CHAPTER 12 - RADIATION PROTECTION

TABLE OF CONTENTS

110

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

CHAPTER 12 - RADIATION PROTECTION

LIST OF TABLES

110

CHAPTER 12 - RADIOACTION PROTECTION

LIST OF FIGURES

CHAPTER 12 - RADIATION PROTECTION

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

This chapter describes the radiation protection measures of the station design and the operating policies to ensure that internal and external radiation exposure to station personnel, contractors, and the general population due to station operation, 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 that are sources of significant ionizing radiation; use of remotely operated equipment and remote-operating attachments; ventilation of areas that contain equipment that has the potential for creating airborne radiation; 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

It is the policy of the management of Korea Hydro & Nuclear Power Company (KHNP) 446
to maintain occupational radiation exposure to personnel as low as is reasonably achievable (ALARA), consistent with plans for maintenance requirements and expected operational requirements, while satisfying the applicable regulations. This ALARA policy is applied to total man-Sv 446
accumulated by personnel and to individual exposures. The development of the proper attitude and awareness of potential problems related to health physics is accomplished by proper training of all plant personnel.

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 and documentation. These reviews were conducted and documented consistent with Korean Atomic Energy Act (AEA).

The plant design was reviewed, updated, and modified as necessary during the design and construction phases. Engineers reviewed the plant design and integrated the layout, shielding, ventilation, and monitoring designs with traffic control, security, access control, maintenance, inservice inspection, and radiation protection aspects to ensure that the overall design produced a plant which will achieve exposures that are ALARA.

Piping containing radioactive fluids was routed as part of the engineering design effort. This ensured that lines expected to contain significant radiation sources are adequately shielded and properly routed to minimize exposure to personnel. Onsite inspections are also conducted, as necessary during construction, to ensure that the shielding and piping layout meets established criteria. During construction, visual inspections were made to ensure that there were no major defects in the shield walls as they were placed. During initial power operations, radiation surveys will be conducted to ensure that the shielding meets design requirements 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 that operator exposures are ALARA.

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 was reviewed by qualified engineers as a normal part of the plant architect-engineer design activities.

The Architect/Engineer personnel responsible for ALARA design and review are plant architect-engineers competent in the application of radiation protection principles, including radiological dose assessment, shielding design, and radwaste systems design. The project radiation protection personnel are selected from a Radiation Management Staff group and are given the tasks of assisting the design effort and conducting reviews of plant design. In addition, the central Radiation Management Staff group reviews operating plant exposures and designs to establish criteria and guidelines for radiation protection. This group reviews various aspects of the plant design to ascertain conformance with the established criteria of maintaining exposures ALARA and recommends design modifications as necessary. During construction, field changes submitted by KEPCO to the plant architect-engineer were reviewed by their nuclear engineers to ensure that occupational exposures would be ALARA. The reviews, performed by competent nuclear engineering personnel under the supervision of the plant architect-engineer chief engineers, meet the intent of Regulatory Guide 8.8.

The NSSS vendor is committed to ensuring that occupational radiation exposures are ALARA in pressurized water reactors by providing systems and components whose designs take into consideration the radiation exposures associated with operation, inspection, and maintenance. In keeping with this commitment, the NSSS vendor has defined the responsibilities of the radiation protection groups and have provided an environment in which the radiation protection groups can properly perform their duties. These policies and commitments are reflected in the organizational charters and written procedures for the groups involved in the radiation protection. The design considerations of the NSSS vendor for keeping exposure ALARA are discussed in Subsection 12.1.2.1.3.

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 YGN 1&2 Health Physics Program is in conformance with the recommendations contained in Regulatory Guides 8.8 and 8.10. This program shall be expanded to include YGN 3&4, and it will incorporate the modern design philosophy of these stations. Korean Atomic Energy Act (AEA) and related regulations, provide the regulatory framework under which the ALARA philosophy is implemented.

The plant manager shall be responsible for the radiation protection program for YGN 3&4. The program consists of written health physics procedures, intensive training of radiation protection fundamentals, and periodic reviews by plant management.

The manager, Radiation Management 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, Radiation Management Section, and exercises supervisory control over the health physics technicians. The health physics technicians are responsible for radiation surveys, contamination surveys, and air sampling. Section 13.1 presents the Organization of the Planned Radiation Management Section for YGN 3&4.

Plant personnel shall be trained or indoctrinated in radiation protection by the manager, Radiation Management Section. The radiation protection functions of the Health Physics group include the following:

a. Prepare and approve detailed procedures for radiation protection before plant operation to ensure that dose limits are established in accordance with the approved radiation protection program, and that plant personnel or visitors' exposure can be maintained ALARA.

b. Control 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. Determine requirements for and extent of the use of personnel monitoring devices and maintaining records of personnel exposure.

d. Control and account for all radioactive material entering or leaving the plant site.

e. Establish procedures for dealing with potential or actual emergency conditions.

f. Train the plant staff and visitors in radiation protection policy and procedures, as required.

g. Include a dosimetry program as part of the site health physics program. This program includes whole body counting to measure the uptake of radioactive material by personnel.

h. Provide for periodic calibration for process radiation, area radiation, portable radiation, and airborne radioactivity monitoring instrumentation in the station design.

i. Monitor and maintain records of all significant radioactive effluent pathways from the station.

