiring shielding consideration for the Westinghouse nuclear steam supply system are the neutron source materials used in the primary and secondary source rods. Each of the two primary source rods contains a californium-252 spontaneous fission neutron source which emits approximately 6×10^8 neutrons per second when initially placed in the reactor core. The gamma ray and neutron source strengths for the secondary source rods, irradiated for 400 days, are given in table 12.2-20.

The absorber materials used in the hybrid control rods are boron carbide (B_4C) and silver-indium-cadmium (Ag-In-Cd). The gamma ray source strengths associated with the Ag-In-Cd absorber are listed in table 12.2-21 for various times after shutdown. The values are based on an irradiation period of 4 years. There are no significant gamma ray sources associated with the B_4C absorber.

The absorber material used in the burnable poison rods is borosilicate glass. There are no significant gamma ray sources associated with this absorber.

The material used for the control rod cladding, primary source rod cladding, secondary source rod cladding, and burnable poison rod cladding and inner sheath is type-304 stainless steel with a maximum cobalt content of 0.12 weight %. The gamma ray source strengths associated with the stainless steel are listed in table 12.2-22 for various times after shutdown. The values are based on an irradiation time of 15 years.

Irradiated incore detector and drive cable maximum gamma ray source strengths are given in table 12.2-23. These source strengths are used in determining shielding requirements when detectors are being moved during or following a flux mapping of the reactor core. These source strengths are given for a detector irradiation period of 30 days and a drive cable irradiation period of 400 days and are given in terms of per cubic centimeter of detector and drive cable. Irradiated incore detector drive cable average gamma ray source strengths are given in table 12.2-24. These source strengths are used in determining shielding requirements when the detectors are not in use and for shipment when the detectors have failed. The values are given in terms of per cubic centimeter of drive cable after an irradiation period of 400 days. Irradiated

in-core flux thimble gamma ray source strengths are given in table 12.2-25. These source strengths are used in determining shielding requirements during refueling operations when the flux thimbles are withdrawn from the reactor core. The values are given in terms of per cubic centimeter of stainless steel for an irradiation period of 15 years. The flux thimbles are made of type-316 stainless steel with a maximum cobalt impurity content of 0.12 weight %.

12.2.1.3 Sources for Design Basis Events

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,

12.2.1.3.1 Recirculation Loop Sources

The fission product sources considered to be released to the containment building sump following the loss-of-coolant accident are based on the assumptions listed below:

Power level: 2,900 MWt

Three-region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.02 MW/MTU for 288, 575, and 863 EFPD, respectively.

Sump water volume:

Reactor coolant volume (ft ³) ^[a]	8,910
Refueling water volume (ft ³) ^[b]	40,000

Fraction of total core fission product inventory released^[c]:

Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01

Cleanup rate following accident: 0.0

The sources for which system radiation levels are calculated are based on a maximum credible accident, which assumes that 100 percent of the noble gases, 50 percent of the halogens,

a. Water at 590°F and 2,250 psia.

b. Water at S.T.P.

c. DiNunno, J. J., "Calculation of Distance Factors for Power and Test Reactor Sites." TID-14844, March 1962.

and 1 percent of the remaining core fission products are released to the reactor coolant system (RCS) and then to the containment. These assumptions and source terms are consistent with those identified in reference 2, item II.B.2. The nongaseous activity is then assumed to be transferred to the sump water, which flows into the components associated with the safety injection system and containment spray system during recirculation. The noble gases formed by decay of the halogens in the sump water are assumed to be released to the containment and not retained in the water.

Source strengths of the radiation sources circulating in the residual heat removal loop and associated equipment are given in table 12.2-26.

12.2.1.3.2 Control Room Sources

The shielding requirements for the control room are dictated by the post-LOCA dose to control room personnel from direct gamma radiation from the containment, the external radioactive cloud, and control room airborne activity. The source terms are discussed in section 15.6 and appendix 15A.

12.2.1.3.3 Other Sources

Accident parameters and sources are further discussed and evaluated in section 15.6.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

This subsection deals with the models, parameters, and sources required to evaluate airborne concentrations of radionuclides during plant operations in various plant radiation areas where personnel occupancy is expected. The airborne activities are listed in table 12.4-6. These sources include the concentrations of radionuclides in the turbine, radwaste, auxiliary, and containment areas. The assumptions and parameters required to evaluate the isotopic airborne concentrations in the various regions are listed in table 11.3-7. The final design of the plant will ensure that all the expected airborne isotopic concentrations in all accessible regions will be Flow maximum permissible concentration (MPC) for the critical organ for the appropriate isotope for occupational workers adjusted on the basis of expected weekly occupancy in the regions.

12.2.2.1 Model for Calculating Airborne Concentrations

For those regions which are characterized by a constant leak rate of the radioactive source at constant source strength and a constant exhaust rate, the peak or equilibrium airborne

concentration of the radioisotope in the regions can be calculated using the following equation:

$$C_i(t) = (LR)_i A_i (PF)_i (1 - e^{-\lambda_i t}) / V \lambda_i \quad (1)$$

Where :

$(LR)_i$ = leak or evaporation rate of the i^{th} radioisotope in g/s, in the applicable region

and

A_i = activity concentration of the i^{th} leaking or evaporating radioisotope in $\mu\text{Ci/g}$

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

λ_i = total removal rate constant for the i^{th} radioisotope in s^{-1} from the applicable region

$$= (\lambda_{di} + \lambda_{ei})$$

(λ_{di} and λ_{ei} are the removal rate constants in s^{-1} due to radioactive decay and the ventilation 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 cm^3

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

$$C_{Eqi} = (LR)_i A_i (PF)_i / V \lambda_i \quad (2)$$

with high exhaust rates, this peak concentration will be reached within a few hours.

Parameters and assumptions for calculating airborne radioactivity concentrations are shown in tables 11.3-7 and 11.3-8

12.2.3 REFERENCES

1. Lutz, R. J. and Chubb, W., "Iodine Spiking - Cause and Effect," ANS Transactions, Vol. 28, p. 649, June 1978.
2. NUREG-0737, "Clarification of TMI Action Plan Requirements," Enclosure 3, Item II.B.2, November 1980.



Table 12.2-1

MAXIMUM NEUTRON SPECTRA OUTSIDE REACTOR VESSEL
(Sheet 1 of 2)

Upper Energy Limit	Maximum Neutron Spectra ^(a)
ev	Neutrons/cm ² -S
15.0 x 10 ⁶	1.15 x 10 ⁶
12.2 x 10 ⁶	4.39 x 10 ⁶
10.0 x 10 ⁶	1.05 x 10 ⁷
8.18 x 10 ⁶	2.23 x 10 ⁷
6.36 x 10 ⁶	4.03 x 10 ⁷
4.96 x 10 ⁶	3.35 x 10 ⁷
4.06 x 10 ⁶	6.14 x 10 ⁷
3.01 x 10 ⁶	7.98 x 10 ⁷
2.46 x 10 ⁶	2.71 x 10 ⁷
2.35 x 10 ⁶	1.62 x 10 ⁸
1.83 x 10 ⁶	5.99 x 10 ⁸
1.11 x 10 ⁶	1.99 x 10 ⁹
5.5 x 10 ⁵	6.34 x 10 ⁹
1.11 x 10 ⁵	4.64 x 10 ⁹
3.35 x 10 ³	1.40 x 10 ⁹
5.82 x 10 ²	1.26 x 10 ⁹
1.01 x 10 ²	8.62 x 10 ⁸
2.90 x 10 ¹	5.62 x 10 ⁸
1.07 x 10 ¹	6.13 x 10 ⁸
3.06	4.31 x 10 ⁸

Table 12.2-1

MAXIMUM NEUTRON SPECTRA OUTSIDE REACTOR VESSEL
(Sheet 2 of 2)

Upper Energy Limit	Maximum Neutron Spectra ^(a)
ev	Neutrons/cm ² -S
1.12	3.55 x 10 ⁸
0.414	1.66 x 10 ⁹

- (a) The maximum neutron fluxes at the side of the Reactor Vessel are given for a point at the outer boundary of reactor vessel, and at a position adjacent to maximum axial power.

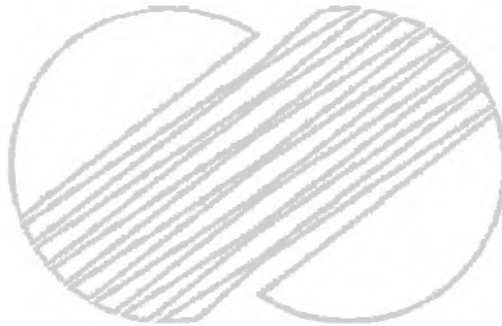


Table 12.2-2

MAXIMUM GAMMA SPECTRA OUTSIDE REACTOR VESSEL

Upper Energy Limit	Maximum Gamma Spectra ^(a)
Mev	Gammas/cm ² -S
10.0	1.78×10^8
8.0	7.47×10^8
6.5	5.86×10^8
5.0	5.21×10^8
4.0	7.01×10^8
3.0	4.41×10^8
2.5	7.21×10^8
2.0	5.56×10^8
1.66	6.18×10^8
1.33	7.94×10^8
1.0	6.06×10^8
0.8	7.92×10^8
0.6	1.95×10^9
0.4	1.18×10^9
0.3	1.88×10^9
0.2	2.42×10^9
0.1	1.80×10^8
0.05	1.19×10^6

- (a) The maximum gamma fluxes at the side of the Reactor Vessel are given for a point at the outer boundary of reactor vessel, and at a position adjacent to maximum axial power.

Table 12.2-3

RADIATION SOURCES
REACTOR COOLANT NITROGEN-16 ACTIVITY

Position in Loop	Loop Transit Time (sec)	Nitrogen-16 Activity (microcuries/g)
Leaving core	0.0	217
Leaving reactor vessel	1.2	193
Entering steam generator	1.5	187
Leaving steam generator	5.5	127
Entering reactor coolant pump	6.1	120
Entering reactor vessel	6.8	112
Entering core	7.9	110
Leaving core	8.7	217
<u>Nitrogen-16 Energy Emission</u>		
<u>Energy (Mev/gamma)</u>	<u>Intensity (%)</u>	
1.75	0.13	
2.74	0.76	
6.13	69.0	
7.12	5.0	

Table 12.2-4

PRESSURIZER ACTIVITY

Liquid Phase (84 ft ³)	
Energy Group (Mev/gamma)	Specific Source Strength (Mev/gm-sec)
0.20 - 0.40	4.5 x 10 ^{5(a)}
0.40 - 0.90	6.3 x 10 ⁵
0.90 - 1.35	2.9 x 10 ⁵
1.35 - 1.80	1.7 x 10 ⁵
1.80 - 2.20	1.7 x 10 ⁵
2.20 - 2.60	1.8 x 10 ⁵
2.60 - 3.00	2.2 x 10 ⁴
3.00 - 4.00	9.0 x 10 ³
Vapor Phase (560 ft ³)	
Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -sec)
0.20 - 0.40	1.2 x 10 ^{6(a)}
0.40 - 0.90	1.1 x 10 ⁵
0.90 - 1.35	2.4 x 10 ³
1.35 - 1.80	4.2 x 10 ³
1.80 - 2.20	6.3 x 10 ³
2.20 - 2.60	1.3 x 10 ⁴
2.60 - 3.00	6.9 x 10 ¹

a. Includes 80 kev Xenon-133

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YGN 1 & 2 FSAR

RADIATION SOURCES

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Table 12.2-5

ISOTOPIC COMPOSITION AND SPECIFIC ACTIVITY OF
TYPICAL OUT-OF-CORE CRUD DEPOSITS ^(a)

Composition (Nuclide)	Activity (microcuries per milligram) of Deposited Crud for Effective Full Power Years of Plant Operation			
	1 Year	2 Years	5 Years	10 Years
Mn-54	1.0	1.1	1.3	1.4
Fe-59	0.5	0.5	0.5	0.5
Co-58	12.0	12.0	12.0	12.0
Co-60	1.5	2.3	4.0	6.0

- a. In addition to corrosion products, about 1.0 microgram of mixed actinides and fission products may be present for each 1 gram of deposited crud.

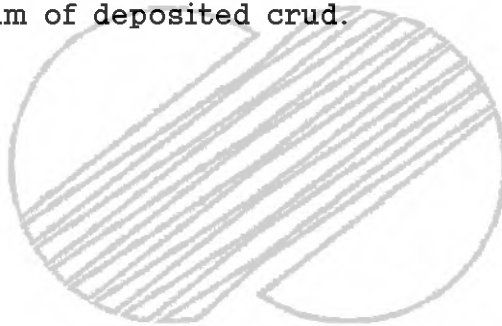


Table 12.2-6

LETDOWN COOLANT SOURCE STRENGTH
(OUTSIDE THE CONTAINMENT)

Energy Group (Mev/gamma)	Specific Source Strength (Mev/g-s)
0.2 - 0.4	$4.5 \times 10^{5(a)}$
0.4 - 0.9	6.3×10^5
0.9 - 1.35	2.9×10^5
1.35 - 1.8	1.7×10^5
1.8 - 2.2	1.7×10^5
2.2 - 2.6	1.8×10^5
2.6 - 3.0	2.2×10^4
3.0 - 4.0	9.0×10^3

a. Includes 80 kev Xe-133

Table 12.2-7

CHEMICAL AND VOLUME CONTROL SYSTEM DEMINERALIZER SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -s)			
	Mixed Bed Demineralizer (30 ft ³ of Resin)	Caution Bed Demineralizer (30 ft ³ of Resin)	Boron Thermal Regeneration Demineralizers (150 ft ³ of Resin)	Concentrated Boric Acid Polishing Demineralizer (30 ft ³ of Resin)
0.2 - 0.4	1.4×10^8	6.0×10^6	2.2×10^6	—
0.4 - 0.9	6.2×10^8	5.2×10^8	1.6×10^6	6.3×10^5
0.9 - 1.35	6.5×10^7	3.5×10^7	4.2×10^5	—
1.35 - 1.8	2.3×10^7	1.2×10^7	2.3×10^5	—
1.8 - 2.2	1.4×10^6	5.8×10^5	2.0×10^4	—
2.2 - 2.6	7.9×10^5	2.0×10^5	1.3×10^4	—
2.6 - 3.0	2.0×10^5	1.9×10^5	—	—
3.0 - 4.0	5.9×10^4	4.6×10^4	—	—

Table 12.2-8

VOLUME CONTROL TANK SOURCE STRENGTH ^(a)

Specific Source Strengths		
Energy Group (Mev/gamma)	Vapor Phase (Mev/cm ³ -s)	Liquid Phase (Mev/g-s)
0.2 - 0.4	5.4×10^{-6} ^(b)	3.8×10^{-5} ^(b)
0.4 - 0.9	4.7×10^{-5}	3.0×10^{-5}
0.9 - 1.35	1.1×10^{-5}	1.7×10^{-5}
1.35 - 1.8	3.4×10^{-5}	7.0×10^{-4}
1.8 - 2.2	6.0×10^{-5}	1.1×10^{-5}
2.2 - 2.6	1.2×10^{-6}	7.2×10^{-4}
2.6 - 3.0	6.9×10^{-3}	2.1×10^{-4}
3.0 - 4.0	3.4×10^{-3}	6.1×10^{-3}

- a. These sources correspond to a nominal operating level in the tank at 200 ft³ in the vapor phase and 200 ft³ in the liquid phase.
- b. Includes 80 keV Xe-133

Table 12.2-9

CHEMICAL AND VOLUME CONTROL SYSTEM FILTER ^(a) SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -s)			
	Reactor Coolant Filter	Seal Water Injection Filter	Seal Water Return Filter	Concentrated Boric Acid Polishing Filter
0.4 - 0.9	5.7×10^{-7}	4.8×10^{-7}	1.1×10^{-7}	6.3×10^{-5}
0.9 - 1.35	1.5×10^{-7}	1.2×10^{-7}	3.0×10^{-6}	

- a. Considered to be homogeneous sources with the following dimensions and compositions :

<u>Filter</u>	<u>Radius (in.)</u>	<u>Length (in.)</u>	<u>Source Composition (Vol %)</u>
Reactor coolant	3.375	19	62% air, 38% water
Seal water return	3.375	19	62% air, 38% water
Seal water injection	1.3	20	90% air, 10% stainless steel
Concentrated Boric acid polishing	3.375	19	67% air, 33% water

Table 12.2-10

CHEMICAL AND VOLUME CONTROL SYSTEM HEAT EXCHANGER SOURCE STRENGTH

Energy Group (Mev/gamma)	Source Strength (Mev/gram-s)				
	Regenerative Heat Exchanger		Excess Letdown Heat Exchanger	Letdown Heat Exchanger	Seal Water Heat Exchanger
	Tube side	Shell side	Tube side	Tube side	Tube side
0.2 - 0.4	3.8×10^5	4.5×10^5	4.5×10^5	4.5×10^5	4.5×10^5
0.4 - 0.9	3.0×10^5	6.3×10^5	6.3×10^5	6.3×10^5	6.3×10^5
0.9 - 1.35	1.7×10^5	2.9×10^5	2.9×10^5	2.9×10^5	2.9×10^5
1.35 - 1.8	7.0×10^4	1.8×10^5	1.8×10^5	1.7×10^5	1.7×10^5
1.8 - 2.2	1.1×10^5	1.7×10^5	1.7×10^5	1.7×10^5	1.7×10^5
2.2 - 2.6	7.2×10^4	1.8×10^5	1.8×10^5	1.8×10^5	1.8×10^5
2.6 - 3.0	2.1×10^4	1.2×10^5	1.2×10^5	2.2×10^4	2.2×10^4
3.0 - 4.0	6.1×10^3	9.0×10^3	9.0×10^3	9.0×10^8	9.0×10^3
4.0 - 7.0	-	1.9×10^7	1.9×10^7	-	-
7.0 - 7.5	-	1.6×10^6	1.6×10^6	-	-

a. Includes 80 kev Xe-133.

Table 12.2-11

BORON THERMAL REGENERATION HEAT EXCHANGER SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength (Mev/gram-s)			
	Letdown Reheat Heat Exchanger		Letdown Chiller Heat Exchanger	Moderating Heat Exchanger
	Tube Side (0.85 ft ³)	Shell Side (1.12 ft ³)	Tube Side (9.5 ft ³)	Tube Side (9.8 ft ³) Shell Side (17.2 ft ³)
0.2 - 0.4	4.5×10^5 (a)	4.2×10^5 (a)	4.2×10^5 (a)	4.2×10^5 (a)
0.4 - 0.9	6.3×10^5	3.4×10^5	3.4×10^5	3.4×10^5
0.9 - 1.35	2.9×10^5	1.8×10^5	1.8×10^5	1.8×10^5
1.35 - 1.8	1.7×10^5	1.1×10^5	1.1×10^5	1.1×10^5
1.8 - 2.2	1.7×10^5	1.6×10^5	1.6×10^5	1.6×10^5
2.2 - 2.6	1.8×10^5	1.8×10^5	1.8×10^5	1.8×10^5
2.6 - 3.0	2.2×10^4	2.2×10^4	2.2×10^4	2.2×10^4
3.0 - 4.0	9.0×10^3	8.2×10^3	8.2×10^3	8.2×10^3

a. Include 80 kev Xe-133.

Table 12.2-12

BORON RECYCLE HOLDUP TANK SOURCE STRENGTH

Vapor Phase (500 ft ³)	
Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -s)
0.2 - 0.4	8.6×10^{-5} (a)
0.4 - 0.9	1.2×10^{-5}
0.9 - 1.35	3.7×10^{-4}
1.35 - 1.8	1.1×10^{-5}
1.8 - 2.2	1.9×10^{-5}
2.2 - 2.6	3.6×10^{-5}
2.6 - 3.0	3.7×10^{-3}
3.0 - 4.0	5.0×10^{-3}
Liquid Phase (84,000 gal)	
Energy Group (Mev/gamma)	Specific Source Strength (Mev/g-s)
0.2 - 0.4	4.0×10^{-5} (a)
0.4 - 0.9	1.2×10^{-5}
0.9 - 1.35	4.4×10^{-4}
1.35 - 1.8	6.4×10^{-4}
1.8 - 2.2	9.4×10^{-4}
2.2 - 2.6	1.6×10^{-5}
2.6 - 3.0	3.8×10^{-3}
3.0 - 4.0	2.9×10^{-3}

a. Include 80 kev Xe-133.

Table 12.2-13

BORON RECYCLE SYSTEM DEMINERALIZER SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -s)	
	Evaporator Feed ^(a) Demineralizers (30 ft ³ of Resin)	Recycle Evaporator Condensate Demineralizer (30 ft ³ of Resin)
0.2 - 0.4	6.9×10^{-6}	2.8×10^{-4}
0.4 - 0.9	2.4×10^{-8}	1.9×10^{-4}
0.9 - 1.35	3.3×10^{-7}	3.9×10^{-3}
1.35 - 1.8	5.6×10^{-6}	1.9×10^{-3}
1.8 - 2.2	2.9×10^{-4}	1.7×10^{-2}
2.2 - 2.6	1.9×10^{-4}	1.1×10^{-2}

- a. Assumes a particulate decontamination factor of 10 across these beds.