12.1.2 <u>Design Considerations</u>

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

YGN 3&4 radiation protection design considerations establish a practical basis for maintaining radiation exposures ALARA. The direction is established by a set of radiation protection design goals. Conservatively set criteria in facility and equipment design, experience from past designs and operating plants incorporated to improve the present design, and mechanisms established for design review were implemented to fulfill the ALARA requirement. (Radiation protection design features which are provided to maintain personnel radiation exposures ALARA are described in Section 12.3.)

12.1.2.1 General Design Considerations for ALARA Exposure

General design considerations and radiation protection design goals employed to maintain in-plant radiation exposures ALARA in accordance with Regulatory Guide 8.8 have three objectives:

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

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

c. Minimizing maintenance exposure utilizing equipment selection, location, system design, and structure design.

The above design consideration was given the proper direction by applying the following sequence of radiation protection design goals:

a. Establish design dose rates for general access areas based on KEPCO's experience and Korean AEA and related regulation.

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

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

d. Determine shielding required to maintain design dose rates.

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

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

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

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

a. Considerations related to minimizing personnel time spent in radiation areas.

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

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

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

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

5. Pumps with flanged connections are provided to allow quick removal and installation.

6. Provisions, where practicable, are provided to remotely or mechanically operate, repair, service, monitor, or inspect equipment.

7. Redundancy of equipment or components are provided to reduce the need for immediate repair thus allowing radioactive decay to reduce radiation levels.

b. Considerations related to equipment selection and design that reduce occupational radiation exposure.

1. Plug valves that require less maintenance in place of diaphragm valves.

2. Diaphragm seal valves that require no packing.

3. Longer-lived graphite-filled packing, instead of standard packing.

4. Fluid connections for the capability to backflush.

5. Remote systems (or connections) for remote chemical cleaning where practicable.

6. Air connections to tanks containing spargers to allow air injection to uncake contaminates.

7. Crossties between redundant equipment and/or related equipment capable of redundant operation to allow removal of contaminated equipment from service.

8. Mechanical seal flushing lines on pumps to reduce the accumulation of radioactive material in the seals.

9. Remote filter handling equipment for radwaste disposal.

10. Drains on tanks flush with inside surface of the tanks.

c. Considerations directed towards minimizing radiation levels to personnel working on radioactive equipment.

1. Frequently operated valves of radioactive systems are segregated from large components where practicable.

2. Valves that cannot be segregated (especially those that have to be located in high radiation areas) are remotely operated to the extent practicable. The radiation zone and operation frequency are considered when selecting remote valve operators.

3. The following design provisions are incorporated in order to minimize radioactive system leakage:

   a) All valves larger than 2 inches (5 cm) are provided with double packing and lantern ring leakoffs.

b) All piping high-point vents and low-point drains in high-pressure systems (900-pound (408-kg) rating and higher) are provided with double valves.

4. 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 to facilitate radwaste processing.

5. Filters are supplied with the means to perform cartridge replacement with remote tools.

6. Demineralizers are designed to remotely remove spent resins hydraulically and replace with new resins from a remote location.

7. Equipment, piping, connections, and valves are designed to minimize the buildup of radioactive material and facilitate flushing of crud traps.

8. Direct drain connections are provided for minimizing the spread of contamination into equipment service areas.

9. Heat exchangers are 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.

10. 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.

11. Equipment that processes fluids with low radioactivity is located in separate cubicles from equipment that processes highly radioactive fluids.

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

12.1.2.1.2 Facility Layout General Design Considerations

YGN 3&4 radiation protection design considerations are categorized into several radiation protection concerns, which are described in the following paragraphs. Station layout considers direct radiation (for this section, direct radiation is defined as scattered and unscattered gamma and/or neutron rays from nonairborne radiation source(s)), and ventilation design considers airborne radioactivity. Health physics and access control are concerned with both direct and airborne radioactivity. Control of radioactive fluids and effluents are concerned with the processing and detection of radioactive materials. Specific ALARA design features are discussed in Subsection 12.3.1.1.

a. The following facility design considerations are for maintaining radiation exposure to station personnel ALARA:

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

   2. The radiation levels in general access areas are kept ALARA.

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

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

5. Segregation of radiation zones are employed whenever practicable.

6. Shielding accommodates equipment removal and maintenance. Localized shielding or space adequate for localized shielding are incorporated into the shielding design when an exposure savings will result.

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

8. The radiation protection design is based upon the design criteria given in Section 12.3.

b. The following Health Physics considerations encompass the safety of station personnel and the general public as it relates to radiation exposure:

1. The NSSC Notice 2014-34(방사선방호 등에 관한 기준) limits are maintained for operating personnel and the general public. 735

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

3. Radiation protection design objectives related to 10 CFR 100 are given in Chapter 15.

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

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

6. Ventilation systems are designed to direct potentially airborne radioactive material from occupied areas through potential radiation areas and to the appropriate vent exhaust.

7. Radioactive effluent release paths to the environment are monitored for radioactive content.

8. Facilities for analysis of radioactive samples are furnished.

9. Access to radioactive equipment is designed so that, with properly trained personnel, radiation exposures during all modes of station operation will meet the ALARA requirement.

10. Radioactive fluids (liquids and gases) are contained and controlled to keep the release of radioactive materials to general access areas and the environment ALARA.

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

12.1.2.1.3 NSSS Design Considerations

Experience from past designs and operating reactors has been employed in establishing radiation protection design guidelines. A program of data acquisition and retrieval has been employed to establish equipment and system design bases. The engineering effort was directed toward characterizing the mechanisms of radiation level buildup, evaluating the performance of plant systems in mitigating radioactive buildup, and establishing the role of operating procedures in reducing radiation level buildup. Data and experience gained from this effort have been directly applied to all disciplines in the design and development of equipment employed in the YGN 3&4 NSSS.

Systems and equipment employed in the YGN 3 & 4 NSSS are designed with the objective of reducing the need for maintenance within radiation areas. Whenever possible, components requiring frequent maintenance are located in low radiation zones or flanged to facilitate removal to a low radiation zone. Whenever possible, materials have been selected to withstand a 40-year service life, thus minimizing the need for replacement and reducing maintenance frequencies. Controls are remotely mounted in low radiation zones. Equipments such as heat exchangers and valves have been designed for ease of access during maintenance. Equipment is environmentally qualified to meet its performance requirements under the environmental and operating conditions in which it is required to function. An objective in equipment design is to keep the occupational radiation exposures ALARA by ensuring that operators spend a minimum amount of time in a radiation environment. 110

Experience from past NSSS designs and inservice inspection programs have enabled design features to be incorporated into the YGN 3 & 4 NSSS that reduce occupational radiation exposure. The most significant feature for performing inservice inspection is the minimization of linear feet of weld in the major components. The minimization of weld footage was accomplished by component redesign, use of forged sections versus forged-welded plate sections, and increasing the size of certain sections.

Systems and equipments employed in the YGN 3 & 4 NSSS were designed with the objective of ensuring that occupational exposure due to decommissioning procedures will be ALARA. Decommissioning can be facilitated through design features which minimize the buildup of in-plant radiation and contamination. Decommissioning can be accomplished through the application of one of several available alternative methods, e.g., mothballing, entombment, immediate or delayed dismantling, etc. The experience gained in the continued application of these methods, and any developing variations, will further minimize occupational radiation exposures.

The YGN 3 & 4 NSSS incorporates many of the design features recommended in Regulatory Guide 8.8, in addition to other specific designs and established guidelines to keep in-plant exposures ALARA. The following design features are specifically effective in reducing in-plant exposures during decommissioning.

a. Components containing radioactive material, such as primary coolant, resin and concentrates, are provided with connections for flushing with water or decontamination chemicals.
b. Equipment is designed to minimize crud buildup and facilitate decontamination.
c. Spaces are provided where appropriate to place shielding for the purpose of reducing neutron activation.
d. Activated corrosion product buildup over the life of the NSSS is minimized in the design stage through appropriate selection of corrosion-resistant materials, specification of an appropriate chemistry control program and limitation applied to the cobalt (Co) and antimony (Sb) content of materials exposed to the primary coolant.

Radiation protection guidelines are employed to meet the intent of Regulatory Guide 8.8. These guidelines were provided to design engineers in each discipline to ensure occupational radiation exposures are maintained ALARA. Radiation protection design reviews were performed based upon these guidelines pursuant to the guidance provided in Regulatory Guide 8.8. The general design objectives for systems and equipment are to reduce exposure to operating personnel ALARA to meet the intent of Regulatory Guide 8.8, and to operate within the limits of radiation protection in restricted areas given in Korean AEA, related regulations and 10 CFR 50. Subsection 12.3.1.2 lists some of the design features that are incorporated into the NSSS design.

Radiation protection design review is an ongoing activity throughout all phases of design and construction. Design guidelines are prepared to maintain occupational radiation exposure ALARA. Radiation control engineers ensure that guidelines are provided to engineers working in other disciplines involved in the NSSS design. Since all disciplines involved in the NSSS design are covered by the guidelines, every discipline is involved in the review. Process radiation control engineers work with engineers and designers in other disciplines to ensure that all radiation protection considerations are taken into account. These engineers provide advice on the most desirable design option for radiation protection when alternate designs are possible in satisfying process requirements. 110

Radiation protection design reviews, based on established guidelines, took place prior to the release of design drawings, system design requirements, or component design requirements. Comments from the engineers performing the review are transmitted by appropriate documentation to the appropriate engineers for resolution. Follow-up reviews are conducted as necessary to ensure resolution within the established radiation protection guidelines.

12.1.3 Operational Considerations

Radiological health and safety procedures will be developed and continually reviewed to ensure 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 have influenced 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, Radiation Management 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 Korean Atomic Energy Act (AEA) & Enforcement Regulations.

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

b. Systems and equipment that are subject to crud buildup, such as the chemical and volume control system; shutdown cooling system; liquid radwaste system; and various pumps, filters and demineralizers; have been equipped with connections that can be used for flushing the system to eliminate potential hot-spot buildup.