Table 12.2-14

BORON RECYCLE SYSTEM RECYCLE EVAPORATOR SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength ^{(a) (b)}	
	Vent Condenser Vapor	Evaporator Concentrate
	(Mev/cm ³ -s)	(Mev/g-s)
0.2 - 0.4	1.3×10^{-7}	1.6×10^{-5}
0.4 - 0.9	1.8×10^{-6}	8.5×10^{-5}
0.9 - 1.35	5.6×10^{-5}	4.3×10^{-5}
1.35 - 1.8	1.7×10^{-6}	4.6×10^{-4}
1.8 - 2.2	2.8×10^{-6}	3.6×10^{-3}
2.2 - 2.6	5.4×10^{-6}	2.1×10^{-3}
2.6 - 3.0	5.6×10^{-4}	—
3.0 - 4.0	7.5×10^{-4}	—

- a. Considered to be homogeneous, cylindrical sources with the following dimensions and compositions :

<u>Section</u>	<u>Source Radius (in.)</u>	<u>Dimensions Length (in.)</u>	<u>Source Composition (vol%)</u>
Vent Condenser	4	20	30% stainless steel, 22% water, 48% vapor,
Evaporator Concentrates	21	119	10% stainless steel, 77% water, 13% air

- b. Evaporator concentrates source strengths are based on an evaporator process limit of 40 microcuries of long-lived radionuclides per gram of concentrates.

Table 12.2-15

BORON RECYCLE SYSTEM FILTER^(a) SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength (Mev/cm ³ -s)		
	Recycle Evaporator Feed Filter	Recycle Evaporator Condensate Filter	Recycle Evaporator Concentrates Filter
0.2 - 0.4	—	2.8×10^{-4}	1.6×10^{-5}
0.4 - 0.9	1.1×10^{-7}	1.9×10^{-4}	8.5×10^{-5}
0.9 - 1.35	3.0×10^{-6}	3.9×10^{-3}	4.3×10^{-5}
1.35 - 1.8	—	1.9×10^{-3}	4.6×10^{-4}
1.8 - 2.2	—	1.7×10^{-2}	3.6×10^{-3}
2.2 - 2.6	—	1.1×10^{-2}	2.1×10^{-3}

- a. Considered to be homogeneous sources with the following dimensions and compositions :

<u>Filter</u>	<u>Source Radius (in.)</u>	<u>Dimensions Length (in.)</u>	<u>Source Composition (vol%)</u>
Recycle Evaporator Feed	3.375	19	62% air, 38% water
Recycle Evaporator Condensate and Concentrated	1.25	19	30% air, 70% water

Table 12.2-16

RESIDUAL HEAT REMOVAL SYSTEM SOURCE STRENGTH

Energy Group (Mev/gamma)	Specific Source Strength (Mev/g-s)	
	4 Hours After Shutdown	8 Hours After Shutdown
0.2 - 0.4	3.8×10^5 (a)	3.3×10^5 (a)
0.4 - 0.9	2.3×10^5	1.2×10^5
0.9 - 1.35	1.1×10^5	5.7×10^4
1.35 - 1.8	3.3×10^4	1.2×10^4
1.8 - 2.2	3.6×10^4	9.4×10^3
2.2 - 2.6	3.8×10^4	9.6×10^3
2.6 - 3.0	2.9×10^3	7.1×10^2
3.0 - 4.0	1.1×10^3	2.8×10^2

a. Include 80 keV Xe-133.

Table 12.2-17

CORE AVERAGE GAMMA RAY SOURCES STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/watt-s)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0.20 - 0.40	1.8×10^9	1.5×10^9	8.2×10^8	5.8×10^8	1.4×10^8
0.40 - 0.90	1.1×10^{10}	9.4×10^9	6.2×10^9	5.3×10^9	3.3×10^9
0.90 - 1.35	1.9×10^9	1.2×10^9	5.8×10^8	4.2×10^8	1.2×10^8
1.35 - 1.80	3.8×10^9	3.3×10^9	2.7×10^9	2.3×10^9	6.6×10^8
1.80 - 2.20	2.6×10^8	1.8×10^8	1.2×10^8	9.2×10^7	3.7×10^7
2.20 - 2.60	2.5×10^8	1.9×10^8	1.6×10^8	1.4×10^8	4.0×10^7
2.60 - 3.00	6.5×10^6	3.3×10^6	2.7×10^6	2.4×10^6	6.9×10^5
3.00 - 4.00	4.5×10^6	1.3×10^6	1.1×10^6	9.4×10^5	2.7×10^5
	3 Months	6 Months	1 Year	5 Years	
0.20 - 0.40	4.6×10^7	2.6×10^7	1.5×10^7	1.6×10^6	
0.40 - 0.90	1.8×10^9	9.0×10^9	3.2×10^8	8.0×10^8	
0.90 - 1.35	3.0×10^7	1.8×10^7	1.3×10^7	4.8×10^6	
1.35 - 1.80	3.8×10^7	1.1×10^7	8.1×10^7	1.3×10^6	
1.80 - 2.20	1.6×10^7	1.2×10^7	7.5×10^6	2.2×10^5	
2.20 - 2.60	1.5×10^6	1.2×10^4	—	—	
2.60 - 3.00	2.7×10^4	2.0×10^2	—	—	
3.00 - 4.00	1.0×10^4	—	—	—	

Table 12.2-18

SPENT FUEL GAMMA RAY SOURCES STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (MeV/watt-s)				
	12 Hours	24 Hours	100 Hours	1 Week	1 Month
0.20 - 0.40	1.8×10^9	1.6×10^9	8.4×10^8	6.0×10^8	1.6×10^8
0.40 - 0.90	1.1×10^{10}	9.7×10^9	6.5×10^9	5.6×10^9	3.3×10^9
0.90 - 1.35	2.0×10^9	1.4×10^9	7.0×10^8	5.3×10^8	1.6×10^8
1.35 - 1.80	3.6×10^9	3.2×10^9	2.6×10^9	2.2×10^9	6.4×10^8
1.80 - 2.20	3.2×10^8	2.5×10^8	1.7×10^8	1.4×10^7	5.7×10^7
2.20 - 2.60	2.4×10^8	1.8×10^8	1.5×10^8	1.3×10^8	3.8×10^7
2.60 - 3.00	5.9×10^6	3.2×10^6	2.6×10^6	2.3×10^6	6.6×10^5
3.00 - 4.00	4.2×10^6	1.2×10^6	1.0×10^6	9.0×10^5	2.6×10^5
	3 Months	6 Months	1 Year	5 Years	
0.20 - 0.40	5.3×10^7	3.1×10^7	1.8×10^7	2.2×10^6	
0.40 - 0.90	2.0×10^9	1.1×10^8	4.8×10^8	1.3×10^7	
0.90 - 1.35	4.6×10^7	3.0×10^7	2.3×10^7	8.7×10^6	
1.35 - 1.80	4.6×10^7	1.8×10^7	1.3×10^6	2.4×10^6	
1.80 - 2.20	2.0×10^7	1.4×10^7	8.8×10^6	2.5×10^5	
2.20 - 2.60	1.5×10^6	1.1×10^4	—	—	
2.60 - 3.00	2.6×10^5	2.0×10^2	—	—	
3.00 - 4.00	1.0×10^4	—	—	—	

Table 12.2-19

CORE AVERAGE AND SPENT FUEL NEUTRON SOURCE
STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN ^(a)

Time After Shutdown	Core Average (n/watt-s)	Spent Fuel (n/watt-s)
12 hours	8.1	20
24 hours	8.1	20
100 hours	8.1	20
1 week	8.0	20
1 month	7.7	19
3 months	7.0	17
6 months	6.2	16
1 year	5.3	13
5 years	3.9	10

- a. 83 to 93% of the neutron source strength is due to the spontaneous fission of curium-242 and curium-244. The curium spontaneous fission neutron spectrum is quite similar to that of californium-252. The californium-252 spontaneous fission neutron spectrum may be expressed by a watt formula as follows :

$$\chi(E) = 0.37 \text{ EXP } (-0.88E) \text{ SINH } \left(\sqrt{\frac{E}{1.04}} \right)$$

where E is the neutron energy and $\chi(E)$ is normalized so that

$$\int_0^{\infty} \chi(E) dE = 1$$

Table 12.2-20

IRRADIATED Sb-Be SECONDARY SOURCE ROD
SOURCE STRENGTHS ^(a) (NEUTRON) (Sheet 1 of 2)

Time After Shutdown	Neutron Source Strength (n/cm ³ -s)
1 day	4.5×10^8
1 week	4.2×10^8
1 month	3.2×10^8
6 months	5.8×10^7
1 year	6.8×10^6
5 years	—

- a. The secondary source rod cross-sectional area is 0.582 cm² per rod, the average neutron energy is 30 kev, and the Sb-Be material density is 3.38 g/cm³. The rod has been irradiated for 400 days.

Table 12.2-20

IRRADIATED Sb-Be SECONDARY SOURCE ROD
SOURCE STRENGTHS ^(a) (GAMMA RAY) (Sheet 2 of 2)

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm ³ -s)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	3.0×10^{10}	2.9×10^{10}	2.5×10^{10}	1.1×10^{10}	3.7×10^9	2.2×10^7
0.40 - 0.90	1.1×10^{13}	7.0×10^{12}	4.6×10^{12}	8.1×10^{11}	9.7×10^{10}	1.8×10^8
0.90 - 1.35	6.7×10^{11}	4.8×10^{11}	3.4×10^{11}	6.0×10^{10}	7.0×10^9	-
1.35 - 1.80	7.6×10^{12}	7.1×10^{12}	5.5×10^{12}	9.7×10^{11}	1.2×10^{11}	-
1.80 - 2.20	9.8×10^{11}	9.1×10^{11}	7.0×10^{11}	1.2×10^{11}	1.5×10^{10}	-

- a. The secondary source cross-sectional area is 0.582 cm² per rod and the Sb-Be material density is 3.38 g/cm³. The rod has been irradiated for 400 days.

Table 12.2-21

IRRADIATED Ag-In-Cd CONTROL ROD SOURCE STRENGTHS ^(a)

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (MeV/cm ³ -s)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	2.3×10^{-8}	2.3×10^{-8}	2.2×10^{-8}	1.4×10^{-8}	8.5×10^{-7}	1.5×10^{-6}
0.40 - 0.90	1.1×10^{-12}	1.1×10^{-12}	1.0×10^{-12}	6.6×10^{-11}	4.0×10^{-11}	7.1×10^{-9}
0.90 - 1.35	2.0×10^{-11}	1.9×10^{-11}	1.8×10^{-11}	1.2×10^{-11}	7.2×10^{-10}	1.3×10^{-9}
1.35 - 1.80	3.7×10^{-11}	3.7×10^{-11}	3.4×10^{-11}	2.3×10^{-11}	1.4×10^{-11}	2.5×10^{-9}

- a. The absorber cross-sectional area is 0.589 cm² per rod the absorber material density is 10.17 g/cm³.

Table 12.2-22

IRRADIATED TYPE-304 STAINLESS STEEL (0.12 WEIGHT % Co)
CLADDING SOURCE STRENGTHS ^(a)

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (MeV/cm ³ -s)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	7.1×10^9	6.1×10^9	3.4×10^9	8.3×10^7	9.9×10^5	-
0.40 - 0.90	3.1×10^{10}	2.9×10^{10}	2.6×10^{10}	1.2×10^{10}	6.4×10^9	2.3×10^8
0.90 - 1.35	2.4×10^{11}	2.3×10^{11}	2.3×10^{11}	2.1×10^{11}	2.0×10^{11}	1.2×10^{11}
1.35 - 1.80	1.9×10^8	1.8×10^8	1.4×10^8	3.3×10^7	5.4×10^6	-

a. The various cladding cross-sectional areas per rod are :

Ag-In-Cd control rod - 0.136 cm^2

Sb-Be secondary source rod - 0.136 cm^2

Burnable poison rod
(including inner sheath) - 0.159 cm^2

Cf-252 primary source rod - 0.136 cm^2

Table 12.2-23

IRRADIATED INCORE DETECTOR AND DRIVE CABLE MAXIMUM
WITHDRAWAL SOURCE STRENGTHS ^(a)

Energy Group (Mev/gamma)	Incore Detector (Mev/cm ³ -s)	Drive Cable (Mev/cm ³ -s)
0.20 - 0.40	3.8×10^{10}	6.0×10^8
0.40 - 0.90	1.6×10^{11}	5.1×10^{10}
0.90 - 1.35	1.1×10^{11}	1.6×10^{10}
1.35 - 1.80	1.1×10^{11}	3.1×10^8
1.80 - 2.20	2.9×10^{10}	3.8×10^{10}
2.20 - 2.60	3.1×10^{10}	1.3×10^9
2.60 - 3.00	1.6×10^{10}	1.3×10^9
3.00 - 4.00	2.1×10^{10}	3.1×10^8
4.00 - 5.00	1.5×10^{10}	—
5.00 - 6.00	1.4×10^9	—

- a. The effective diameter and length of the incore detector are 0.48 and 5.33 cm, respectively, and the effective cross-sectional area of the drive cable is 0.095 cm².

Table 12.2-24

IRRADIATED INCORE DETECTOR DRIVE CABLE AVERAGE SOURCE STRENGTHS ^(a)

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm ³ -s)						
	8 Hours	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	5.8×10^8	5.7×10^8	4.9×10^8	2.8×10^8	9.8×10^6	2.8×10^5	—
0.40 - 0.90	1.7×10^{10}	1.2×10^{10}	1.2×10^{10}	1.1×10^{10}	7.8×10^9	5.0×10^9	2.0×10^8
0.90 - 1.35	1.6×10^{10}	1.6×10^{10}	1.5×10^{10}	1.2×10^{10}	5.6×10^9	4.2×10^9	2.5×10^9
1.35 - 1.80	2.1×10^7	1.1×10^7	1.1×10^7	8.5×10^6	2.0×10^6	3.3×10^5	—
1.80 - 2.20	4.5×10^9	6.1×10^7	—	—	—	—	—
2.20 - 2.60	1.5×10^8	2.0×10^6	—	—	—	—	—
2.60 - 3.00	1.6×10^8	2.1×10^6	—	—	—	—	—
3.00 - 4.00	3.6×10^7	5.0×10^5	—	—	—	—	—

a. The drive cable effective cross-sectional area is 0.095 cm²

Table 12.2-25

IRRADIATED TYPE-316 STAINLESS STEEL (0.12 WEIGHT % Co)
FLUX THIMBLE SOURCE STRENGTHS^(a)

Energy Group (MeV/gamma)	Source Strength at Time After Shutdown (MeV/cm ³ -s)						
	12 Hours	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 – 0.40	6.5×10^9	6.4×10^9	5.5×10^9	3.1×10^9	7.5×10^7	9.0×10^5	–
0.40 – 0.90	5.7×10^{10}	3.7×10^{10}	3.3×10^{10}	2.8×10^{10}	1.2×10^{10}	6.2×10^9	2.2×10^8
0.90 – 1.35	2.4×10^{11}	2.4×10^{11}	2.4×10^{11}	2.3×10^{11}	2.2×10^{11}	2.0×10^{11}	1.2×10^{11}
1.35 – 1.80	2.9×10^8	2.2×10^8	2.1×10^8	1.7×10^8	3.9×10^7	6.4×10^6	–
1.80 – 2.20	2.1×10^{10}	8.2×10^8	–	–	–	–	–
2.20 – 2.60	6.8×10^8	2.7×10^7	–	–	–	–	–
2.60 – 3.00	7.2×10^8	2.9×10^7	–	–	–	–	–
3.00 – 4.00	1.7×10^8	6.7×10^6	–	–	–	–	–

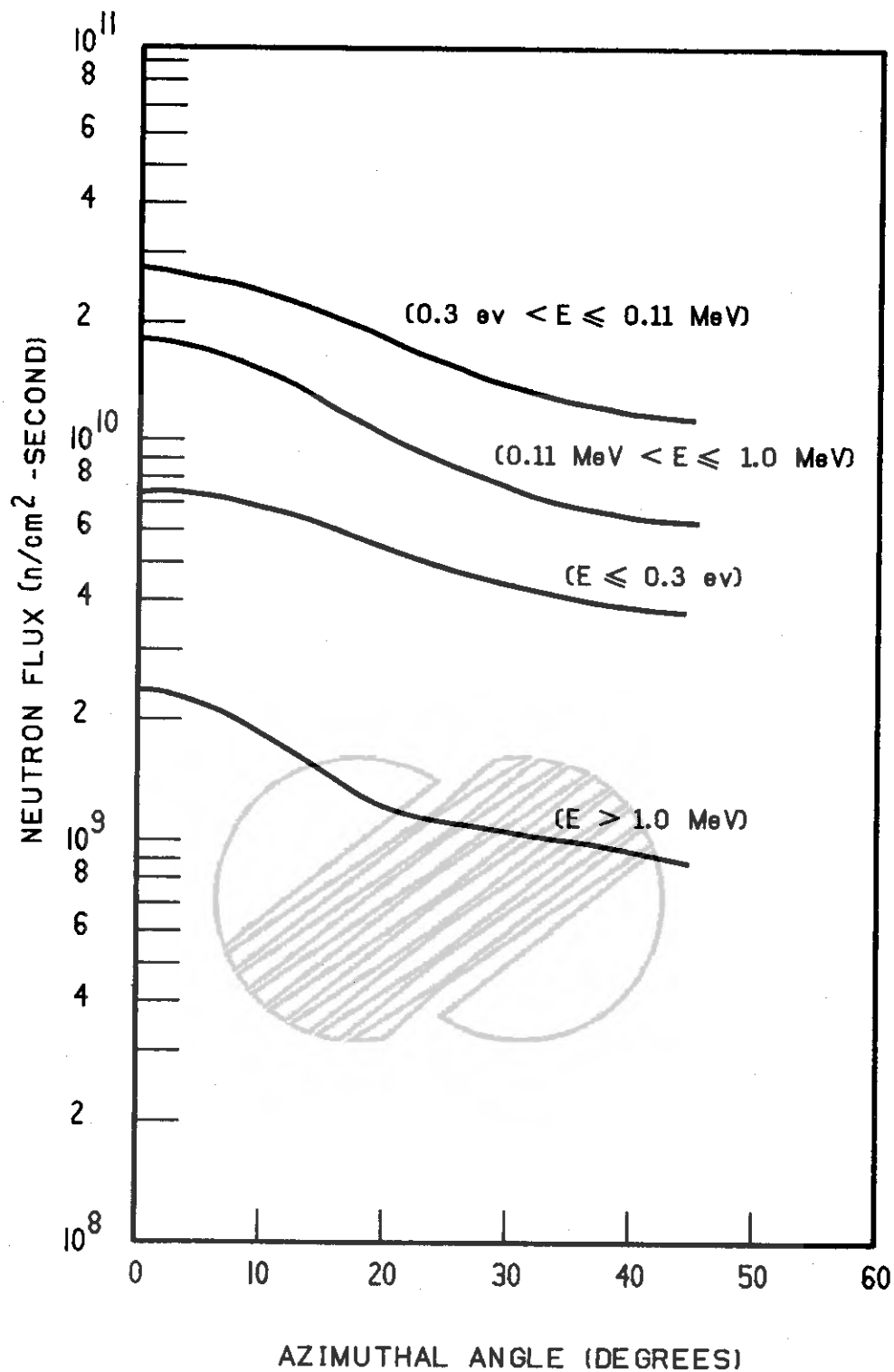
a. The Flux thimble cross-sectional area is 0.270 cm²

Table 12.2-26

SOURCE STRENGTH IN THE RESIDUAL HEAT REMOVAL LOOP AT
VARIOUS TIMES FOLLOWING A MAXIMUM CREDIBLE ACCIDENT
(TID-14844 Release Fractions)