   Before performing maintenance work, flushing and/or chemically decontaminating the system or piece of equipment will be considered to reduce the crud levels and, hence, personnel exposure.

c. Work involving whole body exposure rates in excess of 1 mSv/hr or removable contamination levels in excess of 166 Bq/cm^2 will be preplanned so the job can be performed safely with a minimum of personnel exposure. 446

d. On complex jobs or jobs with exceptionally high radiation levels, dry-run training, and in some cases mockups, 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 esposure received. These special tools will be used 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 containment, e.g., glove bags, poly bottles, and tents, is 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 with 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-Sv exposure estimates for the job. At the 446
completion of the work, a debriefing session will be held 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-Sv expended, will be compiled and filed 446
for future reference. All radiation aspects, i.e., radiation, contamination, airborne radioactivity, 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 Subsection 12.1.3.1 are generally used when inspecting and plugging steam generator tubes. A multi-stud tensioner will be used to quickly open and close the steam generator manway, and consequently, reduce the work time in the radiation area.

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 that 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

The techniques described in Subsection 12.1.3.1 are normally used when removing and installing the reactor head. Quick disconnect electrical cables and reactor head ventilation ducts that can be quickly removed are used to reduce time spent in radiation areas. A multi-stud tensioner will be used to quickly remove and install the reactor vessel head. Temporary shielding can be installed around the outer control rod drive mechanisms to reduce exposure if crud collects in the control rod drive housings. A temporary shield may be provided in the refueling cavity, if necessary, for personnel shielding.

Communications are provided up to the refueling floor so that personnel can communicate with supervisors, thus reducing lost time and exposure to workers and supervisors.

12.1.3.4 Specific ALARA Considerations for Inservice Inspection

The techniques in Subsection 12.1.3.1 are normally used when performing inservice inspections. Remote testing devices will be 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. "An Engineering Evaluation of Nuclear Power Plant Decommissioning Alternatives," Atomic Industrial Forum, AIF/NESP-009, November 1976.

6. "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor," Battelle Pacific Northwest Laboratory, NUREG/CR-0130, June 1978.

12.2 RADIATION SOURCES

This section discusses and identifies the sources of radiation that form the basis for shield design calculations for the design of personnel protective measures and for dose assessment.

12.2.1 Contained Sources

The shielding design source terms are based on full power operation with 0.25% fuel cladding defects and no gas stripping. Sources in the primary coolant include fission products released from fuel cladding defects, activation products, and corrosion products. Throughout most of the reactor coolant system, activation products, principally nitrogen-16 (N-16), are the primary radiation sources for shielding design during power operation. 110

The design sources are presented in this subsection by system.

12.2.1.1 Containment

12.2.1.1.1 Reactor Core

The primary radiation emanating from the reactor core during normal operation are neutrons and gamma rays. Tables 12.2-1 and 12.2-2 list neutron and gamma fluxes in the reactor cavity at the side of the reactor vessel. These tables are consistent with the parameters discussed in Section 4.3. Table 12.2-3 lists gamma fluxes outside the reactor vessel after shutdown for shielding requirements during shutdown and inservice inspection.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the reactor coolant system are fission products released from the fuel, and activation and corrosion products that are circulated in the reactor coolant. These sources are listed in Table 12.2-4, and their bases are shown in Table 12.2-5. Reactor coolant fission product sources are obtained via equation 11.1-1 and 11.1-2.

Tables 11.1-14 and 11.1-16 list maximum expected activities due to crud deposits on steam generator tubing and primary system piping.

The activation product nitrogen-16 (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 on the total transit time to the component. The derivation of N-16 activity is shown in Subsection 11.1.5. Table 12.2-6 lists N-16 activities at various locations in the primary coolant loop. 110

12.2.1.1.3 Main Steam System

The radionuclide concentrations in the steam generator liquid and in the main steam are calculated using the maximum reactor coolant activities with 0.25% fuel cladding defects, a primary to secondary leak rate of 1 gpm (3.8 L/min), and the assumptions given in Table 11.2-4. The results are listed in Table 12.2-14. For further discussion of this system, refer to Subsection 11.1.1.3. In addition, design basis radioactivities for each component of the steam generator blowdown system are given in Table 12.2-15.

12.2.1.1.4 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant long-term source of radiation in the containment after plant shutdown for refueling. A reactor operating time necessary to establish near-equilibrium fission product buildup for the reactor at rated power is used in determining the source strengths. The initial fuel composition that produced the maximum decay source is used. The spent fuel decay gamma source is given in Table 12.2-7.

12.2.1.1.5 Processing Systems

12.2.1.1.5.1 Chemical and Volume Control System (CVCS) 110

Radiation sources in the CVCS consist of those radioisotopes carried in the reactor coolant, discussed in subsection 12.2.1.1.2. N-16 is the predominant radiation source in the regenerative heat exchanger. The design of the letdown system ensures that most of the N-16 has decayed before the letdown stream leaves the containment.

The shielding design is based on the full power operation with 0.25% fuel cladding defects and no gas stripping. The sources in the regenerative heat exchanger and reactor drain tank are given in Tables 12.2-8 through 12.2-9 and Table 12.2-12, respectively. Their design parameters are discussed in Subsection 12.2.1.2.1.

12.2.1.2 Auxiliary Building

12.2.1.2.1 Chemical and Volume Control System

The major CVCS subsystems include the letdown purification subsystem, the boron recovery subsystem, the seal injection subsystem, and the reactor drain and equipment drain processing subsystem.

The major CVCS equipment items include the regenerative and letdown heat exchangers, purification filter, purification mixed bed and deborating anion demineralizers, volume control tank, and charging pumps for the letdown purification subsystem. The boron recovery subsystem includes the boric acid concentrator package and the boric acid condensate ion exchanger. The seal injection subsystem for the reactor coolant pumps includes the seal injection filter and the seal injection heat exchanger. The reactor drain and equipment drain processing subsystem includes the reactor drain filter, the preholdup mixed bed demineralizer, and the gas stripper. 110

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

The mixed bed demineralizers retain the cation and anion fission product activity and the corrosion product metals (i.e., crud). Each of the two purification mixed bed demineralizers is sized to pass the maximum letdown flow. One of these demineralizers is used intermittently to control the reactor coolant lithium concentration.

The deborating anion demineralizer is used near the end of core life when low boron concentrations are present. This method is used because the normal feed and bleed method for boron control becomes less efficient at low boron concentrations.

Boron concentration in the reactor coolant is normally reduced by the feed and bleed method which consists of adding demineralized water to the volume control tank. When the water level in the volume control tank exceeds the high level setpoint, the letdown stream is automatically diverted to the reclamation subsystem.

Boron concentration in the reactor coolant is normally increased by pumping boric acid solution from the refueling water tank into the volume control tank.

The seal injection heat exchanger heats the water going to the reactor coolant pump seals. 110

The demineralizers and filters could concentrate significant amounts of radioactive materials. These items are, therefore, of main concern when determining radiation shielding requirements.

All CVCS major components listed below, except for the regenerative heat exchanger and reactor drain tank, the holdup tank and reactor makeup water tank, and the refueling water tank, are located in the auxiliary building. (The regenerative heat exchanger and reactor drain tank are located in the containment; the holdup tank and reactor makeup water tank are located in the yard, and the refueling water tank is located in the fuel building.) The radiation sources in the major CVCS components are listed in Tables 12.2-8 through 12.2-12. The design parameters of the major CVCS components are discussed below.

a. Heat Exchangers (Tables 12.2-8 and 12.2-9)

1. Regenerative Heat Exchanger

Letdown side volume is based on 6.7 gallons (25.40 L) of water with conservative data for reactor coolant specific activity. Charging side volume is based on 46.2 gallons (174.9 L) of water with volume control tank specific activity.

2. Letdown Heat Exchanger

Total tube volume is based on 76.9 gallons (290.4 L) of water with conservative data for reactor coolant specific activity.

110

3. Seal Injection Heat Exchanger

Total tube volume is based on 2.4 gallons (9.16 L) of water with volume control tank specific activity.

b. Ion Exchangers (Table 12.2-10)

1. Purification Ion Exchanger

Total curie inventory is based on resin buildup of 571 EFPDs. This ion exchanger is used for lithium removal and normal purification of RCS letdown. When it is used for lithium removal, it is on line for an average of 95 days prior to placing it in service as a purification ion exchanger for 476 EFPDs.

Anions have a decontamination factor (DF) of 100 and efficiency of 99%. Crud has a DF of 50 and efficiency of 98%.

All nuclides except Xe, Kr, Rb, Sb, Y and Cs have a DF of 50 and efficiency of 98%. Sb, Y, Xe and Kr have a DF of 1.0 and efficiency of 0%. Rb and Cs have a DF of 2.0 and efficiency of 50%.

2. Preholdup Ion Exchanger

Total curie inventory is based on resin buildup of 476 EFPDs. All nuclides except Xe, Kr, Sb, Y, Rb, and Cs have a DF of 10 and efficiency of 90%. Rb and Cs have a DF of 100 and efficiency of 99%. Xe, Sb, Y, and Kr have a DF of 1.0 and efficiency of 0%. 110

Sources processed by the pre-holdup ion exchanger include: 3,767 gpd (14,260 L/day) of letdown previously processed through the purification ion exchanger (14,260 L/day) and purification filter, 200 gpd (757 L/day) from the reactor drain tank (RDT), and 50 gpd (189 L/day) from the equipment drain tank (EDT).

3. Boric Acid Condensate Ion Exchanger

Total curie inventory is based on resin buildup of 476 EFPDs. An anion DF of 10 and an efficiency of 90% were used. All other ions have a DF of 1.0 and an efficiency of 0%. Crud has a DF of 1.0 and an efficiency of 0%. The processed liquid is based on a design flowrate of 20 gpm (75.7 L/min).