Energy Group (MeV/gamma)	Source Strength at Time After Release ^{a)} (MeV/gm-Sec-Watt)				
	0 Hours	0.5 Hours	1 Hour	2 Hours	8 Hours
0.20 - 0.40	3.0×10^{-1}	2.0×10^{-1}	1.8×10^{-1}	1.6×10^{-1}	1.4×10^{-1}
0.40 - 0.90	3.4	2.6	2.1	1.5	5.9×10^{-1}
0.90 - 1.35	1.7	1.1	9.2×10^{-1}	7.3×10^{-1}	3.3×10^{-1}
1.35 - 1.80	1.3	7.1×10^{-1}	6.1×10^{-1}	4.8×10^{-1}	2.1×10^{-1}
1.80 - 2.20	1.2×10^{-1}	7.4×10^{-2}	5.8×10^{-2}	4.2×10^{-2}	1.5×10^{-2}
2.20 - 2.60	2.5×10^{-1}	5.0×10^{-2}	4.1×10^{-2}	3.1×10^{-2}	1.4×10^{-2}
2.60 - 3.00	3.2×10^{-1}	5.3×10^{-3}	2.5×10^{-3}	6.8×10^{-4}	-
3.00 - 4.00	1.7×10^{-1}	3.6×10^{-2}	1.7×10^{-2}	4.9×10^{-3}	-
4.00 - 5.00	3.0×10^{-1}	4.9×10^{-4}	2.5×10^{-4}	-	-
5.00 - 6.00	1.5×10^{-3}	-	-	-	-
	1 Day	1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	1.2×10^{-1}	7.1×10^{-2}	1.0×10^{-2}	2.0×10^{-4}	1.1×10^{-4}
0.40 - 0.90	3.1×10^{-1}	5.8×10^{-2}	2.7×10^{-2}	6.7×10^{-3}	2.3×10^{-3}
0.90 - 1.35	7.8×10^{-2}	3.3×10^{-3}	8.6×10^{-4}	1.3×10^{-4}	9.6×10^{-5}
1.35 - 1.80	5.8×10^{-2}	1.8×10^{-2}	5.0×10^{-3}	8.3×10^{-5}	5.9×10^{-5}
1.80 - 2.20	3.4×10^{-3}	6.7×10^{-4}	2.7×10^{-4}	8.9×10^{-5}	5.7×10^{-5}
2.20 - 2.60	3.6×10^{-3}	1.0×10^{-3}	3.0×10^{-4}	-	-

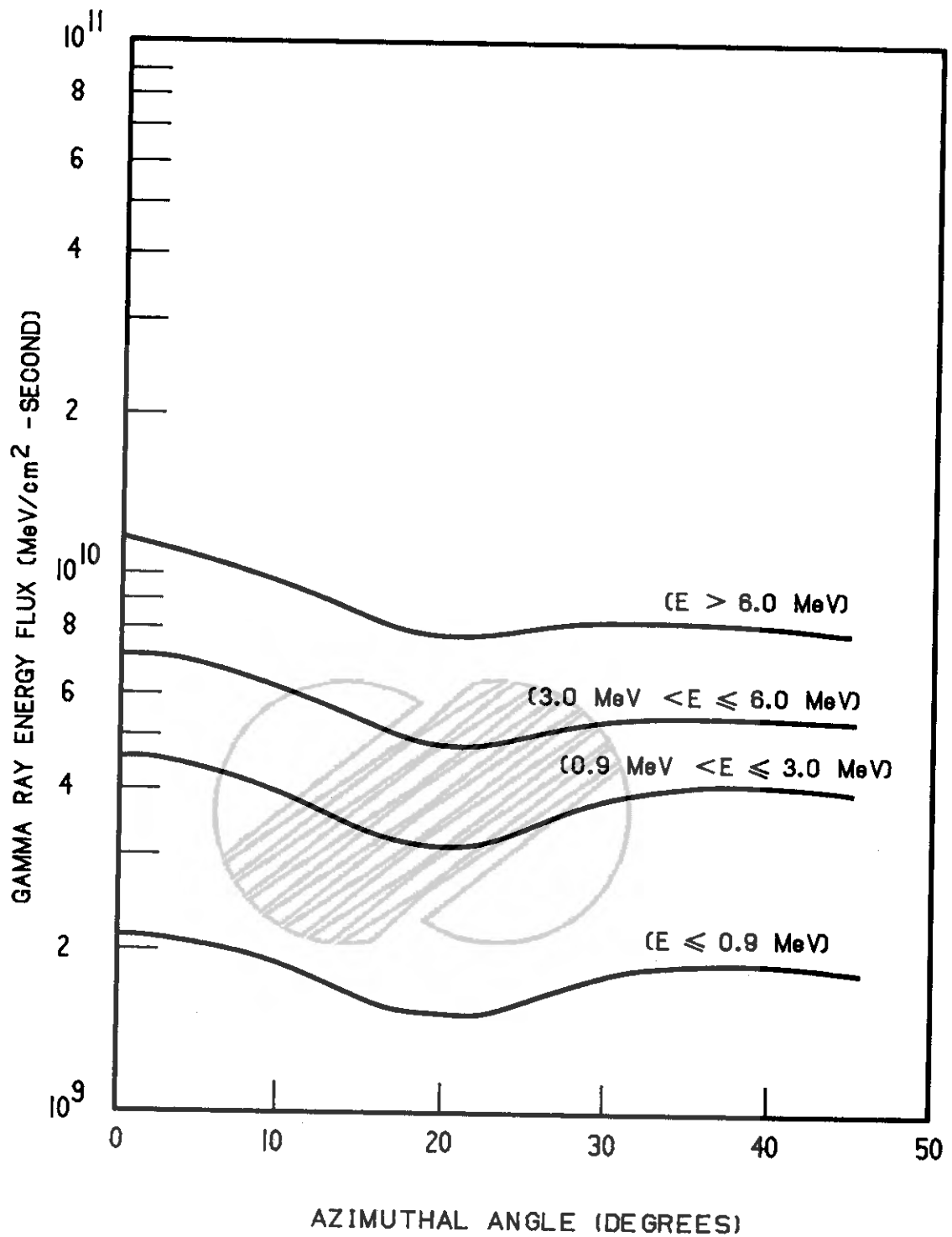
a) the source strengths in this table should be increased by 14% based on a power level of 2,958 MWt. | 479



KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 7 & 8
FSAR

AZIMUTHAL DISTRIBUTION OF NEUTRON
FLUX INCIDENT ON THE PRIMARY
SHIELD AT THE REACTOR CORE MIDPLANE

Figure 12.2-1

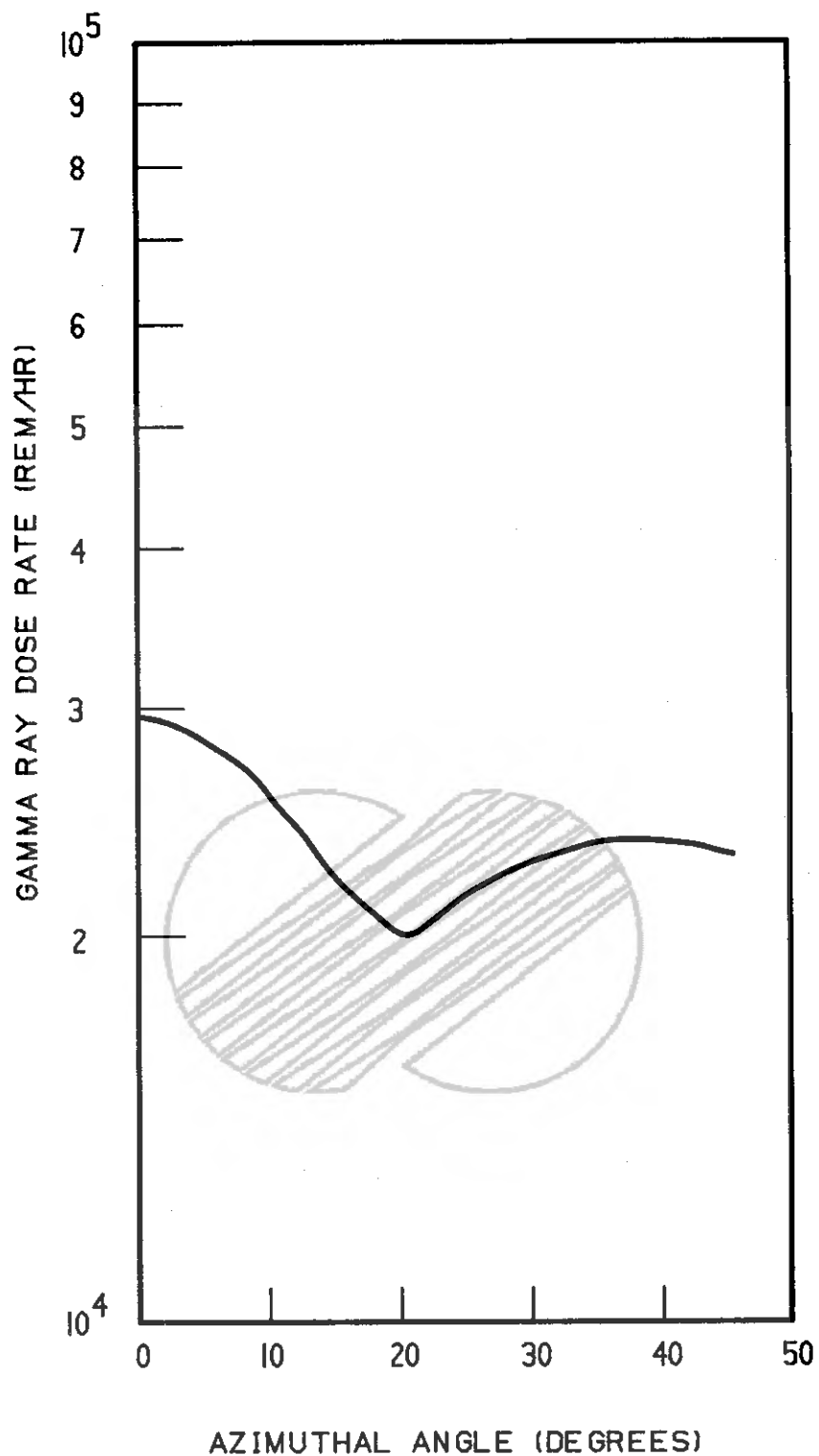


KOREA ELECTRIC POWER CORPORATION
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FSAR

AZIMUTHAL DISTRIBUTION OF GAMMA
RAY ENERGY FLUX INCIDENT ON THE
PRIMARY SHIELD AT THE REACTOR
CORE MIDPLANE

Figure 12.2-2

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KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 7 & 8
FSAR

AZIMUTHAL DISTRIBUTION OF GAMMA RAY
DOSE RATE INCIDENT ON THE PRIMARY
SHIELD AT THE REACTOR CORE MIDPLANE

Figure 12.2-3

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

2

In this subsection specific design features for maintaining personnel exposures as low as reasonably achievable (ALARA) are discussed. The radiation protection guidance given in Paragraph C.3 of Regulatory Guide 8.8 is followed to minimize exposures to personnel.

Facilities and equipment of a specialized nature for handling special nuclear source and byproduct material are not required except for fuel handling and radioactive waste processing. Fuel handling and radwaste processing equipment are described in section 9.1 and chapter 11, respectively. Materials handled in the radiochemistry laboratory and sealed sources used for calibration purposes are of a low activity level and do not require special handling equipment. Unsealed sources and radioactive samples are handled in conventional hoods which exhaust to the auxiliary building ventilation system, described in section 9.4.

12.3.1.1 Plant Design Description for as Low as Reasonably Achievable Exposures

The equipment and plant design features employed to maintain radiation exposures ALARA are based upon the design considerations of subsection 12.1.2 and are outlined in this subsection for several general classes of equipment (subparagraph 12.3.1.1.1) and several typical plant layout situations (subparagraph 12.3.1.1.2).

2

12.3.1.1.1 BOP Equipment and Component Designs for as Low as Reasonably Achievable Exposures

This paragraph 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:

12.3.1.1.1.1 Filters. All filters in the plant are of the cartridge type. Filters which accumulate significant quantities of radioactivity are designed for cartridge replacement with semi-remote (i.e., long handled) tools. Adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste area.

RADIATION PROTECTION
DESIGN FEATURES

Liquid systems containing radioactive filters, except the low activity blowdown filters, and the liquid radwaste system pre-filter, are provided with a semi-remote filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the drumming station for packaging and shipment from the site for disposal. In use, the handling system is placed over the filter in the space normally occupied by its concrete hatch. The lead base of the system adequately attenuates cartridge radiation. A remote, closed circuit TV and a leaded glass window provide the operator with a complete view of the filter housing and cartridge while he is performing the changeout with remote tools. The cartridge can then be lifted into a shield cask placed on the base. The operator is never exposed to unattenuated radiation from the cartridge. An overhead monorail is used to transport cask and cartridge to the radwaste storage area. The process is accomplished in such a manner that exposure to personnel and the possibility of inadvertent radioactive release to the environment is minimized. Each radioactive filter is contained in a shielded compartment provided with external vent and drain valving and compartment drainage capabilities.

12.3.1.1.1.2 Demineralizers. Demineralizers for highly 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 loaded into the demineralizer remotely. Retention screens are designed for full system pressure drop. The demineralizers and piping are designed with provisions for being flushed with demineralized water. Strainers are installed in the vent lines to prevent entry of spent resin.

12.3.1.1.1.3 Evaporators. Evaporators are provided with chemical addition connections to allow the use of chemicals for descaling operations. Provisions are provided to allow removal of heating tube bundles. The highly radioactive evaporator components are separated from those that are less radioactive by a shield wall. All instruments and controls are located on the accessible side of the shield wall. All valves in radioactive lines are located on or can be operated from the accessible side of the shield wall. Alternately, the lines containing the valve can be flushed prior to access for valve lineup.

12.3.1.1.1.4 Pumps. Wherever practicable, pumps are purchased with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner which allows easy removal, if necessary.

All pumps in 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.

12.3.1.1.1.5 Tanks. Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system in order to control any contamination within plant structures. For tanks outside structures containing significant quantities of radioactivity, the overflows are directed to the liquid radwaste system. All tanks have high level alarms to alert the operator prior to an overflow occurrence.

12.3.1.1.1.6 Heat Exchangers. Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials to minimize leakage. Impact baffles are provided and tube side and shell side velocities are limited to minimize erosive effects. Wherever possible, the radioactive fluid passes through the tube side of the heat exchanger.

12.3.1.1.1.7 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. Transmitters and readout devices are located in low radiation zones, such as corridors and the control room, for servicing. Some instruments (such as thermocouples) in high radiation zones are provided in duplicate to reduce access and service time required.

Diaphragm seals are provided on instrument sensing lines on process piping which may contain highly radioactive solids 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.

12.3.1.1.1.8 Valves. To minimize personnel exposures from valve operations, motor-operated, diaphragm, or other remotely actuated valves are used where justified by the activity levels and frequency of use.

Valves are located in valve galleries so that they are shielded separately from the major components. Long runs of exposed piping are minimized in valve galleries. In areas where manual

valves are used on frequently operated process lines, either valve reach rods or local shielding are provided such that personnel need not enter a high radiation area for valve operation.

When equipment in a Zone 6 area 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. To the maximum practicable extent, simple straight reach rods will be used to retain the feel of whether the valves are tightly closed or not. Where possible, valves with reach rods are installed with their stems horizontal so that the reach rods are also horizontal but above the heads of personnel to permit ready access.

All manually-operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are equipped with reach rods extending through the valve gallery wall. Personnel do not enter the valve gallery during spent resin or cartridge transfer operations. The valve gallery shield walls are designed to minimize personnel exposure during maintenance components within or adjacent to the gallery and to protect personnel who remotely operate the valves.

Selection and layout of valves is done with consideration to minimize exposure to maintenance personnel. Wherever practical, space is provided for erection of temporary shielding during maintenance. For valves located in radiation areas, provisions are made to drain adjacent radioactive components where maintenance is required. Wherever practicable, valves for clean, nonradioactive systems are separated from radioactive sources and are located in readily accessible areas. Valve type and hardware selection involves evaluation of the associated maintenance times, durability of plug linings and stem packings, and requirements for limiting stem leakage. Wherever possible, only those packing materials are used which have demonstrated satisfactory performance in nuclear plant operation. Valve designs with minimum internal crevices are used where trapping of radioactive solids could become a problem. These areas include piping carrying spent resin or evaporator bottoms.

12.3.1.1.1.9 Piping. The piping in pipe chases is designed for the lifetime of the unit. Whenever practical, valves and instrumentation are not located in the pipe chase. Wherever radioactive piping is routed through areas where routine maintenance is required, chases are provided to reduce the radiation contribution from these pipes to levels permitting maintenance. Wherever practical, piping containing radioactive material is routed to minimize radiation exposure to plant personnel.

12.3.1.1.1.10 Floor Drains. Floor drains and properly sloped floors are provided for each room or cubicle containing serviceable components containing radioactive liquids. When practicable, shielded pipe chases are used for radioactive pipes. If a radioactive drain line must pass through a plant area requiring personnel access, shielding is provided as necessary to ensure radiation levels consistent with the required access.

12.3.1.1.1.11 Lighting. Wherever practicable, 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, the personnel exposure is reduced. The fluorescent lights which are used in some areas do not require frequent service due to the increased life of the tubes. However, when the system is secured and flushed out, the burned out lamps can be replaced rapidly so as to minimize the exposure of personnel.

12.3.1.1.1.12 Heating, Ventilating, and Air Conditioning System. The heating, ventilating, and air conditioning (HVAC) system design provides, among other things, for rapid replacement of the filter elements and housings. The design features are described in detail in subparagraph 12.3.3.3.

1.23.1.1.1.13 Hydrogen Recombiners. The location of the hydrogen recombiner control panel has been selected to assure adequate shielding from highly radioactive components.

12.3.1.1.1.14 Sample Stations. Sample stations for routine sampling of process fluids are located in accessible areas. Proper shielding and ventilation are provided at the local sample stations to maintain radiation zoning in proximate areas and minimize personnel exposure during sampling. The counting room and laboratory facilities are described in section 12.5.

12.3.1.1.1.15 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 Westinghouse Component Design Considerations.

In terms of radiation protection features of Westinghouse designed systems, Westinghouse provides the systems and components but does not provide layout or shielding for those systems and components. Thus, the following discussion is limited to only the systems designs supplied by Westinghouse. Radiation protection design features of Westinghouse systems and components are given below:

12.3.1.1.2.1 Reactor vessel. The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors. The nozzle area is tapered along the reinforced areas to assure a smooth transition, and pipe branch locations are selected to assure no interference from one branch to the next. All weld-to-pipe interfaces require a smooth, high quality finish.

12.3.1.1.2.2 Reactor coolant pumps. The reactor coolant pump design includes the use of an assembled cartridge seal for the number 3 pump seal which reduces the time required for replacement. The cartridge seal is also expected to have a useful life double that of the old design. The reactor coolant pump design also includes a spool piece to facilitate separation and replacement of the motor from the pump.

12.3.1.1.2.3 Reactor vessel insulation. Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate removal for inspection of the welds.

12.3.1.1.2.4 Steam generators. The Model F steam generators incorporate several design features to facilitate maintenance and inspection in reduced radiation fields. The tube ends are designed to be flush with the tube sheet in the steam generator channel head to eliminate a potential crud trap. The steam generator manways (entrance to channel head) are sized to facilitate entrance and exit with protective clothing. Hand-holes to the secondary side are positioned to facilitate maintenance operations. The use of all volatile treatment (AVT) chemistry on the secondary side serves to increase steam generator reliability and also will reduce occupational radiation exposures. Changes in steam generator design such as improved tube support plates (stainless steel and quatrefoil flow holes) also contribute to reduced exposures through increased reliability.

12.3.1.1.2.5 Reactor coolant pipe connections. To minimize crud buildup in branch lines, piping connections to the reactor coolant loops are located on or above the horizontal centerline of the pipe wherever possible.

12.3.1.1.2.6 Evaporator package (boron recycle system). The evaporator packages are skid mounted with a 24-inch space for shielding the concentrator tank, gas stripper, and vent condenser from the pumps, valves, and instrument panel, thus reducing the radiation field during operation and maintenance.

12.3.1.1.2.7 Heat exchangers. The heat exchangers are designed such that the shell-to-tube sheet joint need not be broken for inspection. The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection and/or testing for leaks.

12.3.1.1.2.8 Valves. Valves are of the bolted body-to-bonnet forging type. This permits the use of ultrasonic testing in place of radiography for inspection and facilitates assembly and disassembly. This reduces inspection and maintenance time. Additionally, manual valves under 2 inches in diameter are designed for zero stem leakage.

12.3.1.1.2.9 Pumps. Pumps (other than the reactor coolant pumps) are designed with flanged connections to facilitate removal for maintenance. Depending on expected conditions, either canned pumps or pumps with high quality mechanical seals are used to reduce leakage and maintenance requirements.

12.3.1.1.2.10 Demineralizers. Demineralizer resin screens are constructed for substantially higher than normal pressure differential to accommodate higher than design flows without breakdown.

12.3.1.1.2.11 Filters. Filters are designed to be removable from the top with lifting bails in the middle of the head. The filter assemblies have bolt lead-ins for tool entry and the filter element in the form of a disposable cartridge. These features facilitate remote removal, disposal, and assembly.

12.3.1.1.2.12 Materials. Equipment specifications for components exposed to high temperature reactor coolant contain specific limitations on the cobalt impurity content of the base metal as given in table 12.3-3, thereby controlling the potential for production of radio-active cobalt-60 from the base metal impurity cobalt-59. The estimated surface area of material in contact with the reactor coolant is given in table 12.3-4. The use of hard facing material with cobalt content such as stellite is limited to applications where its use is necessary for reliability considerations (table 12.3-5 shows the estimated total surface area of stellite). Nickel based alloys (cobalt-58 is produced from activation of the base metal nickel-58) are

similarly used only where component reliability may be compromised by the use of other materials. The major use of nickel based alloys is in the Inconel steam generator tubes. The surface area in contact with the reactor coolant system is given in table 12.3-4. From tables 12.3-4 and 12.3-5, it can be seen that the Inconel surface is the predominant area in contact with the reactor coolant system and that the stellite area is minimal.