4. Deborating Ion Exchanger.

Total curie inventory is based on resin buildup of 476 EFPDs. An anion DF of 10 and an efficiency of 90% were used. All other ions have a DF of 1.0 and efficiency of 0%. Crud has a DF of 10 and an efficiency of 90%. The liquid processed is based on a design flowrate of 150 gpm (567.8 L/min).

c. Filters (Table 12.2-11)

Total curie inventories on all CVCS filters are based on crud buildup of 476 EFPDs. All CVCS filters remove crud with a DF of 10 and an efficiency of 90%. 110

d. Tanks (Table 12.2-12)

1. Reactor Drain Tank
The total curie inventory in the RDT is based on an expected maximum water volume of 2,138 gallons (8,092 L) and an expected maximum vapor volume of 183 ft^3 (5,182 L). The tank vapor-liquid phases are in equilibrium and the tank liquid activity fraction is 1.0 of conservative data for the RCS.

2. Equipment Drain Tank

The total curie inventory in the EDT is based on an expected maximum water volume of 4,935 gallons (18,679 L) and an expected maximum vapor volume of 913 ft^3 (25,853 L). The tank vapor-liquid phases are in equilibrium and the tank liquid activity fraction is 0.1 of conservative data for the RCS.

3. Volume Control Tank (VCT)

The total curie inventory in the VCT is based on an expected maximum water volume of 3,196 gallons (12,097 L) and an expected maximum vapor volume of 447 ft^3 (12,658 L). No gas stripping was considered and the VCT vapor gas is in equilibrium with the liquid.

4. Holdup Tank (HT)

The total curie inventory in the HT is based on an expected maximum water volume of 91,200 gallons (3.452×10^5 L). No gas stripping was considered and the tank vapor-liquid phases are assumed not in equilibrium. Activity in the tank is based on holdup of 200 gallons per day (gpd) (757 L/day) from the RDT, 50 gpd (189 L/day) from the EDT, and 3,767 gpd (14,260 L/day) from RCS letdown. 110

5. Reactor Makeup Water Tank (RMWT)

The total curie inventory in the RMWT is based on an expected maximum water volume of 513,000 gallons (1.942×10^6 L). Activity in the tank is based on 1.06×10^6 gallons (4.012×10^6 L) processed by the boric acid concentrator.

6. Refueling Water Tank (RWT)

The total curie inventory in the RWT is based on an expected maximum water volume of 809,100 gallons (3.062×10^6 L). Activity in the tank is based on 1.2×10^5 gallons (4.542×10^5 L) processed by the boric acid concentrator.

12.2.1.2.2 Shutdown Cooling System

In the reactor shutdown condition, the significant sources requiring shielding consideration are the spent fuel, the shutdown cooling system (SCS), and the incore detector system. Individual components may require shielding during shutdown due to deposited crud material. Deposited crud activities are discussed in Subsection 11.1.3.

The radiation sources in the RCS and the CVCS are 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 phenomenon and crud bursts can result in increased radiation sources. The spiking phenomenon 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 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: 110

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.

The shutdown cooling system removes decay heat from the reactor following a shutdown. A description of the shutdown cooling system is provided in subsection 5.4.7. The shutdown cooling system will be placed in operation approximately 4 hours after reactor shutdown. The radioactive sources in the shutdown cooling system are provided in Table 12.2-13.

12.2.1.2.3 Sampling Systems

Radiological shielding for sampling systems and piping is based on the design basis sources for the radioactive system associated with the sampling system. (Considerations are given to the conditions under which sampling occurs and the effect these conditions have on the design basis system sources). Refer to Subsection 9.3.2 for a discussion of the sampling systems. 110

12.2.1.2.4 Control Room Boundary

There are no radioactive sources within the control room boundary. 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 HVAC filters used post-LOCA. The source terms are discussed in Sections 6.4 and 15.6.

12.2.1.3 Fuel Building

The predominant radioactivity sources in the spent fuel storage and transfer areas in the fuel building are the spent fuel assemblies, which are discussed in Subsection 12.2.1.1.4. The spent fuel decay gamma source to be used in shielding design is given in Table 12.2-7.

The design basis source of radioactivity in the spent fuel pool cooling and cleanup system is based on radioactive sources contained within the refueling pool water during refueling operations. Radionuclides in the refueling pool water are removed using one train of the spent fuel pool cleanup system.
The 0.25% failed fuel value is used in shielding design. Spent fuel pool and refueling pool maximum activities are given in Table 12.2-16. Radionuclide activity inventories on the spent fuel pool cleanup system filters and demineralizers are given in Table 12.2-17.

12.2.1.4 Turbine Building

Potential radioactivity in the main steam supply and power conversion systems is a result of steam generator tube leakage. Radioactive sources in the primary coolant (e.g., due to fuel cladding defects and activation) enter the secondary system via the leaking tubes. This radioactivity is sufficiently low that normally no radiation shielding for steam or condensate lines is required. The condensate polishing demineralizer activity is given in Table 12.2-15, and the radionuclidic crud concentrations in the high capacity flash tank are discussed in Subsection 11.1.1.3. 110

12.2.1.5 Radwaste Building

12.2.1.5.1 Liquid and Solid Radwaste Systems

Radioactive sources in the radwaste system include fission and activation radionuclides produced in the core and in the reactor coolant. Radwaste system components contain varying degrees of radioactivity. The level of radioactivity depends on the components and operating parameters of the particular radwaste system under consideration. Radionuclide concentrations in the radwaste systems were determined using the DIJESTER computer code (Reference 2). With this code, the accumulation of and decay of radioactivity in radwaste fluid flow systems can be modeled.