The design features delineated above to maintain occupational radiation exposures to ALARA levels are in compliance with the guidance and considerations given in Regulatory Guide 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable." The Westinghouse designed systems and components take into consideration the potential use of remote and semi-remote equipment. These considerations are primarily focused on inservice inspection of the reactor vessel and inspection/maintenance of the steam generators. These components are designed to be compatible with remote and semi-remote equipment developed by Westinghouse Nuclear Service Division or other similar tooling.

12.3.1.1.3 Common Facility and Layout Designs for as Low as Reasonably Achievable Exposures

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

12.3.1.1.3.1 Valve Galleries. Valve galleries are provided with shielded entrances for personnel protection. Where practical, the valve galleries are divided so that personnel requiring access are exposed only to valves and piping associated with one component at any given location. Floor drains are provided to control radioactive leakage. To facilitate decontamination in valve galleries, concrete surfaces are covered with a smooth surfaced coating which will allow easy decontamination.

12.3.1.1.3.2 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 dose rate. Where it is necessary that radioactive piping be routed through corridors or other low radiation zone areas, shielded pipe chases 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 always located in appropriately zoned and restricted areas. Process piping is monitored to ensure that access is controlled to limit exposure (subsection 12.5.3).

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. Butt welds are used in lieu of socket welds wherever the use of socket welds might lead to significant crud trap formation. For pipes carrying resin slurries, large radius bends are utilized instead of elbows and piping direction changes are minimized. Wherever practicable, pipe runs are sloped to promote drainage and to minimize crud and particulate depositions. Provisions are made for flushing of piping and equipment.

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

12.3.1.1.3.3 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.

12.3.1.1.3.4 Contamination Control. Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at joints. Where practicable, drains are run from pump baseplates to the gravity drainage system so that seal leakage contamination is minimized. In those systems handling radioactive fluids and using components which intrinsically leak, the spread of contamination is prevented by use of monitoring, detection, or collection methods. Leakage from components in high radiation zones can be monitored by local or building sump

level indications and alarms or by area and process radiation monitors. System process instrumentation can provide an indication of system leakage. Additional information on design features which permit monitoring and control of leakage are presented in subsections 5.2.5, 9.3.3, and 12.3.4.

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

Floor drains with properly sloping floors are provided in all potentially contaminated areas of the plant.

In controlled access areas where contamination is expected, radiation monitoring equipment is provided (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.

12.3.1.1.3.5 Equipment Layout. In those systems where process equipment is a major radiation source (such as chemical volume control, radwaste, boron recycle, 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 (Zones 1, 2, or 3).

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 entranceway 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. For potentially high radiation components (such as filters and demineralizers) completely enclosed shielded compartments with hatch openings are used.

Figure 12.3-1 provides typical layout arrangements for valve compartments or galleries.

Exposure from routine in-plant inspection is controlled by locating, whenever possible, inspection points in properly shielded low background radiation areas. Radioactive and non-radioactive 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. 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 direct supervision of the health physics personnel.

12.3.1.1.3.6 Field Run Piping. Routing and Shielding of all radioactive process piping of 1 inch and larger diameter is performed in the design office. Field routing of smaller radioactive lines is done by providing field engineering with criteria for routing these lines.

12.3.1.2 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yard areas is regulated and controlled by radiation zoning and access control (section 12.5). Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding. During plant operation, personnel normally gain access to radiation controlled areas through the access control building. The flow of personnel is shown in figure 12.3-10.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal operation and shutdown. Radiation zones are determined on the basis of maximum radiation level and occupancy requirements. Radiation zone categories employed and their descriptions are given in table 12.5-1 and the specific zoning for each plant area is shown in figures 12.3-2 through 12.3-9. All frequently accessed areas, i.e., corridors are shielded for Zone 1 or Zone 2 access.

Ingress or egress of plant operating personnel to controlled access areas is controlled by the plant health physics staff to assure that radiation levels and allowable working time are within the limits prescribed by 10 CFR 20.

Any area having a radiation level which could cause a whole body exposure in any 1 hour in excess of 5 mrem, or in any 5 consecutive days in excess of 100 mrem, is posted with signs bearing the radiation symbol and the words, "CAUTION, RADIATION AREA." Access alert barriers (e.g., signs, chain, rope, door,

etc.) are provided for all radiation areas. Any area having a radiation level which could cause a whole body exposure in any 1 hour in excess of 100 mrem is posted with the radiation symbol and the words, "CAUTION, HIGH RADIATION AREA." 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.

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

12.3.2 SHIELDING

In this subsection, the bases for the nuclear radiation shielding and the shielding configurations are discussed.

12.3.2.1 Design Objectives

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

Three plant conditions are considered in the nuclear radiation shielding design: normal full power operation, plant shutdown, and accident situations.

The shielding design objectives for the plant are:

- A. To ensure that radiation exposure to plant operating personnel, contractors, administrators, and visitors, are ALARA and within the dose limits of 10 CFR 20.
- B. To assure sufficient personnel access and occupancy time to allow normal anticipated operation and maintenance, inspection, and testing required for each plant equipment and instrumentation area.
- C. To reduce potential equipment neutron activation and mitigate the possibility of radiation damage to materials.

- D. Shielding of the plant for accident conditions assures that control room personnel direct and inhalation doses (calculated in chapter 15) will not exceed the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

12.3.2.2 General Shielding Design

Shielding is provided to attenuate direct radiation through walls and penetrations and scattered radiation to less than the upper limit of the radiation zone for each area. The minimum shielding requirements for all plant areas are presented in figures 12.3-2 through 12.3-9. Design criteria for penetrations comply with the intent of Regulatory Guide 8.8 and are discussed in subsection 12.3.1.

The material used for most of the plant shielding is ordinary concrete with a minimum bulk density of 2.25 g/cm^3 . In selected areas of the auxiliary building, a high density concrete (3.45 g/cm^3) is used. Whenever poured-in-place concrete has been replaced by concrete blocks or other material, design assures protection on an equivalent shielding basis as determined by the characteristics of the material selected. Compliance of concrete radiation shield design with Regulatory Guide 1.69 is discussed in appendix 3A. Water is used as the primary shield material for areas above the spent fuel transfer and storage areas.

Visual inspections of plant shielding are made during the construction phase. Because of the massive nature of the shielding, these inspections are limited to detecting major defects. Once the reactor is in operation, radiation surveys are made to ensure that:

- A. There are no defects or inadequacies in the shielding that might affect personnel exposure during normal operation and maintenance of the plant.
- B. Areas of the plant are correctly posted and barricaded in high radiation areas.

These surveys consist of both gamma and neutron monitoring, where appropriate. Continued routine radiation surveys ensure the adequacy of the shielding (see section 12.5).

12.3.2.2.1 Containment Shielding Design

During reactor operation, the containment protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete containment wall, together with the

reactor vessel and steam generator compartment shield walls, reduces radiation levels outside the containment to less than 0.5 mrem/h.

For design basis accidents, the containment reduces the plant radiation intensities from fission products inside the containment to acceptable emergency levels, as defined by 10 CFR 50 Appendix A, General Design Criterion 19, for the control room (see subparagraph 12.3.2.2.7).

Where personnel and equipment hatches or penetrations pass through the containment wall, additional shielding is provided, where necessary, to attenuate radiation to the required level defined by the outside radiation zone during normal operation and shutdown, and to acceptable emergency levels as defined by 10 CFR 50 during design basis accidents.

12.3.2.2.2 Containment Interior Shielding Design

During reactor operation, areas inside the containment are Zone 6 and normally inaccessible. Figures 12.3-2 through 12.3-6 show the inside regions of the containment Zone 6. However, shielding is provided to reduce dose rates to less than 100 mrem/h in certain areas of the containment where periodic access may be necessary (e.g., the seal table area).

The main sources of radiation are the reactor vessel and the primary loop components, consisting of the steam generators, pressurizer, reactor coolant pumps, and associated piping. The reactor vessel is shielded by the concrete reactor cavity shield which is approximately 7-foot thick, and by the concrete secondary shield, which is 4-foot thick and surrounds all other primary loop components. Air cooling is provided to prevent overheating, dehydration, and degradation of the shielding and structural properties of the primary shield.

Shielding of the reactor core at full power is done using the program ANISN (see paragraph 12.3.2.3) and assuming 2900 MWt core output. The ANISN shielding analysis uses the specific core, vessel, and shielding details pertinent to this plant and uses generally accepted core physics and transport theory methods. The analysis assumes a full power core, near beginning-of-life, with no rods and no xenon. The DLC-23D CASK cross sections library is used for the ANISN scattering data base. Layout details, such as air gap, concrete thickness, and core intervals, are conservatively incorporated into the ANISN model. Post-shutdown shielding considers the source strengths given in tables 12.2-18, 12.2-21 and 12.2-22. Shielding codes such as QAD (see paragraph 12.3.2.3) are used for these shutdown shielding analyses.

The reactor cavity is designed in such a manner as to minimize the leakage of scattered neutrons from the top of the cavity annulus near the reactor vessel flange. The design is such that above the elevation of the active core height, the radius of the primary shield narrows to close the gap between the vessel and the inside surface of the concrete. In addition, the primary shield profile generally follows the contours of the reactor vessel in the region of the vessel head flange in order to provide a more arduous path for streaming neutrons. Design considerations for the cavity to facilitate inservice inspection and adequate venting areas, in the event of design basis pipe ruptures, have been incorporated. Detailed analysis of the YGN 1 & 2 reactor cavity design utilizing three-dimensional neutron transport methods with the Monte Carlo code MORSE⁽¹³⁾ have been performed to ascertain the effectiveness of this design in reducing neutron streaming. The evaluation of this analysis indicates that the YGN 1 & 2 reactor cavity design will minimize the effect of streaming from the cavity annulus proper.

The analysis indicates that streaming into the primary shield openings such as the hot and cold leg penetrations and inspection manway is limited by the use of concrete inspection manway plugs during power operation. In addition, streaming into and out of the steam generator compartments is limited by the design of secondary shield wall penetrations. The net result of this design is that neutron levels in accessible regions will be within acceptable ranges to allow limited access during operation.

Components of the CVCS are located in shielded compartments inside the containment which are normally Zone 6 areas. Shielding is provided for each piece of equipment in the CVCS consistent with its postulated maximum activity (subsection 12.2.1) and with the access and zoning requirements of the adjacent areas. This equipment includes the excess let-down heat exchanger and the regenerative heat exchanger.

After shutdown, most of the containment is accessible for limited periods of time and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1000 mrem/h depending on the location inside the containment (excluding reactor cavity) and the time following reactor shutdown. These dose rates result from residual fission products, neutron-activated materials, and corrosion products in the reactor coolant system.

Spent fuel is the primary source of radiation during refueling. Because of the extremely high activity of the fission products contained in the spent fuel elements, extensive shielding has

been provided for areas surrounding the refueling canal to ensure that radiation levels remain below zone levels specified for adjacent areas. Water provides the shielding over the spent fuel assemblies during fuel handling.

12.3.2.2.3 Auxiliary Building Shielding Design

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

Shielding is provided for each piece of equipment consistent with its postulated maximum activity (subsection 12.2.1) and with the access and zoning requirements of adjacent areas (figures 12.3-2 through 12.3-6).

In addition, a high density concrete (3.45 g/cm^3) shielding is provided around selected components in the auxiliary building which has the potential for high radioactivity under post-accident conditions. The use of high density concrete shielding is required to minimize exposure to operating personnel requiring access to vital equipment areas. Where applicable, appropriate delay times in system operation as well as decay and dilution factors were considered in evaluating the high density concrete shielding.

High density concrete shielding is provided in the compartment walls and slabs of the following components:

- Residual Heat Removal (RHR) Pump
- RHR Heat Exchangers
- Hydrogen Recombiner
- Pipe Penetration Area
- Portion of the Hot Pipe Chase 88-ft elevation slab
- Charging Pumps
- Containment Spray Pump
- Letdown Heat Exchanger
- Nuclear Sample Station
- Moderating Heat Exchanger
- Letdown Chiller Heat Exchanger

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- Volume Control Tank
- Letdown Reheat Heat Exchanger
- Seal Water Heat Exchanger
- Auxiliary Building Demineralizer and Filter Vaults.

Depending on the equipment in the compartments, the access varies from Zones 2 through 6. Corridors are shielded to allow Zone 2 access, and operator areas for valve compartments are generally limited to Zone 4 access.

Removable sections of block shield walls and concrete plugs are utilized as necessary for equipment maintenance and spent filter cartridge replacement. Permanent or temporary shielding is used between equipment in compartments with more than one piece of equipment to permit access for maintenance.

Following reactor shutdown, the residual heat removal (RHR) system pumps and heat exchangers are in operation to remove heat from the reactor coolant system. The radiation levels in the vicinity of this equipment will temporarily reach Zone 6 levels due to corrosion and fission products in the reactor water. Shielding is provided to attenuate radiation from RHR equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas. The shielding around the RHR equipment is a minimum of 2.5 feet of concrete.

12.3.2.2.4 Fuel Building Shielding Design

The concrete shield walls surrounding the spent fuel cask loading and storage area, as well as the shield walls surrounding the fuel transfer canal and spent fuel pool, are sufficiently thick to limit radiation levels outside the shield walls in all accessible areas to Zone 2. The building external walls are sufficient to shield external plant areas to Zone 1.

Water in the spent fuel pool provides shielding above the spent fuel transfer and storage areas. Radiation levels at the fuel handling equipment are limited to less than 2.5 mrem/h during normal operations and fuel handling. The peak dose rate is minimized through operation of the spent fuel pool cleanup system and fission product decay.

The spent fuel pool cooling and cleanup (SFPPC) system (subsection 9.1.3) shielding is based on the maximum activity discussed in subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the SFPPC system to be

shielded includes the SFPCC heat exchangers, pumps and piping. (SFPCC filters and ion exchangers are located in the auxiliary building.)

12.3.2.2.5 Radwaste Building Shielding Design

Shielding is provided as necessary around the following boron recycle and radwaste systems in the radwaste building to ensure that the radiation zone and access requirements are met for surrounding areas:

- Radwaste and boron recycle system (BRS) holdup tanks
- Monitor tanks
- Chemical drain tanks and pumps
- Radwaste and BRS evaporators
- Waste solidification equipment
- Waste drumming and storage areas
- Concentrate tanks and pumps
- Radwaste and BRS piping
- Filters and demineralizers
- Spent resin tank and pumps
- Gaseous radwaste system (GRS) and compressors
- GRS carbon delay tanks.

Shielding is based upon operation with maximum activity conditions as discussed in chapter 11.

Depending on the equipment in the compartments, the access varies from Zones 2 through 6. Corridors are shielded to allow Zone 2 access, and operator areas for valve compartments are generally limited to Zone 4 access, as shown on the radiation zone drawings of figures 12.3-7 through 12.3-9.

Removable sections of block shield walls and concrete plugs are utilized as necessary to maintain equipment and replace spent filter cartridges.

12.3.2.2.6 Turbine Building Shielding Design

Radiation shielding is not required for any process equipment located in the turbine building. The condensate polishing demineralizers are arranged to allow the addition of temporary shielding walls if substantial primary to secondary system leaks develop. All areas in the turbine building are normally classified as Zone 1.

12.3.2.2.7 Control Room Shielding Design

Figure 12.3-5 shows the layout of the control room and its relationship to the containment.

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the control room. Shielding is provided to permit access and occupancy of the control room under LOCA conditions with radiation doses limited to 5 rem whole body from all contributing modes of exposure for the duration of the accident, in accordance with 10 CFR 50, Appendix A, General Design Criterion 19. A thickness of 2 feet of concrete is adequate for this purpose.

The design basis LOCA is described in subsection 15.6.5. The direct radiation from airborne fission products inside the containment and the radioactive cloud surrounding the control room contribute less than 0.15 rem to personnel inside the control room following a postulated LOCA. Thus, the shielding of the control room ensures compliance with 10 CFR 50, General Design Criterion 19, Appendix A.

In addition to the parameters of Regulatory Guide 1.4, the following are considered in the demonstration of the control room habitability: (The ventilation system parameters are listed in section 6.4.)

- A. Shielding by the containment outer wall
- B. Shielding by control room walls and ceilings
- C. Radiological decay and recirculating filtration
- D. Iodine species distribution and containment spray removal of iodine as discussed in paragraph 6.2.2.1.
- E. No credit is taken for shielding by the internal structures in the containment.

12.3.2.2.8 Diesel Generator Building Shielding Design

There are no radiation sources in the diesel generator building; therefore, no shielding is required for the building.

12.3.2.2.9 Miscellaneous Plant Areas and Plant Yard Areas

608 Sufficient shielding is provided for all plant buildings containing radiation sources so that radiation levels at the outside surfaces of the buildings are maintained below Zone 1 levels. Sufficient shielding is also provided for YGN maintenance
63 working shop including Old Reactor Vessel Head Storage Facility containing radiation sources. Plant yard areas which are frequently occupied by plant personnel are fully accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public. All outside tanks except refueling water storage tank are expected to contain concentrations of radionuclides which yield a surface dose rate of 0.5 mrem/h or less. Shielding wall or fence which limits the external dose rate of 0.5 mrem/h or less is provided for the refueling water storage tank.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in subsection 12.2.1, rather than annual average activities. The thickness of each shield wall surrounding radioactive equipment is determined by approximating, as closely as possible, the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from Martin and Blichert-Toft,⁽¹⁾ Martin,⁽²⁾ Bowman and McMurdo,⁽³⁾ Meek and Gilbert,⁽⁴⁾ and Jaeger, et al.⁽⁵⁾

The geometric model generally assumed for shielding evaluation of tanks, heat exchangers, filters, demineralizers, evaporators, piping and the containment is a finite cylindrical volume source.

A computer program (CYLSO) which calculates the gamma dose rate external to a homogeneous cylindrical source through a laminated shield is used in shielding analysis. This program utilizes the methods and equations of Rockwell's Reactor Shielding Design Manual,⁽⁶⁾ Chapter 9, sections 4.1 and 6.1. Depending upon the geometry of the problem, the actual cylindrical source is approximated to a line source or an infinite slab source of a finite thickness such that a conservative yet realistic result is obtained. Buildup is calculated using Taylor coefficients presented in ONRL-RSIC-10⁽⁷⁾ to effectively integrate the buildup over the entire source volume. Broder's method of buildup determination, presented in the Engineering Compendium on Radiation Shielding,⁽⁸⁾ is used for laminated shields.

For the shielding of gamma sources that present a more complex geometry, the point Kernal code QAD is used.

For the analysis of the primary shield, the computer code ANISN is used.

The code MORSE-CG is used to analyze neutron streaming problems. The results of this analysis is used to evaluate the design of the reactor cavity.

Descriptions of the above computer codes are included in table 12.3-1.

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 is less than this maximum dose and therefore less than the radiation zone upper limit.

Where shielded entryways to compartments containing high radiation sources are necessary, labyrinths or mazes are designed using a general purpose gamma-ray scattering code employing methods summarized in ORNL-RSIC-21.⁽⁸⁾ The mazes are so constructed that the scattering dose rate plus the transmitted dose rate through the shield wall from all contributing sources is below the upper limit of the radiation zone specified for each plant area.

12.3.3 VENTILATION

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

12.3.3.1 Design Objectives

The plant HVAC systems for normal plant operation and anticipated operational occurrences are designed to meet the requirements of National Nuclear Regulation and 10 CFR 20, Standards for Protection Against Radiation and 10 CFR 50, Licensing of Production and Utilization Facilities, as discussed in paragraph 12.3.3.2.

12.3.3.2 Design Criteria

Design Criteria for the plant HVAC systems include the following:

- A. During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in restricted areas of the plant, are as low as is reasonably achievable (ALARA) and within the limits specified in National Nuclear Regulation and 10 CFR 20 Appendix B, Table I. The average and maximum airborne radioactivity levels in unrestricted areas of the plant during normal operation and anticipated operational occurrences will be ALARA and within the limits of 10 CFR 20, Appendix B, Table II.
- B. During normal operation and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary, will be ALARA and within the limits specified in National Nuclear Regulation and 10 CFR 20 and 10 CFR 50, Appendix I.
- C. The plant siting dose guidelines of 10 CFR 100 will be satisfied following the hypothetical design basis accidents described in chapter 15.
- D. The dose to control room personnel shall not exceed the limits specified in General Design Criterion 19 of Appendix A to 10 CFR 50 following those hypothetical accidents described in chapter 15.

12.3.3.3 Design Guidelines

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

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

- A. Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.
- B. Equipment vents and drains are piped directly to a collection device connected to the collection system instead of allowing any contaminated fluid to flow across the floor to the floor drain.

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- C. All-welded piping systems are employed on contaminated systems to the maximum extent practicable to reduce system leakage.
- D. The valves in some systems are provided with leak-off connections piped directly to the collection system.
- E. Suitable coatings are applied to the concrete floors and walls of potentially contaminated areas to facilitate decontamination.
- F. 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 may be piped to a drain header. Metal diaphragm or bellows seal valves are used on those systems where essentially no leakage can be tolerated,
- G. 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. The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination,
- B. In building compartments with a potential for contamination, a greater volumetric flow is exhausted from the area than is supplied to the area to minimize the amount of uncontrolled exfiltration from the area.
- C. Consideration is given to the possible disruption of normal airflow patterns by maintenance operations, and provisions are made in the design to prevent adverse airflow direction.
- D. Air cleaning systems criteria are discussed in detail in the response to Regulatory Guides 1.52 and 1.140 found in Appendix 3A. An illustrative example of the air cleaning system design is given in paragraph 12.3.3.5.

- E. Air being discharged from potentially contaminated areas is passed through high efficiency particulate air (HEPA) filters and charcoal adsorbers to remove particulates and halogens or means are provided to isolate these areas upon indication of contamination to prevent the discharge of contaminants to the environment.
- F. Suitable containment isolation valves are installed in accordance with General Design Criteria 54 and 56, including valve controls, to assure that the containment integrity is maintained. See additional discussion in section 3.1.
- G. Redundant, Seismic Category I systems and/or components are provided for portions of the ventilation system required for safe shutdown of the reactor plant. Included herein are the plant main control room and selected engineered safety feature equipment rooms.
- H. Atmospheric tanks which contain radioactive materials are vented to the respective building exhaust ventilation system.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC equipment

- A. Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The HVAC system is designed to allow rapid replacement of components.
- B. Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts to the extent practicable.
- C. Ventilating air is recirculated in clean areas only.
- D. Access and service of ventilation systems in potentially radioactive areas is provided by component location to minimize operator exposure during maintenance, inspection, and testing as follows:

The outside air supply units and building exhaust system components are enclosed in ventilation equipment rooms. These equipment rooms are located in radiation Zone 2 and are accessible to the operators. Work space will be provided around each unit for anticipated maintenance, testing and inspection. Filter-adsorber units generally comply with the access and service

requirements of Regulatory Guide 1.52. (Refer to response to Regulatory Guide 1.52 in Appendix 3A.)

- E. The exhaust ventilation systems contain fully or partially redundant units. If any maintenance is done on any portion of the HVAC ventilation system, the redundant units will provide adequate ventilation for the work area.

No maintenance on other radioactive equipment which could potentially result in spread of airborne contaminants will be performed until the restoration of HVAC ventilation exhaust system capacity in the maintenance area.

12.3.3.4 Design Description

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

- A. Containment building (see subsection 9.4.7)
- B. Auxiliary building (see subsection 9.4.2)
- C. Fuel building (see subsection 9.4.6)
- D. Radwaste building (see subsection 9.4.3)
- E. Turbine building and switchgear building (see subsection 9.4.4)
- F. Access control building (see subsection 9.4.5).

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

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

12.3.3.5 Air Cleaning System Design

The guidance and recommendations of Regulatory Guides 1.52 and 1.140 concerning maintenance and in-place testing provisions for atmospheric cleanup systems, air filtration and adsorption units have been used as a reference in the design of the various ventilation systems. The extent to which Regulatory Guide 1.52 has been followed is discussed also in Appendix 3A, section 9.4 and subsection 6.5.1.

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

- A. The loading of the filters and adsorbers with radioactive material during normal plant operation is a slow process; therefore, in addition to monitoring for pressure drop, the filters 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. The performance efficiencies of these filter elements are for the control room 99.5 percent, the auxiliary building 99.5 percent, and the containment 99.5 percent. The performance of the adsorbers is periodically tested by testing the sample cartridge at the laboratory. Filters whose radioactivity level (due to a postulated accident) is such that a change of filter elements would constitute a personnel hazard, can be removed intact. No shielding is provided since it is not required for the level of radioactivity developed during normal operation. In case of excessive radioactivity caused by a postulated accident, the whole filter is replaced before normal personnel access is resumed. It will not be necessary for workers to handle filter units immediately after a design basis accident so that exposures can be minimized by allowing the shortlived 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. Minimum distances for access and servicing filters are maintained in accordance with Regulatory Guides 1.52 and 1.140. No filter bank is more than three filter units high, where each filter unit is 2 by 2 feet. The access to the level or platform at which the filter is serviced is by stairs or elevators.
- E. The clear space for doors is a minimum of 3 by 4.5 feet.
- F. The filters are designed with replaceable 2 by 2 foot 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.

- G. The maximum allowable pressure drop across the moderate efficiency filter is 1 inch WG. For HEPA filters the maximum is 3 inches WG. When the pressure drop across a filter reaches these limits, the filter will be changed.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Area radiation and airborne radioactivity monitoring systems are provided to assess radiation levels at various in-plant locations,

12.3.4.1 Area Radiation Monitoring System

The area radiation monitoring system (ARMS) is designed to continuously monitor gross beta-gamma radiation at selected locations throughout the plant. The ARMS consists of various radiation monitors that provide operating personnel with a continuous indication, locally and in the control room, of radiation levels at selected locations within various plant buildings where radioactive materials may be present, stored, handled, transported, or inadvertently introduced. The containment refueling pool area radiation monitor and the fuel building spent fuel pool area radiation monitor perform safety-related functions to actuate systems which mitigate the consequences of postulated fuel handling accidents.

The ARE supplements the personnel and area radiation survey provisions of the plant health physics program described in section 12.5 and ensures compliance with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50 and Regulatory Guides 8.2, 8.8, and 8.12. The ARMS monitors post-accident radiation levels are in compliance with Regulatory Guide 1.97.

12.3.4.1.1 Design Bases

12.3.4.1.1.1 Safety Design Bases

12.3.4.1.1.1.1 Safety Design Basis One. The digital radiation monitoring systems (D&S) shall be capable of identifying the occurrence of a fuel handling accident in the containment or fuel buildings and to initiate an actuation signal for the appropriate engineered safety features (ESF) systems.

12.3.4.1.1.1.2 Safety Design Basis Two. The DRMS required to meet safety design basis one shall be designed to remain functional following a safe shutdown earthquake (SSE).

12.3.4.1.1.1.3 Safety Design Basis Three. The DRMS equipment required to meet safety design basis one shall be capable of withstanding a single active failure.

12.3.4.1.1.2 Power Generation Design Basis

- A. Local detectors are designed to function properly in an ambient temperature range of 40 to 120°F.
- B. Local detectors are designed to function properly in 0 to 100 percent relative humidity.
- C. Local detectors in the auxiliary, shared radwaste, control, fuel, containment, and access control buildings operate at normal atmospheric pressures. Detectors located in the containment building are removed temporarily during containment test pressures (these are not containment high range area monitors listed in paragraph 12.3.4.1.1.2.F).
- D. Area monitors are of a nonsaturating design and will not produce a decreasing output signal when a detector is exposed to an increasing radiation field above its range.
- E. All monitor alarms are electronic and are continuously adjustable over the entire instrument range to ensure compliance with 10 CFR 50, Appendix A, General Design Criterion 13.
- F. High range containment building area monitors are designed to function following a loss-of-coolant accident (LOCA), and will not be adversely affected by periodic containment integrity pressure tests.
- G. All area radiation monitors are powered from the instrumentation ac buses, with the exception of the spent fuel pool area, the refueling pool area and the high range containment area monitors which are powered from the Class 1E instrument ac power system.
- H. Areas that contain a liquid, gaseous, or particulate radiation source that potentially could produce a dose rate during normal operation greater than

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2.5 mrem/h are provided with an area monitor unless one of the following conditions exists:

1. Another area monitor in the vicinity is capable of monitoring the area in question; i.e., there is line-of-sight access between the monitor and the area in question, and the monitor alarm point would be the same value for all areas served.
 2. The frequency of personnel access to the area is minimal, which is defined as access required only for infrequent repairs, unscheduled maintenance, or periodic surveillance. Portable monitors are used to monitor such areas during personnel access.
 3. The probability of accidental release within the area is minimal: i.e., an area with only sealed containers or where the material in the space during normal operations has a low activity level. Examples include the volume control tank area and the boron recycle tank area.
 4. Airborne radioactivity monitors are provided that perform a function equivalent to an area radiation monitor in an area in which gaseous or airborne particulate activity is the major constituent. An example would be the shared radwaste building ventilation systems which are provided with gaseous process monitors to monitor for leakage from the waste gas compressors and waste gas charcoal delay tank valves.
- I. The ARMS is designed to continuously monitor gross beta-gamma radiation at selected locations throughout the plant. The ARMS consists of multiple channels that provide operating personnel with a continuous indication, locally and in the control room, of radiation levels at selected locations within various plant buildings. The area monitors listed in table 12.3-2 accomplish the following design objectives:
1. Assist plant operators in decisions on deployment of personnel in the event of an accident resulting in a release of radioactive material in the plant.

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2. Warn of unauthorized or inadvertent movement of radioactive material in the plant.
3. Warn of abnormally high radiation levels in selected areas of the plant.
4. Provide local alarms and readouts where a substantial change in radiation levels might be of immediate importance to personnel frequenting or working in an area.
5. Comply with the requirements of 10 CFR 50, Appendix A, General Design Criterion 63, for monitoring fuel and waste storage and handling areas.
6. Provide criticality alarm monitors and conform to the requirements of Regulatory Guide 8.12.
7. Provide monitoring for use in assuring that solidified radwaste containers meet the requirements of shipping and storage with respect to the radiation level on contact with the container and at certain distances from the container. This monitor will warn personnel when containers need additional shielding or when shielding limits have been reached.

Each area monitor indicates and alarms locally and in the control room. Monitor alarm set points are determined by the normal background radiation at the detector location, limits for personnel exposure in restricted areas, and the calculated levels for normal operating conditions in the area.

The spent fuel pool bridge and the containment refueling machine bridge area monitors perform an additional safety function. A high radiation signal in the fuel building initiates closure of the normal fuel building ventilation system and activates the fuel building emergency ventilation system. A high radiation signal in the containment initiates containment building purge isolation. These area monitors provide redundancy to the fuel building ventilation exhaust monitor and the containment purge exhaust radiation monitor described in section 11.5.

The containment high range area monitors perform two safety functions. These monitors will initiate containment purge isolation upon detecting increasing radiation levels typical of a fuel handling accident. This capability is in addition to that provided by refueling

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machine bridge area monitor and complies with Section II.E.4.2 of NUREG-0660. The containment high range area monitors are qualified for operation under post-LOCA environment. These monitors provide post-accident indication of radiation level inside the containment. This capability complies with Section II.F.1. of NUREG-0660.

- J. Each detector can be checked with a long half-life radiation check source to verify equipment operation. The energy emission ranges of the check source are similar to the energy spectra being monitored.
- K. Each panel module has at least one level trip and one failure trip alarm device. The level trip indicates high radiation and the failure trip indicates channel failure.
- L. To minimize occupational radiation exposure to maintenance personnel, the ARMS is designed for rapid replacement of components, calibration and installation. For a general discussion of ALARA techniques, refer to paragraph 12.1.3.1.

12.3.4.1.2 System Description

The ARMS consists of multiple channels that provide operating personnel with a continuous indication, locally and in the control room, of radiation levels at selected locations within various plant buildings, where radioactive materials may be present, stored, handled, transported, or inadvertently introduced. The ARMS plus certain monitors in the airborne radioactivity monitoring system act together to provide surveillance in the containment, fuel building, auxiliary building, shared radwaste building, control room, radiochemical laboratory, hot machine shop, turbine building, and drum storage building. Infrequently accessed radiation areas which are not monitored continuously are monitored with portable devices prior to and during access.

Each channel of the ARMS consists of a fixed-position Geiger-Mueller (G-M) tube or ion chamber detector with appropriate electronics to amplify the detector signal, to supply electric power to the instruments, to provide local and remote readouts of the radiation level, and to alarm on a high radiation level. A remote circuit power failure alarm has been provided.

The radiation energy in the monitored area is detected by the G-M tube or ion chamber and is converted into electric signals. These pulses are carried to the control room for displaying of the radiation dose rate level. The radiation level is displayed at the local control unit and in the control room. The range of the ARMS monitors is indicated in table 12.3-2. Abnormal radiation

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levels are annunciated by visual and audible alarms, both locally and in the main control room. However, each radwaste building monitor has three control room readouts, one in the main control room panel of each unit and one in the radwaste building control room. Monitor alarm set points are selected on the basis of the normal background radiation at the detector location, limits for personnel exposure in restricted areas, and the calculated levels for normal operating conditions in the area.

Each channel is checked for trend changes each shift, amplifier electronics are tested monthly, and each instrument is calibrated at refueling shutdowns. In addition, calibration is performed following any maintenance that affects system performance or if periodic tests indicate instrument drift,

Area radiation monitors are provided in the following locations:

- A. Fuel building new fuel storage area
- B. Auxiliary building sample room
- C. Radwaste building truck loading bay
- D. Radwaste building drumming control station
- E. Spent fuel pool
- F. Radwaste building control room
- G. Radwaste sample room
- H. Hot machine shop
- I. Main control room
- J. Containment seal table
- K. Containment operating deck
- L. Containment refueling pool
- M. Containment building, elevation 163 ft.
- N. Radiochemistry lab

RADIATION PROTECTION
DESIGN FEATURES

The spent fuel pool and the containment refueling pool area monitors perform an additional safety function. A high radiation signal in the fuel building initiates closure of the normal fuel building ventilation system and activates the fuel building emergency ventilation system. A high radiation signal in the containment initiates containment building purge isolation. These area monitors provide redundancy for the fuel building ventilation exhaust radiation monitor and the containment purge exhaust radiation monitor.

Two area radiation monitors are used in conjunction with the radwaste solidification system. These monitors monitor the radiation level in the waste feed tank and the 55 gallon drum while it is being filled. The indication from these monitors is displayed on the RSS control panel. These monitors are independent of the digital radiation monitoring system.

The ARMS meets the requirements of Regulatory Guide 1.97. The high range containment area radiation monitors will be provided as discussed in Section II.F.1 of NUREG 0660.

12.3.4.1.2.1 Detector Assembly. The detector is a G-M tube or ion chamber housed in a watertight container, sealed to allow decontamination with water and/or solvents. The detectors are operated from the local alarm-readout unit and are suitable for wall mounting.

12.3.4.1.2.2 Local Alarm-Readout Unit. The local digital alarm-readout units housed in a watertight container, sealed to permit decontamination by water and/or solvents. The local alarm-readout unit contains the following:

- A. Power-on indicator
- B. Readout meter
- C. High-radiation alarm lights and status lights
- D. One audible alarm with an auxiliary contact

12.3.4.1.2.3 Control Room Readout. The main control room readout is provided via a CRT. In addition, the safety-related area radiation monitors are provided with Class 1E digital and analog meters and are continuously recorded.

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The electronic components for the readout module are rack-mounted plug-in units and each monitoring unit is identified for maintenance or alignment while energized.

12.3.4.1.2.4 Readout Module Front Panel. Specifications for the front panel of each control room readout module are:

Readout meter

High-level alarm light

Warning-level alarm light

Failure light

Alarm reset (manual)

Check source lamp

Power-on indicator

Shutdown L.E.D

Alarm level adjustment over the entire range of the channel

Channel identification nameplate includes channel tag number

Reset of remote audible alarm for each remote or local alarm readout unit

The readout device will not go to zero during periods of saturation of the detector.

12.3.4.1.2.5 Readout Module Outputs. Each control room readout module provides the following independent outputs:

396 | Digital recorder output

Indicator analog output

Control room contact output

Alarm level

Warning level

12.3.4.1.2.6 Check Source. A remotely operated, long half-life radiation check source is furnished in each channel. The energy emission ranges are similar to the radiation energy spectra being monitored. The source strength is sufficient to cause easily perceivable positive scale response.

A tabulation of the related monitor, channel range, and location is given in table 12.3-2.

12.3.4.1.3 Safety Evaluation

12.3.4.1.3.1 Safety Evaluation One. The refueling machine bridge area monitor and the spent fuel pool area radiation monitor are designed to activate ESF systems in the event of a fuel handling accident.

12.3.4.1.3.2 Safety Evaluation Two. The spent fuel pool and refueling machine bridge area monitors are designed to meet the instrumentation requirements of IEEE Standard 279 and are classified as Seismic Category I. These monitors are powered from Class 1E power supplies.

12.3.4.1.3.3 Safety Evaluation Three. Redundancy for the spent fuel pool area radiation monitor is provided by the fuel handling building HVAC exhaust gas monitor described in section 11.5.

Redundancy for the refueling machine bridge area monitor is provided by the containment purge exhaust gas monitor described in section 11.5.

12.3.4.2 Airborne Radioactivity Monitoring System

The in-plant airborne radioactivity monitors consist of the following monitors:

- A. Fuel handling building HVAC exhaust gas monitor
- B. Radwaste building HVAC exhaust filter inlet particulate monitor
- C. Gaseous radwaste system Combined ventilation gas monitor
- D. Radwaste building HVAC exhaust filter outlet gas monitor and particulate sampler
- E. Control room HVAC normal intake gas monitor
- F. Auxiliary building HVAC normal exhaust filter inlet gas, particulate and iodine monitor
- G. Auxiliary building HVAC normal exhaust filter outlet particulate and iodine sampler

- H. Containment purge exhaust gas monitor and particulate and iodine sampler
- I. Containment atmosphere gas, particulate and iodine monitor
- J. Gaseous radwaste discharge gas monitor
- K. Condenser air ejector and vacuum pump gas exhaust monitors with particulate and iodine samplers.

These monitors are discussed in section 11.5 and are tabulated in table 11.5-1.

12.3.4.2.1 Design Bases

12.3.4.2.1.1 Safety Design Basis. The safety design basis of the airborne radioactivity monitors is presented in paragraph 11.5.1.1.

12.3.4.2.1.2 Power Generation Design Basis. The power generation design basis for the airborne radioactivity monitors is presented in paragraph 11.5.1.2.

12.3.4.2.2 System Description

The description of the airborne radioactivity monitors is contained in subsection 11.5.2.