Design basis source term for radiation shielding of each component of the radwaste system is based on the maximum activities and assumptions given in Table 12.2-4 and Table 11.2-9, respectively. The calculation results are shown in Table 12.2-18.

12.2.1.5.2 Gaseous Radwaste System

Radiation sources for each component of the gaseous radwaste system are calculated, based on the assumption of 0.25% fuel cladding defects with no gas stripping. Maximum radioactivities on charcoal delay beds are provided in Table 12.2-19.

12.2.1.6 Stored Radioactivity

The principal sources of radioactivity not stored inside plant buildings are the holdup tank, reactor makeup water tank, and the condensate water storage tank. 110
Radiation shielding is provided in such a manner that the dose rate at the surface of these tanks or tank structures does not exceed 0.5 mrem/hr. The sources in the holdup tank and reactor makeup water tank are discussed in Subsection 12.2.1.2.1.

No other radioactive wastes are stored outside the plant buildings. All the spent fuel assemblies will be stored in the spent fuel pool until they are 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. Radioactive wastes stored inside plant structures are shielded such that Zone 1 access is allowed outside the structure.

12.2.2 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.

12.2.2.1 Recirculation Loop Sources

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

The reactor is assumed to have operated continuously at the power level stated above for 3 cycles before the LOCA.

110

The sources for which system radiation levels are calculated are based on a maximum credible accident, which assumes that 100% of the noble gases, 50% of the halogens, and 1% 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 3, 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.

12.2.2.2 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 HVAC filters used post-LOCA. The source terms are shown on the tables in Sections 6.4 and 15.6, and the calculation model is described in Appendix 15C. 110

12.2.2.3 Other Sources

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

12.2.3 Airborne Radioactive Material Sources

With the exception of noble gases, sources of airborne radioactivity are generated from radioactive liquid sources by the mechanisms discussed in the following subsection. The generation of airborne radioactivity in radiation areas can affect the areas normally accessible to operating personnel, mainly pump and valve areas. The airborne radioactivity during normal operation for accessible areas is discussed in Subsection 12.2.3.1.2. The calculational model is given in Subsection 12.2.3.1.3.

In addition to affecting the plant, the airborne radioactive material which is exhausted from the plant via the ventilation systems enters the environment.

12.2.3.1 Production of Radioactive Airborne Material

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

Component	Airborne Method
fuel pool	evaporation
radwaste	evaporation (venting)
high-energy line leak	vapor, evaporation
spray from high-energy line	vapor, droplets, evaporation
moderate-energy line leak	droplets, evaporation
spill	droplets, evaporation

110

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

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

12.2.3.1.1 Sources in Areas Normally Accessible to Operating Personnel

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

The ventilation flow path is from areas of potentially low airborne radioactivity to areas of potentially higher airborne radioactivity. The ventilation system was designed to control the airborne radioactivity in the laboratories, maintenance areas, and the refueling floor of the reactor building. Based on past experience, airborne halogens (particularly iodine) represent the largest fraction of the maximum permissible concentration (MPC). Halogens are therefore the most significant airborne radioactive isotopes with respect to radiological concerns.

Maintenance generally accounts for a sizeable portion of the internal exposure of personnel because plant personnel have to perform many maintenance functions in areas with relatively high airborne radioactivity. The airborne radioactivity is caused by leaks, spills, venting, etc. The airborne concentrations are calculated for the occurrences that are the most common, namely, leaks and venting.

12.2.3.1.2 Calculated Concentrations During Operation

The calculated concentrations of airborne radioactive nuclides in cubicles are based upon the model given in Subsection 12.2.3.1.3. Airborne radioactive noble gas, tritium, and iodine concentrations as a fraction of the MPC are determined for the containment free air volume, for normally accessible cubicles within the auxiliary building and radwaste building, and for general access areas of the power block buildings. Except under the conditions noted in Subsection 12.2.3.1.1, general access areas normally have very little, if any, airborne contaminants (i.e., $<10^{-12}$ $\mu Ci/cm^3$ above background) during normal operation.

12.2.3.1.3 Models and Parameters Used in Calculations of Airborne Radioactivity Concentration

Equilibrium airborne radioactivity concentrations in rooms, cubicles, etc., during normal operation are calculated based on the following equation:

$$C_A = \frac{L\ C_L\ P}{K\ (\lambda\ V + F)} \qquad (12.2\text{-}1)$$

where:

			U.S. Units	Metric Units
C_A	=	airborne concentration in each cubicle	($\mu Ci/cm^3$)	($\mu Ci/cm^3$)
L	=	liquid leak rate	(gpm)	(cm^3/min)
C_L	=	liquid concentration	($\mu Ci/cm^3$)	($\mu Ci/cm^3$)
P	=	fraction of activity released to air	-	-
K	=	conversion factor (7.48)	(gal/ft^3)	1.0
λ	=	decay constant	(min^{-1})	(min^{-1})
V	=	enclosed volume	(ft^3)	(cm^3)
F	=	air exhaust flow rate	(ft^3/min)	(cm^3/min)

Equation 12.2-1 is the equilibrium solution of a basic differential equation that represents the situation in which there is no airborne radioactivity in the ventilation airflow(s) entering the area under consideration. To accommodate the situation in which there exists airborne radioactivity in one or more ventilation airflows entering the area of concern, additional term(s) are added to the basic differential equation. The modified differential equation is then solved to obtain the equilibrium concentration within the specified area. The calculated concentrations are shown on Table 12.2-20.