12.3.5 REFERENCES

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Table 12.3-1

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

CYLSO	Multigroup, line source gamma-ray, shield design code which computes the dose rate for a cylindrical source with slab shields. ⁽⁹⁾
ENDCYL	Multigroup, Kernel integration gamma-ray, shield design code which computes the dose rate for a cylindrical source with slab shields. ⁽¹⁰⁾
ANISN	Multigroup, multiregion code solving the Boltzman transport equation for neutrons or gamma-rays in one dimension slab, cylindrical, or spherical geometry. ⁽¹¹⁾
QAD	Multigroup, multiregion, three-dimensional, point Kernel code which calculates fast neutron and gamma-ray dose and heat generation rates. ⁽¹¹⁾
MORSE-CG	Three-dimensional Monte Carlo neutron and gamma-ray general transport code. ⁽¹³⁾

Shielding calculations are performed using the methods and information found in Walker and Grotenhuis, ⁽¹⁴⁾ Goldstein,⁽¹⁵⁾ ANL-5800⁽¹⁶⁾ Kircher and Bowman,⁽¹⁷⁾ Rockwell,⁽⁶⁾ Tipton,⁽¹⁸⁾ Soodak, ⁽¹⁹⁾ and Blizzard and Abbott.⁽²⁰⁾

Table 12.3-2
AREA RADIATION MONITORS (Sheet 1 of 2)

Monitor	Quantity per Unit	Type	Seismic Category	Medium Monitored	Temp Range (°F)	Pressure Range	Power Source	Range (mrem/hr)	High Alarm Actuation Set Point (mrem/hr)	Function
Containment Operating Deck Area Monitor (GT-RE-133)	1	GM	NA	Air	60-120	Atm	Non-Class 1E 120V ac	1-10 ⁵	250	Monitor operating deck and personnel access hatch
Containment Seal Table Area Monitor (GT-RE-132)	1	GM	NA	Air	60-120	Atm	Non-Class 1E 120V ac	1-10 ⁵	250	Monitor incore instrument handling area
Refueling Machine Bridge Area Monitor (GT-RE-220)	1	GM	1	Air	60-120	Atm	Class 1E 120V ac (Channel B)	0.1-10 ⁴	250	Monitor refueling machine service area and isolate containment purge on high radiation signal
Spent Fuel Pool Area Monitor (GG-RE-113)	1	GM	1	Air	50-104	Atm	Class 1E 120V ac (Channel A)	0.1-10 ⁴	10 ³	Monitor fuel handling operations; actuate emergency ventilation on high radiation signal monitors criticality accident
Fuel Bldg New Fuel Storage Area Monitor (GG-RE-018)	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	12.5	Monitor fuel handling operations monitors for criticality accident
Control Room Area Monitor (GK-RE-054)	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	2.5	Monitor radiation level in control room
Radiochemistry Lab Area Monitor (GX-RE-009)	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	12.5	Monitor radio-chemistry laboratory radiation level
Hot Machine Shop Area Monitor (GH-RE-162)	1 ^(a)	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	12.5	Monitor hot machine shop radiation level

2

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Table 12.3-2
AREA RADIATION MONITORS (Sheet 2 of 2)

Monitor	Quantity per Unit	Type	Seismic Category	Medium Monitored	Temp Range (°F)	Pressure Range	Power Source	Range (mrem/hr)	High Alarm Actuation Set Point (mrem/hr)	Function	
Radwaste Drumming Station Area Monitor (GH-RE-061)	1 ^(a)	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	25	Monitor solid radwaste drumming station radiation levels	384
Containment High Range Area Monitor (GT-RE-001 GT-RE-002)	2 2	GM IC	I	Air	60-300	0-69 psig	Class 1E 120V ac (Channel A and B)	0.1-10 ⁴ 10 ³ -10 ¹⁰	10 ⁴	Monitor fuel handling operations, initiate containment purge isolation, and monitor post-LOCA conditions	2
Radwaste Truck Loading Bay Area Monitor (GH-RE-060)	1 ^(a)	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	1-10 ⁵	100	Monitors truck loading by radiation level	
Radwaste Control Room (GH-RE-062)	1 ^(a)	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	12.5	Monitor radwaste control room radiation level	
Radwaste Sample Room Area Monitor (GH-RE-063)	1 ^(a)	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	12.5	Monitor radwaste sample room radiation level	
Auxiliary Building Sample Room Area Monitor (GL-RE-087)	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120V ac	0.1-10 ⁴	100	Monitor sample room radiation level	8 384
ORVH Storage Area Monitor(269-RE-001)*	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120 V ac	0.1-10 ⁴	500	ORVH Storage room radiation level	
Decontamination Room Area Monitor(269-RE-002)*	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120 V ac	0.1-10 ⁴	400	Decontamination room radiation level	608
Truck Bay Area Monitor (269-RE-003)*	1	GM	NA	Air	50-104	Atm	Non-Class 1E 120 V ac	0.1-10 ⁴	100	Truck Bay radiation level	

Table 12.3-3

EQUIPMENT SPECIFICATION LIMITS FOR COBALT IMPURITY LEVELS

Component	Material	Maximum Weight Percent Cobalt
Reactor internals (non-active region)	SS*	0.20
Reactor internals (active region)	SS	0.12
Reactor vessel clad	SS	0.20
Reactor coolant piping	SS	0.20
Reactor internal bolting material	SS	0.25
Reactor coolant pumps	SS	0.20
Pressurizer	SS	0.20
Steam generators	Inconel	0.10
Fuel(nonactive region)	SS	0.12
Fuel(active region)	SS	0.08
Fuel	Inconel	0.10
Fuel	Zircaloy	0.002

*SS = Stainless steel

Table 12.3-4

EQUIPMENT REACTOR COOLANT SYSTEM WETTED SURFACE AREAS

Component	Material	Surface Area (ft ²)
Reactor internals	SS*	4236
Reactor vessel clad	SS	2190
Reactor coolant piping	SS	2758
Reactor internal bolting material	SS	Negligible
Reactor coolant pumps	SS	Negligible
Steam generators	Inconel	1.90×10^5
Fuel(nonactive region)	SS	2000
Fuel(active region)	SS	3600
Fuel	Inconel	7.80×10^3
Fuel	Zircaloy	7.78×10^4

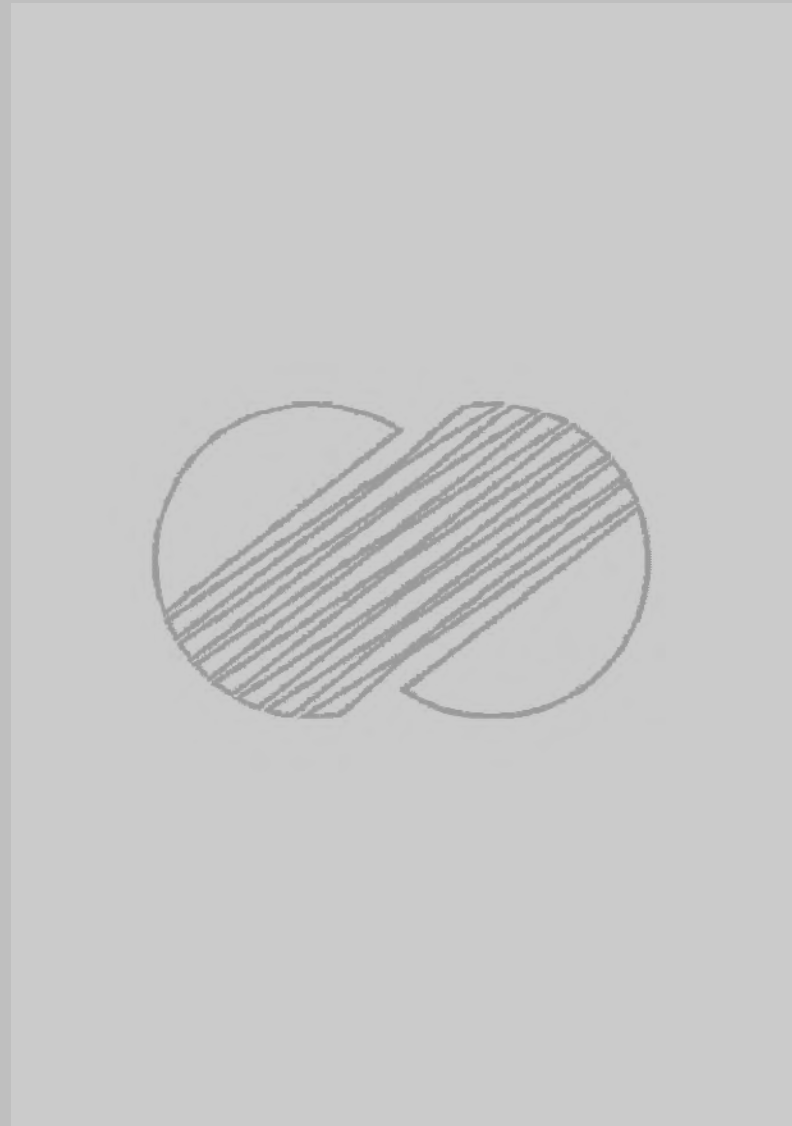
SS* = Stainless steel

Table 12.3-5

APPROXIMATE REACTOR COOLANT SYSTEM WETTED
SURFACE AREA OF STELLITE

Component	Surface Area (ft ²)
Reactor internals	3.2
Reactor coolant pump journals	17.2
Control rod drive mechanisms	10.8
Reactor coolant system valves	2.6





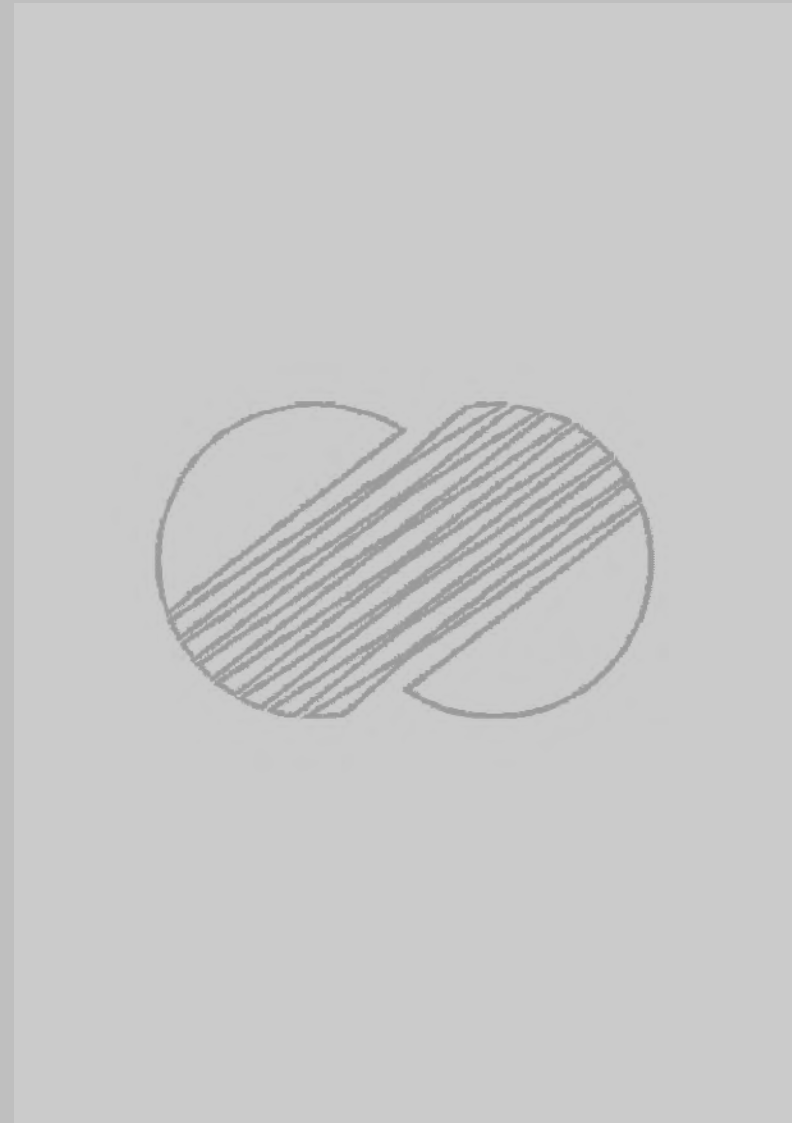
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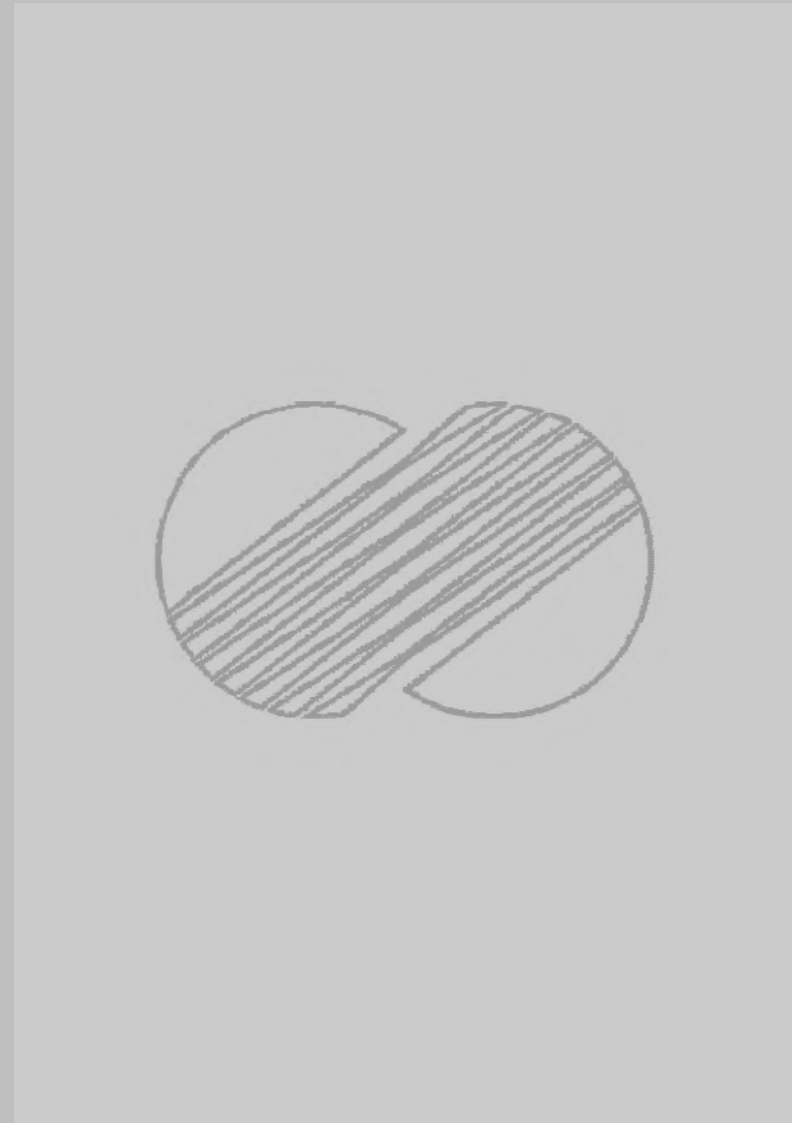


KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 7 & 8
FSAR

TYPICAL VALVE COMPARTMENT
ARRANGEMENT
(Sheet 1 of 3)

Figure 12.3-1





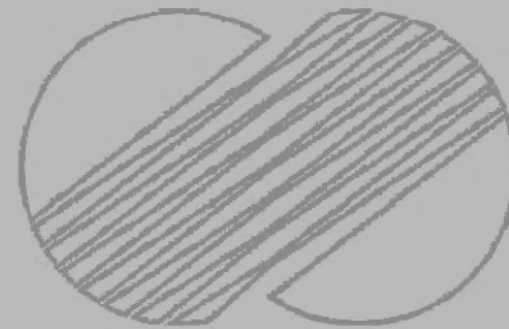


KOREA HYDRO & NUCLEAR
POWER COMPANY
YGN 1&2 FSAR

RADIATION ZONES PLAN BELOW GRADE

FIGURE 12.3-2






KOREA HYDRO & NUCLEAR
POWER COMPANY
YGN 1&2 FSAR

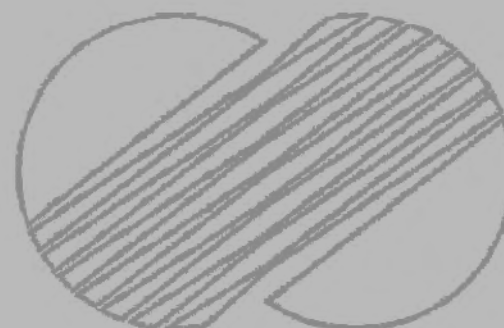
RADIATION ZONES PLAN AT GRADE


FIGURE 12.3-4




	KOREA HYDRO & NUCLEAR POWER COMPANY YGN 1&2 FSAR
RADIATION ZONES PLAN AT INTERMEDIATE LEVEL	
FIGURE 12.3-5	

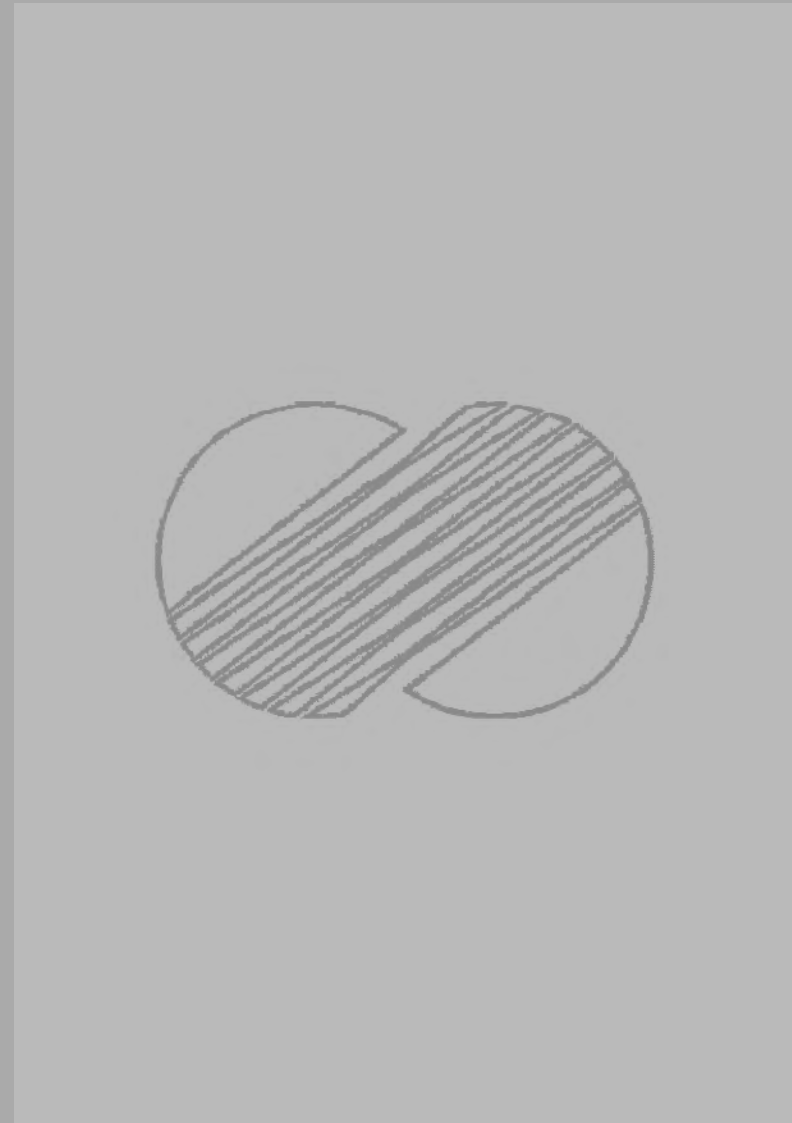




	KOREA ELECTRIC POWER CORPORATION KOREA NUCLEAR UNITS 7 & 8 FSAR
	RADIATION ZONES RADWASTE BUILDING (EL.100'-0" and 68'-0") Figure 12.3-7

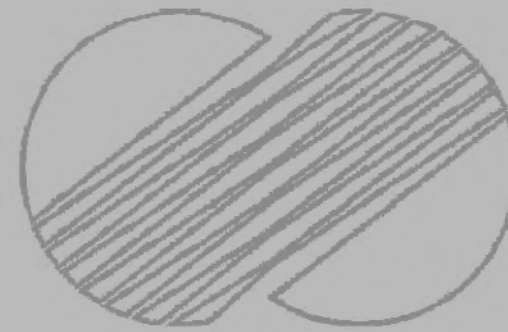


	KOREA ELECTRIC POWER CORPORATION
	KOREA NUCLEAR UNITS 7 & 8
	FSAR
RADIATION ZONES RADWASTE BUILDING	
(EL.118'-0")	
Figure 12.3-8	

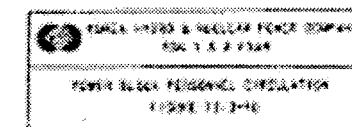


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12.4 DOSE ASSESSMENT

Radiation exposures in the plant are primarily due to direct radiation from components and equipment containing radioactive fluids. In addition, in some plant 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 due to direct radiation at locations outside the plant structures, such as the boundary of the restricted areas, are a function of the plant layout, equipment selection, and detailed system and shielding designs and are expected to be negligible. Radiation exposures due to the airborne radioactive effluent plume at these locations are expected to be insignificant. The radiation exposures at these locations are discussed in subsection 12.4.2.

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

Radiation exposure to construction workers in the Unit 2 construction area due to the operation of Unit 1 is estimated to be much less than National Nuclear Regulation and 10 CFR 20 limits for occupational workers and less than the National Nuclear Regulation and 10 CFR 20 limits for the population and is discussed in subsection 12.4.3.

12.4.1 EXPOSURES WITHIN THE PLANT

12.4.1.1 Direct Radiation Dose Estimates

Determination of annual man-rem doses to plant personnel from direct radiation depends upon a multiplicity of factors such as occupancy times, the number of personnel involved in operations, and the radiation levels in plant areas.

The radiation exposure estimates were developed from exposure models for each of the major jobs. Each exposure model has been developed by breaking the job into individual tasks and identifying expected radiation fields, time spent in each radiation field, and number of men required for each task. The estimates for each task are based primarily on feedback from operating plants. The feedback used to develop these estimates includes frequency of the operation, background exposure rate, contact exposure rate, time required for the operation, number of men required for the operation, and total exposure accumulated during the operation.

The time required for a given task includes ingress and egress time from the specified radiation area as well as orientation time and setup time for the job. In many cases, the exposure models have been developed from a number of radiation/time studies from different plants. Engineering judgment has been used to define typical values for each parameter in the exposure model. As such, the resultant exposure estimates should be used as typical values keeping in mind the variability of the input data from which the estimates were calculated. In many cases, the design differs from that at operating plants from which the operating data has been accumulated. In these cases, engineering judgment has been applied to the field data to develop adjusted exposure estimates for the new system and component designs.

All the parameters required to make a reasonable estimate of man-rem doses for the entire plant will strongly depend upon the design and operating conditions of the plant. However, based on operating plant experiences of pressurized water reactors (PWRs) between the years 1970-1974 and the distribution of personnel and man-rem doses according to function for light water reactors in 1974 (taken from Murphy⁽¹⁾ and given in tables 12.4-1 and 12.4-2), an annual exposure of 435 man-rem/year-unit is estimated.

For YGN 1 and 2, exposure to plant personnel from direct radiation during the performance of routine functions is estimated to be approximately 437 man-rem/year-unit. Details of the man-rem estimates are given in table 12.4-3. A breakdown of the estimated exposures by work functions is provided in table 12.4-4.

The maximum and expected average dose rates in the plant radiation areas are given in table 12.4-5.

Total occupancy time for personnel involved in different work functions in various radiation zones is shown in table 12.4-3.

12.4.1.2 Exposures Due to Airborne Radioactivity

Exposures to workers in the accessible areas of the plant due to airborne radionuclides are expected to be insignificant due to plant design features such as airflow patterns, which prevent spreading of contaminated air to normally accessible areas of the plant, and operating procedures that are described in sections 12.3 and 12.5. Consequently, all the normally accessible areas will have airborne concentrations well below National Nuclear Regulation and 10 CFR 20 limits for occupational workers.

Table 12.4-6 gives the predicted peak air concentrations for the containment building, auxiliary building, radwaste building, and turbine building. The model and assumptions used to determine the airborne concentrations are discussed in subsection 12.2.2. Table 12.4-7 gives the estimated doses from air concentrations in each of these locations. The calculational methods used to estimate the doses are taken from Regulatory Guide 1.109.

12.4.2 EXPOSURES AT LOCATIONS OUTSIDE PLANT STRUCTURES

12.4.2.1 Direct Radiation Dose Estimates

The direct radiation from the containment, auxiliary, radwaste, and turbine buildings is negligible compared to that from outside storage tanks. The principal sources of radioactivity not stored in the plant structures are the reactor makeup water storage tank, the refueling water storage tank, and the condensate storage tank.

These tanks are expected to contain concentrations of radionuclides which yield a surface dose rate of 0.5 mrem/hr or less. The dose rate at the nearest site boundary, based on 8760 hours occupancy, is less than 100 mrem/yr.

12.4.2.2 Exposures Due to Airborne Radioactivity

Doses at the site boundary due to released activity are given in subsection 11.3.3.

12.4.3 EXPOSURES TO CONSTRUCTION WORKERS

12.4.3.1 Direct Radiation and Dose Estimates.

The estimated dose rates from direct radiation received by construction workers on Unit 2 due to the operation of Unit 1 are well within the limits of 10 CFR 20 for exposure to individuals in unrestricted areas.

The estimated dose rates are the sum of the direct radiation from the Unit 1 containment, auxiliary building, turbine building, radwaste building, and the outside storage tanks. The annual individual dose and the total man-rem dose are estimated by using occupancy times based on projected construction manpower requirements and the following assumptions:

- A 80 percent of the annual workers are located inside areas of construction of major plant buildings and

structures. The remaining 20 percent are outside of plant structures.

- B. Construction workers assumed to be located outside are exposed to cloud immersion doses as well as direct shine doses from major structures and contaminated ground.
- C. Construction workers are assumed to be on site for 2500 h/yr (50 h/wk for 50 wk/yr).
- D. One hundred percent reactor power with a capacity factor of 80 percent and 1.0 percent failed fuel.
- E. Once the major plant buildings and structures are completed, workers inside them or behind them with respect to Unit 1 sources receive negligible direct radiation due to shielding provided by the concrete and steel in those structures.
- F. Construction personnel on site during last year of Unit 2 construction is estimated to be 2,000 worker.

Based on the above assumptions, the total dose from direct shine to all construction workers during the last year of construction is estimated to be 79 man-rem due to the operation of Unit 1. The annual dose to a construction worker is estimated to be a maximum of 198 mrem.

Section 20.202 of 10 CFR 20 specifies that personnel monitoring equipment would be required if the maximum expected dose per calendar quarter for workers in an area would exceed 300 mrem. It was determined that, even in areas with the highest radiation levels, no construction worker would receive a dose greater than this, so personnel monitoring equipment will not be necessary. However, periodic radiation surveys will be made by the radiation protection staff. Personnel dosimetry devices will be located in areas where construction personnel are working to verify that no person will receive a dose greater than 500 mrem/yr for their total exposure.

12.4.3.2 Exposures Due to Airborne Radioactivity

Based on expected annual releases of gaseous effluents and a conservative annual average atmospheric dispersion factor of 6.2×10^{-5} sec/m³ for Unit 1 at the construction site and an annual occupancy of 2500 h (50 h/wk for 50 wk/yr) the average whole body gamma, beta skin and thyroid doses to a construction worker have been estimated to be 3.6, 5.3 and 13.9 mrem/yr respectively, during the construction stage.

12.4.4 REFERENCES

1. T. D. Murphy, et al., NUREG-75/032, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974, USNRC Radiological Assessment Branch, June 1975.
2. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purposes of Evaluating Compliance with 10 CFR 50, Appendix I, "Regulatory Guide 1.109, Revision 1, October 1977.

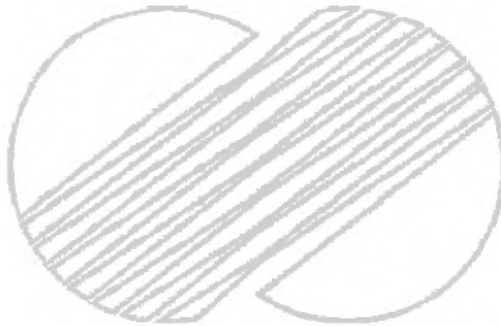


Table 12.4-1
DATA FROM OPERATING PWR PLANTS (Sheet 1 of 2)

Year	Plant	Designed Power Level (Mwe)	Average Annual Power (Mwe)	Total No. of Personnel	Total Annual Dose (man-rem)
1970	Connecticut Yankee	575	424.7	734	689
	San Onofre - Unit 1	450	365.9	251	155
1971	Connecticut Yankee	575	502.2	289	342
	Ginna	490	327.8	340	430
	San Onofre - Unit 1	450	362.1	121	50
1972	Connecticut Yankee	575	515.6	355	325
	Ginna	490	295.6	677	1,032
	Point Beach - Unit 1	497	378.3	NA ^(a)	580
	Robinson	707	580.0	245	215
	San Onofre - Unit 1	450	372.2	326	256
1973	Connecticut	575	293.1	841	673
	Ginna	490	409.6	421	244
	Pallisades	821	286.6	901	1,109
	Point Beach - Units 1 & 2	497,	693.7(total	729	570
	(2nd Unit, 4/73)	497	for 1 & 2)		
	Robinson	707	455.1	831	695
	San Onofre - Unit 1	450	273.1	878	329
1974	Connecticut Yankee	575	519.1	550	201
	Ginna	490	253.7	884	1,224

a. NA : Not available

Table 12.4-1
DATA FROM OPERATING PWR PLANTS (Sheet 2 of 2)

Year	Plant	Designed Power Level (Mwe)	Average Annual Power (Mwe)	Total No. of Personnel	Total Annual Dose (man-rem)
1974 (cont)	main Yankee	790	432.6	620	420
	Oconee - Unit	886	724.3	844	517
	Pallisades	821	10.5	774	627
	Point Beach - Units 1 & 2	497, 497	760.2(total for 1 & 2)	400	295
	Robinson	707	578.1	853	672
	San Onofre	450	377.8	219	71
	Surry - Units 1 & 2 (Unit 2 - 5/73)	823, 823	717.4(total for 1 & 2)	1,715	884
	Turkey Point - Units 3 & 4	745, 745	966.4(total for 3 & 4)	794	454
	Unit 4 - 9/73				

Table 12.4-2

YEARLY AVERAGES AND GRAND AVERAGE FOR NUMBER OF PERSONNEL
AND MAN-REM DOSES FOR OPERATING PWR PLANTS ^(a)

Year	No. of Units	Total No. of Personnel	Total man-rem Dose	Average No. of Personnel	Average man-rem Dose/Unit
1970	2	985	844	493	422
1971	3	750	822	250	274
1972	5	1,603 ^(b)	2,408	401	482
1973	7	4,601	3,620	657	517
1974	13	7,653	5,365	589	413
1970-74	30	15,592 ^(c)	13,059	538	435

- a. This table is based on the data given in table 12.4-1.
- b. The entry corresponds to four plants only, since no information on personnel is available for Point Beach - Unit 1.
- c. The entry corresponds to a total of 29 plants only.

Table 12.4-3
ESTIMATED OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA
DOSES TO PLANT PERSONNEL (Sheet 1 of 3)

Operation	Zone	Percentage of Occupancy	hr/yr	Hourly Dose Rate (rem/hr)	Yearly Dose Rate (rem/yr)	No. of Men (2 Units)	Annual Exposures (man-rem/yr-unit)
Operation and Surveillance	1	75	1500	1.25×10^{-4}	0.188	160	15.04
	2	19	380	6.25×10^{-4}	0.238	160	19.04
	3	2	40	1.25×10^{-3}	0.05	160	4.0
	4	3.3	66	5×10^{-3}	0.33	160	26.4
	5	0.6	12	2.5×10^{-2}	0.3	160	24.0
	6	0.1	2	1×10^{-1}	0.2	160	16.0
Total		100	2000		1.31	160	104.8
Routine Maintenance	1	80.5	1610	1.25×10^{-4}	0.201	220	22.11
	2	12.1	242	6.25×10^{-4}	0.151	220	16.61
	3	1	20	1.25×10^{-3}	0.025	220	2.75
	4	5	100	5×10^{-3}	0.5	220	55.0
	5	1	20	2.5×10^{-2}	0.5	220	55.0
	6	0.4	8	1×10^{-1}	0.8	220	88.0
Total		100	2000		2.18	220	239.8
Radwaste Processing	1	40	600	1.25×10^{-4}	0.075	20	0.75
	2	43	645	6.25×10^{-4}	0.403	20	4.03
	3	10.5	157.5	1.25×10^{-3}	0.197	20	1.97
	4	5	75	5×10^{-3}	0.375	20	3.75
	5	0.75	11.25	2.5×10^{-2}	0.281	20	2.81
	6	0.75	11.25	1×10^{-1}	1.125	20	11.25
Total		100	1500		2.456	20	24.56

Table 12.4-3
ESTIMATED OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA
DOSES TO PLANT PERSONNEL (Sheet 2 of 3)

Operation	Zone	Percentage of Occupancy	hr/yr	Hourly Dose Rate (rem/hr)	Yearly Dose Rate (rem/yr)	No. of Men (2 Units)	Annual Exposures (man-rem/yr-unit)
Refueling	1	30	96	1.25×10^{-4}	0.012	52	0.312
	2	40	128	6.25×10^{-4}	0.08	52	2.08
	3	4	12.8	1.25×10^{-3}	0.016	52	0.416
	4	20	64	5×10^{-3}	0.32	52	8.32
	5	5	16	2.5×10^{-2}	0.4	52	10.4
	6	1	3.2	1×10^{-1}	0.32	52	8.32
Total		100	320		1.148	52	29.85
Inservice Inspection (ISI)	1	55	176	1.25×10^{-4}	0.022	30	0.33
	2	32	102.4	6.25×10^{-4}	0.064	30	0.96
	3	4	12.8	1.25×10^{-3}	0.016	30	0.24
	4	7	22.4	5×10^{-3}	0.112	30	1.68
	5	1	3.2	2.5×10^{-2}	0.08	30	1.2
	6	1	3.2	1×10^{-1}	0.32	30	0.48
Total		100	320		0.614	30	4.89
Special Maintenance	1	75	300	1.25×10^{-4}	0.0375	20	0.375
	2	18	72	6.25×10^{-4}	0.045	20	0.45
	3	2	8	1.25×10^{-3}	0.01	20	0.1
	4	3	12	5×10^{-3}	0.06	20	0.6
	5	1	4	2.5×10^{-2}	0.1	20	1
	6	1	4	1×10^{-1}	0.4	20	4
Total		100	400		0.653	20	6.53

Table 12.4-3
ESTIMATED OCCUPANCY TIMES IN PLANT RADIATION AREAS AND GAMMA
DOSES TO PLANT PERSONNEL (Sheet 3 of 3)

Operation	Zone	Percentage of Occupancy	hr/yr	Hourly Dose Rate (rem/hr)	Yearly Dose Rate (rem/yr)	No. of Men (2 Units)	Annual Exposures (man-rem/yr-unit)
Health Physics	1	75	1500	1.25×10^{-4}	0.188	20	1.88
	2	19	380	6.25×10^{-4}	0.238	20	2.38
	3	2	40	1.25×10^{-3}	0.05	20	0.5
	4	3.3	66	5×10^{-3}	0.33	20	3.3
	5	0.6	12	2.5×10^{-2}	0.3	20	3.0
	6	0.1	2	1×10^{-1}	0.2	20	2.0
Total		100	2000		1.31	20	13.1
Radio- chemistry	1	75	1500	1.25×10^{-4}	0.188	20	1.88
	2	19	380	6.25×10^{-4}	0.238	20	2.38
	3	2	40	1.25×10^{-3}	0.05	20	0.5
	4	3.3	66	5×10^{-3}	0.33	20	3.3
	5	0.6	12	2.5×10^{-2}	0.3	20	3.0
	6	0.1	2	1×10^{-1}	0.2	20	2.0
Total		100	2000		1.31	20	13.1

Table 12.4-4

DISTRIBUTION DIRECT RADIATION MAN-REM DOSES
ACCORDING TO WORK FUNCTIONS

Operation	Annual Exposures (man-rem/yr-unit)	Percentage
Operation and Surveillance	104.8	24.0
Routine Maintenance	239.8	54.92
Radwaste Processing	24.56	5.62
Refueling	29.85	6.84
Inservice Inspection	4.89	1.12
Special Maintenance	6.53	1.5
Health Physics	13.1	3.0
Radiochemistry	13.1	3.0
Total	436.63	100

Table 12.4-5

THE MAXIMUM AND EXPECTED AVERAGE DOSE RATES
IN THE PLANT RADIATION AREAS

Zone	Maximum Dose Rate (mrem/hr)	Expected Average Dose Rate (mrem/hr) ^(a)
1	0.5	0.125
2	2.5	0.625
3	5	1.25
4	20	5
5	100	25
6	>100	100

- a. The expected dose rates are estimated by assuming 25% of design dose rates.

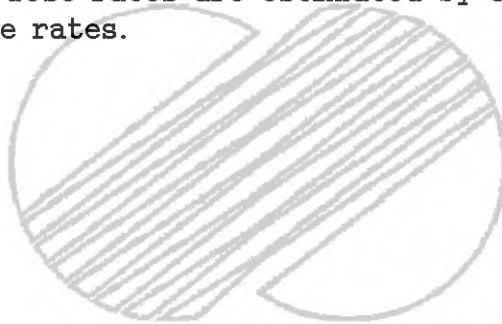


Table 12.4-6

EXPECTED AIRBORNE RADIOACTIVITY
CONCENTRATIONS ^(a) ($\mu\text{Ci}/\text{cm}^3$)
(Sheet 1 of 2)

Isotope	Ventilation Area			
	Turbine Building	Auxiliary Building	Radwaste Building	Containment ^(b) ^(c)
H-3	9.22×10^{-10}	3.68×10^{-8}	6.08×10^{-8}	6.44×10^{-7}
Ar-41	—	—	—	1.73×10^{-7}
Kr-83m		9.35×10^{-10}	1.22×10^{-9}	5.54×10^{-8}
Kr-85m	—	5.34×10^{-9}	7.11×10^{-9}	4.77×10^{-7}
Kr-85	—	2.75×10^{-10}	3.76×10^{-10}	4.35×10^{-8}
Kr-87	—	2.4×10^{-9}	3.09×10^{-9}	1.23×10^{-7}
Kr-88	—	9.66×10^{-9}	1.27×10^{-8}	6.74×10^{-7}
Xe-131m	—	8.44×10^{-10}	1.15×10^{-9}	1.31×10^{-7}
Xe-133m	—	5.26×10^{-9}	7.17×10^{-9}	7.64×10^{-7}
Xe-133	—	2.45×10^{-7}	3.34×10^{-7}	3.73×10^{-5}
Xe-135m	—	2.05×10^{-10}	2.53×10^{-10}	6.98×10^{-9}
Xe-135	—	1.7×10^{-8}	2.3×10^{-8}	1.82×10^{-6}
Xe-138	—	6.43×10^{-10}	7.92×10^{-10}	2.16×10^{-8}

Table 12.4-6

EXPECTED AIRBORNE RADIOACTIVITY
CONCENTRATIONS^(a) ($\mu\text{Ci}/\text{cm}^3$)
(Sheet 2 of 2)

Isotope	Ventilation Area			
	Turbine Building	Auxiliary Building	Radwaste Building	Containment ^{(b) (c)}
I-131	2.2×10^{-14}	1.62×10^{-10}	2.2×10^{-10}	2.49×10^{-9}
I-132	6.85×10^{-15}	5.26×10^{-11}	6.89×10^{-11}	3.38×10^{-10}
I-133	3.27×10^{-14}	2.42×10^{-10}	3.28×10^{-10}	3.12×10^{-9}
I-134	2.12×10^{-15}	1.72×10^{-11}	2.19×10^{-11}	7.75×10^{-11}
I-135	1.62×10^{-14}	1.22×10^{-10}	1.63×10^{-10}	1.18×10^{-9}

(a) Based on 0.12% failed fuel

(b) Average containment tritium concentration during refueling is $6.09 \times 10^{-7} \mu\text{Ci}/\text{cm}^3$, average fuel building tritium concentration during refueling is 1.39×10^{-6} .

(c) Values given are during full power operation.

Table 12.4-7

ESTIMATED AIRBORNE RADIOACTIVITY EXPOSURE ^(a)

Location	Maximum Anticipated Individual Occupancy Times	Inhalation Lung Dose (rem/yr)	Inhalation Thyroid Dose (rem/yr)	Skin Dose (rem/yr)	Whole Body Gamma Dose (rem/yr)	Tritium Dose (rem/yr)
Containment	94 hr/yr (2 hr/entry, 47 entry/yr)	8.24×10^{-3}	5.47×10^{-1}	5.20×10^{-1}	2.81×10^{-1}	3.11×10^{-2}
Containment (Refueling H-3 Only)	168 hr/yr (56 hr/wk, 3 wk/yr)	—	—	—	—	5.26×10^{-2}
Auxiliary Building	300 hr/yr (6 hr/wk, 50 wk/yr)	1.83×10^{-3}	1.19×10^{-1}	2.00×10^{-2}	9.48×10^{-3}	5.67×10^{-3}
Radwaste Building	2000 hr/yr (40 hr/wk, 50 wk/yr)	1.65×10^{-2}	1.07	1.76×10^{-1}	8.39×10^{-2}	6.25×10^{-2}
Fuel Handling H-3 Only	1000 hr/yr (20 hr/wk, 50 wk/yr)	—	—	—	—	7.14×10^{-1}
Turbine Building	2000 hr/yr (40 hr/wk, 50 wk/yr)	1.65×10^{-6}	1.05×10^{-4}	1.93×10^{-7}	1.32×10^{-7}	9.48×10^{-4}

a. Based on 0.12% failed fuel

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

12.5.1.1 Program Organization

The YGN 1 & 2 organization is shown in subsection 13.1. The plant manager is responsible for radiation protection, contamination control, and decontamination of the plants. The Manager, Radiological Control Section, is responsible for administering the plant radiation protection program which encompasses the handling and monitoring of radioactive materials, including special nuclear source, byproduct materials, and contaminated materials. He is also responsible for assuring that the plant operation meets the radiation protection requirements of National Nuclear Regulation and NRC Regulatory Guides that are applicable to the health physics program throughout section 12.5. The corporation commitment to the philosophies embodied in the above documents and the authority to implement them are discussed in subsection 12.1.1.

The plant Health Physicists who are the assistants to the Manager of Radiological Control Section are experts in implementing radiation protection program. They establish and control the radiation work procedure in order to keep the radiation exposure as low as reasonably achievable (ALARA).

| 2

The health physics technicians perform the various surveys for radiation protection and the sample collection and analyses for radioactive waste.

For more detailed discussion of the responsibilities and authority of the supervisory positions mentioned above and the training and qualifications of the personnel presently holding these positions, refer to subsections 13.1.2 and 13.1.3, respectively.

12.5.1.2 Program Objective

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 is within the guidelines of National Nuclear Regulation and that such exposure is kept ALARA.

Administrative control directives will be prepared to ensure that all procedures and requirements are followed by plant personnel.

Limits are developed consistent with the National Nuclear regulation for protection against radiation. The radiation protection manual is designed to ensure that all procedures and requirements relating to radiological protection are uniformly and consistently followed by plant personnel.

- B. To provide administrative control over any plant effluent releases to ensure that these releases are below National Nuclear Regulation.

12.5.1.3 Radiation Protection Program

The plant radiation protection program will be officially initiated at YGN 1 and later at YGN 2, when radioactive material licensed to YGN 1 & 2 is first brought into the respective reactor vessel, and will be in effect continuously thereafter until the units are decommissioned. This program consists of rules, practices and procedures that are used to accomplish the objectives stated above in a practical and safe manner. The program is basically consistent with the recommendations of NRC Regulatory Guides 8.1, 8.2, 8.8, and 8.10.

The radiation protection program will ensure that:

- A. Personnel receive appropriate radiation protection training.
- B. Appropriate access control techniques and protective clothing are used to limit external contamination.
- C. Respiratory protection equipment is used where needed to limit inhalation of radioactive material.
- D. Radiation areas are segregated and appropriately posted to limit radiation exposure.
- E. Instrument and equipment is properly calibrated so that accurate radiation, contamination, and airborne activity surveys can be performed.
- F. Appropriate personnel dosimetry devices are supplied.
- G. An internal dose assessment program (whole body counting and/or bioassay) is supplied.

- H. Incoming and outgoing shipments of radioactive materials are properly handled.
- I. Necessary measures are performed to keep radiation exposures ALARA with safely supplying a reliable source of power to the public.

The detail of the procedure will be discussed in subsection 12.5.3.

The program also assures that appropriate effluent release samples are collected and analyzed in consistent with the appropriate plant procedures to verify that the plant has little radiation effect on the environment and, near site population. In addition, the program assures that the emergency plan can be properly implemented, to limit the consequences of any accidents at the plant as discussed in section 13.3.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

12.5.2.1 Controlled Area

The plant design establishes a controlled area for each unit of the plant separately. The controlled area includes all areas in which radioactive materials are present or potentially present in quantities sufficient to require protective measures, and includes all areas designated as radiation Zone 2 or greater, as shown in table 12.5-1.

The normal ingress to and egress from the controlled area is through the health physics station set up at the access control point in the controlled access building (see figure 12.5-1) and is controlled by the radiation protection personnel. All other potential access points to the controlled area are kept locked or sealed except emergency exit doors. The emergency exit doors to the controlled area have locking devices in order to prevent unauthorized access from outside and allow rapid exit from the controlled area in case of emergency. The supervisory alarm system for emergency exit doors is provided both in the Main Control Room and in the Health Physics station. Temporary controlled access area maybe established in the clean area of the plant and is subject to all rules and procedures of the controlled area. A radiation monitor is provided at the access control point and is used by all persons leaving the area to check themselves for contamination.

The controlled area is extended to include portions of the back-yard area to allow access for trucks to the fuel and radwaste buildings. The extended controlled area boundary is maintained by fencing, rope, or other continuous barricades. The backyard areas are not used to store contaminated materials. Personnel requiring entry, such as a truck driver, are provided with protective clothing, radiation dosimetry devices, and continual escort while in the controlled area. Upon leaving, the vehicle

and personnel are monitored, and a record of the results is maintained. Only personnel who have received a radiation protection orientation or who are escorted by an individual who has received this orientation are permitted to have access to controlled areas.

Individuals permitted by a radiation work permit (RWP) to enter radiation controlled areas are provided with a radiation monitoring device which continuously indicates the dose rate in the area. The RWP is designed to inform the individual of radiation conditions in and adjacent to the area, specify the protective clothing and monitoring device requirements for entry, and specify the maximum occupancy time permitted. Any pertinent additional information is provided on the RWP.

High radiation areas are conspicuously posted and are maintained with a locked barrier which prevents unauthorized access. Personnel entering high radiation areas should receive radiation protection instruction by a health physician at the issuance of a RWP.

12.5.2.2 Facilities Related to Radiation Protection

Facilities for radiation protection are conveniently located for ingress to and egress from potentially contaminated areas of the plant. These facilities include the following:

- A. A sampling room where primary coolant samples are drawn
- B. A radiochemistry laboratory where radioactive samples are chemically analyzed and/or prepared for radiochemical analyses. The laboratory is maintained at negative pressure to contain any airborne or gaseous radioactive materials inside and is equipped with constant airflow fume hoods and radiation monitor
- C. A counting room where radioactive samples are analyzed for isotopic composition and activity levels. It is shielded and air-conditioned to reduce background levels and data fluctuations.
- D. A storage area where bulk quantities of chemicals and laboratory supplies are stored
- E. Radiation protection office and access control checkpoint
- F. A locker room where personnel change into clean work clothes and anticontamination clothing as required

- G. Personnel decontamination facility with showers and a washroom where personnel are monitored for contamination and appropriate measures taken
- H. A clean clothes storage facility
- I. Fixed area radiation monitoring system
- J. An access control station where personnel exiting from the controlled area are checked for radiation contamination prior to leaving the area. The access control station includes portal monitors and frisking probes.
- K. An instrument decontamination facility with equipment handling facilities and a hot instrument shop to provide an area for repair and maintenance of contaminated instrument.

12.5.2.3 Special Shielding

Access to areas with continuous radiation levels in excess of 100 mrem/h are restricted by a locked door or gate, constructed to allow rapid entry or exit in case of emergency. Special shielding materials, such as lead brick or sheet, are provided to personnel working in radiation areas and are used to reduce exposures whenever reasonable and practical. Special equipment, such as remote tools and handling equipment, and shielded transfer casks are provided and used for normal radioactive material handling (i.e., filter changing). Abnormal and nonroutine handling of radioactive materials is planned on a case-by-case basis and special shielding or tools are utilized to the extent practical to limit radiation exposure to personnel.

12.5.2.4 Radiation Protection Instrumentation

12.5.2.4.1 Laboratory Radiation Detection Instrumentation

The laboratory radiation instrumentation located in the counting room includes the following instruments:

- A. G-M counter with G-M detector
- B. Automatic planchet counting system
- C. Two-pi gas flow counter
- D. Liquid scintillation counter.
- E. **[DELETE]**

- F. NaI (Tl) multichannel analyzer
- G. Hyper-pure germanium multichannel analyzer
- H. Automatic TLD reader
- I. Mini-pulser and mini-scaler.

12.5.2.4.2 Portable Radiation Detection Instrumentation

The portable radiation detection instruments are stored in a health physics station. The portable monitoring instruments include the following:

- A. Teletectors
- B. Portable beta-gamma counters
- C. Fast-slow neutron counters
- D. Ratemeter scalers
- E. Portable alpha counters
- F. Radiation monitors
- G. Portable micro-R meters

12.5.2.4.3 Portable Air Sampling Instrumentation

The portable air sampling instrumentation includes:

- A. Alpha air monitors
- B. Beta air monitoring system
- C. Regulated air samplers.

The air samplers are used to collect grab samples of radioactive particulates and halogens for subsequent analysis in the laboratory. These samplers are used for periodic sampling of localized areas prior to entry by operations or maintenance personnel. The continuous air monitors collect and measure gross activity concentrations of airborne radioactive particulates, halogens, and tritium. These monitors are stationed in airborne radioactivity areas during personnel occupancy, and warn of increasing airborne

radioactivity levels. The continuous air monitors can also be employed for routine surveys of gross airborne radioactivity levels throughout the plant.

12.5.2.4.4 Personnel Radiation Monitoring Instrumentation

The radiation monitoring instrumentation includes :

- A. Beta gamma dosimeters
- B. Thermal neutron pocket dosimeters
- C. Fast neutron pocket dosimeters
- D. [DELETE]
- E. Personal air samplers.

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12.5.2.4.5 Emergency Instrumentation

Portable equipment is kept in the control room and in the visitors center for access in the event of an emergency. These instruments are calibrated and checked periodically according to the procedure to assure their proper functioning. They include :

- A. A wide-range G-M survey meter
- B. A low level contamination detection instrument
- C. A portable air samplers
- D. Self-reading dosimeters
- E. Various respirators

12.5.2.4.6 Calibration of Radiation Protection Instrument

The following instruments are tested and calibrated semiannually and after each repair by the radiation protection technicians using the proper calibration facility. But, for some instruments which are calibrated by the Calibration Laboratories authorized by the provision of Article 15 of the Weights and Measures Act, The calibration period can be determined by the Laboratories. Repair is performed by the instrument and control technicians or radiation protection technicians.

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- A. Portable radiation detection instruments
- B. Air samplers
- C. Personnel monitoring instruments
- D. Emergency instruments.

12.5.2.5 Equipment Decontamination Facilities

Decontamination areas are provided in the containment, fuel building, radwaste building, and in the hot machine shop. In the containment, a washdown pad and permanent spray nozzle are provided in the reactor head laydown area for decontamination of the inside of the reactor vessel head. In addition, hose and drain connections are provided for decontamination of the refueling cavity liner following refueling.

In the fuel building, a cask decontamination pit and permanent spray nozzles are provided to decontaminate the spent fuel shipping cask. The radwaste building has a decontamination station in the solid radwaste area for locally decontaminating the outside of drums which contain solidified waste. In the hot machine shop, a centralized decontamination facility is provided for the decontamination of tools and equipment. The central decontamination facility will contain spray, both ultrasonic bath and chemical bath.

Typically, components of process systems will be decontaminated in place prior to maintenance or equipment removal. Each process component; e.g., pumps, heat exchangers, and filters which contain a significant quantity of radioactive fluid, is provided with flush and drain connections. These components will be flushed in place prior to maintenance. If necessary, a portable pump and drum of decontamination chemicals may be used to circulate the decontamination through the component using the flush and drain connections. If further decontamination of components is necessary, they will be enclosed in poly bags and transported to the central decontamination facility.

Decontamination of large components will have to be treated on a case-by-case basis due to the varied configurations and locations of equipment. Where necessary, temporary curtains can be placed around components requiring in-place decontamination.

Components in potentially radioactive areas are coated with decontaminable coatings to facilitate cleanup. If necessary, solvents or sand-blasting will be employed to completely remove coatings for maximum decontamination. Maintenance of large equipment that is impossible to do in the plant building can be performed in YGN Maintenance Working Shop.

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12.5.3 PROCEDURES

12.5.3.1 Radiation and Contamination Surveys

12.5.3.1.1 Policy

The procedures for radiation and contamination survey will be established prior to the initial core-loading. These procedures will specify the conditions, requirements and area for the routine radiation survey and non-routine special survey. The purpose of the survey is to collect and to check the radiation and/or contamination level of various plant area,

and the information collected will be used as the reference guideline for radiation workers in performing their job.

12.5.3.1.2 Responsibility

The Health Physics technicians will conduct the routine survey and keep the records. The plant Health Physicist will review such survey results to take appropriate action. The manager, Radiological Control Section, would take the overall responsibility.

12.5.3.1.3 Types of Surveys

A. Radiation level

Periodic general radiation surveys are performed in clean and controlled areas as frequently as necessary, depending on the type, use, and potential hazard of the area; or whenever radiation conditions are uncertain or changing. Special radiation surveys could be conducted depending on the necessity. Certain radiation conditions in the working area may require continuous survey during the radiation work.

B. Contamination

Periodic contamination survey (smear survey), which is an evaluation of removable surface contamination, is made in clean and controlled areas depending on the type, use, and potential hazard of the area, and also whenever contamination levels is uncertain. Specific smear survey is made upon request to evaluate and determine safe working conditions for specific jobs.

C. Air

Periodic airborne radioactivity survey, which is an evaluation of the concentration of airborne radioactivity present in any area, is made in clean and controlled area depending on the type, use, and potential hazard of the area, or whenever the presence of airborne contamination is uncertain. Unscheduled air sampling is made upon request.

D. Water

Periodic water sampling is performed to analyze the activity concentration present in order to evaluate the plant operation status and the potential personal hazard.

12.5.3.2 Procedures and Methods to Maintain Exposures ALARA

- 2 | Procedures for access control to radiation areas or potential radiation areas are developed by Radiological Control Section Staff and will be well acquainted to all radiation workers. The radiation exposure to individual workers will be carefully investigated and logged in order to provide the guidelines to keep the radiation exposure as low as reasonably achievable. The plant Health Physicist will establish the ALARA program.

12.5.3.3 Controlling Access

12.5.3.3.1 General

- 2 | Persons not thoroughly familiar with the controlled area procedures are escorted by health physics technicians or those who have passed the senior radiation training course, or will be instructed with proper protection guideline to assure adequate radiation protection. Certain controlled area is posted with signs bearing such words as, "Authorized Entry Only." These areas may be entered only by individuals who have obtained proper clearance in the form of a RWP. The purpose of the RWP is to control access to these areas and to limit exposure and contamination problems by advising the workers of radiation and contamination level and of the protective clothing requirement or of other requirements to perform their job safely.

12.5.3.3.2 Entry into the Controlled Area

The following requirements are met prior to entry:

- A. TLD badges, dosimeters, protective clothing and other required personal monitors are worn as specified by the appropriate RWP.
- B. Nobody is allowed to enter the controlled area with open wounds exposed to the air. All open wounds are sealed with a waterproof bandage prior to entry.
- C. All personnel who have not received a radiation protection orientation are escorted by someone who has demonstrated his knowledge in this area to the satisfaction of the Manager, Radiological Control Section, or his designated alternate.
- D. Entry normally is through the access control point. Entrance to the controlled area via any other route must be authorized by the plant manager or his designee.

12.5.3.3.3 Exit from the Controlled Area

The following procedure is followed upon exiting from the controlled area:

- A. Exit is made through the access control point only. Exit via other routes must be authorized by the plant manager or his designee.
- B. All protective clothing is removed at the step-off area.
- C. Before leaving the monitor room area, personnel monitor themselves for possible contamination.
- D. All contaminated personnel report to the Health Physics Group. Decontamination is effected prior to leaving the area.

12.5.3.4 Contamination Control

12.5.3.4.1 Facility Contamination Control

Contamination of general plant area by the movement of personnel between areas is controlled by using the step-off pad technique. A double step-off pad is employed for jobs involving high levels of contamination. Plastic bags and absorbent paper are used to move contaminated tools and equipment between areas. G-M count rate meters (friskers) are located at each step-off pad so that personnel can check themselves prior to entering another area of the plant. The final checkpoint for all personnel leaving all restricted areas of the plant is the access control point where portal monitors are located.

12.5.3.4.2 Personnel Contamination Control

12.5.3.4.2.1 Protective Clothing. Contamination of personnel is controlled by the use of several types of protective clothing when entering contaminated areas.

- A. Lab coat is worn by laboratory personnel during radioactive sample analysis.
- B. Coverall is worn in most instances when entering contaminated areas.
- C. Cloth shoe covers are worn in areas where dry contamination is encountered. In the case of wet contamination, either plastic or rubber shoe covers are worn.
- D. Cloth gloves are worn in areas where dry contamination is encountered. Rubber or plastic gloves are worn in the event of wet contamination.

- 2 | E. Vinyl or/and plastic suits are worn over cloth coveralls in areas where the potential exists for liquid contamination of personnel.
- F. Cloth cap is worn for low level contamination, cloth hoods are for high level contamination, and plastic hoods are for wet contamination.
- G. Protective clothing is worn by personnel when performing maintenance in contaminated areas. In cases where two sets of protective clothing are worn, a double step-off pad is used.

Normally, most of the plant area is accessible to personnel without protective clothes. As a result, and to minimize the area in which protective clothing is required, temporary change areas are set up adjacent to the work areas for special maintenance jobs. Also, permanent change areas are established for areas routinely requiring protective clothing. If, at any time, the number of separate areas requiring protective clothing becomes large enough to make travel about the plant cumbersome, or if general plant contamination occurs, the locker room adjacent to access control becomes the main change area for the entire restricted part of the plant.

12.5.3.5 Airborne Activity Control

The proper respiratory equipment is required under H.P. decision prior to entering the airborne contamination area. It is the responsibility of the Health Physics Group to survey the area and to specify the required protective equipment according to the concentration and type of airborne contaminant. It is the responsibility of the individual and his supervisor to notify the Health Physics Group when an airborne problem is likely to exist, i.e., working with radioactive materials. Air contamination is kept to a minimum through the use of proper ventilation and decontamination of equipment and work areas. Respiratory protective devices may be required to prevent internal exposure in the area where airborne radioactivity exists. In such cases, radiation protection personnel sample the air and recommend the type of respiratory device to be worn. Respiratory devices provided include half-face respirators with particulate and/or iodine filters, full-face respirators with particulate and/or iodine filters, and supplied air breathing apparatus.

12.5.3.6 Personnel Monitoring

12.5.3.6.1 Policy

This portion of the radiation protection program deals with the wearing of proper personnel monitoring devices, accurate recording

of dose received, proper evaluation of the exposure and medical and bioassay examinations and whole-body counting as required. Proper personnel monitoring devices for the purpose of this procedure shall mean thermoluminescent dosimetry, and electronic dosimeters. All radiation workers are issued with TLD and must wear monitoring devices as specified while within the radiation controlled area. | 94

12.5.3.6.2 Plant Personnel Exposure

12.5.3.6.2.1 External Dosimetry. Thermoluminescent dosimeters (TLD) badges are issued to all personnel entering in the controlled area of the plant. The badge has its own identification number and is assigned to each personnel. Badges must be worn at all times while within the controlled area and are put in the designated rack located within the plant office building upon leaving the plant; they may be picked up there again before entering the plant. | 94

Badges are normally processed on a monthly basis. They are also processed as required if an individual has been involved in an emergency incident, or any time exposure to an individual is questionable. In this event, the individual is restricted from further exposure until his TLD is read and evaluation of the situation has been made. The skin and whole-body external exposure is logged upon measuring the personnel TLD.

All personnel having occasion to enter the controlled area or anyone else designated by the manager, Radiological Control Section, or his designated alternate, is issued a self-reading electronic dosimeter. This is identified and stored in a manner similar to the TLD badge. | 94

All regularly assigned dosimeters are normally read, recorded, and set to zero by the user at least daily.

All radiation workers are given a complete baseline physical examination before commencement of work. All other personnel who randomly enter the controlled area of the plant should have a physical examination before entrance.

12.5.3.6.2.2 Internal Dosimetry. The internal deposition of radioactive materials in personnel working in controlled area of the plant is evaluated by whole Body Counting. | 94

Urinalysis is performed in accordance with the tritium

concentration levels in working area and as deemed necessary
21 by the manager, Radiological Control Section.

The urinalysis will also be used for the internal dose assessment. The plant procedure will dictate to all radiation workers to be whole-body scanned at least once a year. Special non-routine scanning may be requested by plant H.P. whenever significant internal radiation exposure is suspected. The over-exposure will be reported to the Plant Nuclear Safety Committee that will initiate further detail investigation.

12.5.3.7 Radioactive Materials Safety Program

The storage, handling, transportation, and disposal of radioactive materials is described in the plant procedure which assures compliance with all applicable regulations so that personnel are not exposed, unnecessary radiation.

12.5.3.7.1 Receiving Radioactive Material

Whenever the order for radioactive material is initiated from plant personnel, the Health Physics Group will be informed of the type and amount of material requested, the activity, and the physical and chemical form of the material. Shipment containing radioactive material from offsite sources must comply with all applicable regulations for packaging and labeling. Before the package of radioactive material is transported inside the site, the plant H.P. technicians will conduct an external radiation survey; the package will be placed in a suitable location and be ensured for the proper labeling. The manager, Radiological Control Section, also assumes responsibility for reporting, all instances of broken, leaking, or defective shipping containers to the applicable agencies. If any contamination is identified, the driver will immediately be notified and decontamination will be performed under the supervision of H. P. Only the radioactive material package within the allowable radiation and contamination level will be permitted to be transported into the site area.

12.5.3.7.2 Storing Radioactive Material

The radioactive material will be stored in the special area designated by the Health Physics Group. All radioactive materials entering and leaving the area are logged and the storage area itself is labeled so that it is clearly recognized as a radioactive material area by all personnel. The storage facility of the radioactive sources is in compliance with National Regulatory Requirements for radioisotope usage.

12.5.3.7.3 Onsite Transfer of Radioactive Material

Onsite transfer of radioactive material is rigorously controlled to minimize exposure. Radioactive materials that could cause unnecessary exposure to personnel are not moved into a work area unless the personnel in that area have been advised of the move. Radioactive material transfer containers for onsite transfer are constructed and/or shielded so that leakage or breakage does not readily occur. The shielding must be adequate to protect personnel in the area and those engaged in transporting the material.

All radioactive material is tagged and/or labeled properly on the container prior to transfer, when the possibility of personnel exposure is deemed as significant.

12.5.3.7.4 Fuel Handling, Storage, and Shipment

The receipt, inventory (including location), disposal, and transfer of all fuel, new and spent, is in accordance with the 10 CFR 70, Special Nuclear Material, and applicable regulations.

The Health Physics Group is responsible for surveys, both radiation and contamination, of all fuel prior to or during unpacking and storage. Surveys of the shipping containers also are performed before shipment from the site.

12.5.3.8 Radiation Protection Training

The objective of radiation protection training is to enable all personnel to safely carry out assignments involving potential exposure to radiation. Training program is designed to cover the subjects in the depth required by various individuals. Each program covers the basic subjects, but additional material is covered according to the level of knowledge required for the individual to accomplish his job assignment safely.

The Manager, Radiological Control Section, is responsible for the radiation protection training of YGN 1 & 2 employees and other individuals assigned to the station. It is his responsibility to ensure that all personnel assigned to YGN 1 & 2 and other personnel working with radioactive materials or in controlled areas are adequately trained. A record is kept of all individuals trained.

Table 12.5-1

RADIATION ZONE DESIGNATIONS

Zone No.	Zone Description	Dose Rate mrem/h
1	Unrestricted and uncontrolled occupancy	<0.5
2	Frequent occupational access	0.5 to 2.5
3	Infrequent occupational access	2.5 to 5
4	Controlled radiation (frequent access)	5 to 20
5	Controlled radiation area (infrequent access)	20 to 100
6	Locked high radiation area (restricted access)	>100



KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 7 & 8
FSAR

ACCESS CONTROL BUILDING
PLAN AT EL. 74'-0" & 100'-0"

Figure 12.5-1