12.2.3.1.4 Vent Effluents 110

All ventilation system exhausts (except for office space and service building HVAC) are routed to the appropriate plant vent discharge. The ventilation systems, containing potentially high airborne radioactivity concentrations are provided with filters specifically designed to hold up or remove radioactive material (Section 9.4).

The dominating radioisotopes released through the plant vent discharges are the noble gases from the radwaste gas system and the vent filter system. The expected yearly releases during normal operation are discussed in Section 11.3.

12.2.4 References

1. Lutz, R. J. and Chubb, W., "Iodine Spiking - Cause and Effect," ANS Transactions, Vol. 28, pg. 649, June 1978.

2. "DIJESTER, A Program to Compute Radioactive Decay in Fluid Flow Systems," S&L Program No. 9.8.060-1.0, D. J. Pichurski, April 1976.

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission November 1980.

TABLE 12.2-1

TYPICAL NEUTRON SPECTRA OUTSIDE REACTOR VESSEL

Average Neutron Energy (MeV)	Neutron Spectra (neutrons/cm^2-sec)
13.60	1.72 (+6)
11.10	6.56 (+6)
9.10	1.52 (+7)
7.27	2.99 (+7)
5.66	4.97 (+7)
4.51	4.11 (+7)
3.53	7.16 (+7)
2.73	9.05 (+7)
2.40	3.01 (+7)
2.09	1.82 (+8)
1.47	6.68 (+8)
8.30×10^{-1}	2.20 (+9)
3.30×10^{-1}	6.38 (+9)
5.70×10^{-2}	4.38 (+9)
1.96×10^{-3}	1.37 (+9)
3.42×10^{-4}	1.26 (+9)
6.50×10^{-5}	8.83 (+8)
1.98×10^{-5}	5.66 (+8)
6.90×10^{-6}	6.20 (+8)
2.09×10^{-7}	4.44 (+8)
7.60×10^{-7}	3.94 (+8)
2.50×10^{-8} (thermal)	2.90 (+9)

110

NOTE : At core midplane, one half-foot (15.24 cm) from vessel surface.

TABLE 12.2-2

TYPICAL GAMMA SPECTRA OUTSIDE REACTOR VESSEL

Average Gamma Energy (MeV)	Gamma Spectra (Gamma/cm^2-sec)
9.00	8.70 (+7)
7.25	4.76 (+8)
5.75	4.30 (+8)
4.50	3.69 (+8)
3.50	5.03 (+8)
2.75	3.33 (+8)
2.25	7.71 (+8)
1.83	4.85 (+8)
1.50	4.51 (+8)
1.16	5.62 (+8)
0.90	4.06 (+8)
0.70	5.21 (+8)
0.50	1.10 (+9)
0.35	9.70 (+8)
0.25	1.59 (+9)
0.15	2.54 (+9)
0.075	4.00 (+8)
0.025	2.33 (+6)

110

NOTE : At core midplane, one-half foot (15.24 cm) from vessel surface.

TABLE 12.2-3

TYPICAL SHUTDOWN GAMMA SPECTRA OUTSIDE REACTOR VESSEL

Average Gamma Energy (MeV)	Decay Gamma ($Gammas/cm^2$-sec)	Material Activation ($Gammas/cm^2$-sec)
4.50	-	3.40 (+2)
3.50	1.57 (+3)	3.71 (+1)
2.75	4.85 (+4)	6.99 (+5)
2.25	8.40 (+4)	8.73 (+4)
1.83	1.45 (+5)	6.44 (+4)
1.50	1.98 (+5)	6.06 (+5)
1.16	2.52 (+5)	5.69 (+5)
0.90	1.92 (+5)	6.75 (+5)
0.70	2.54 (+5)	5.22 (+5)
0.50	3.71 (+5)	7.64 (+5)
0.35	3.56 (+5)	6.58 (+5)
0.25	5.70 (+5)	1.16 (+6)
0.15	9.05 (+5)	1.63 (+6)
0.075	1.35 (+5)	2.27 (+5)
0.025	5.32 (+2)	8.76 (+2)

110

NOTE : At core midplane, one-half foot (15.24 cm) from vessel surface, and 48 hours after shutdown.

TABLE 12.2-4 (sh. 1 of 2)

MAXIMUM ACTIVITIES IN THE REACTOR COOLANT DUE TO CONTINUOUS OPERATION AT MAXIMUM POWER WITH 0.25% FAILED FUEL AND NO GAS STRIPPING

110

TABLE 12.2-4 (sh. 2 of 2)

110

TABLE 12.2-5

BASIS FOR ANALYSIS OF REACTOR COOLANT ACTIVITIES

110

TABLE 12.2-6

YGN 3&4 N-16 ACTIVITY

110

TABLE 12.2-7

SPENT FUEL GAMMA SOURCE

110

TABLE 12.2-8 (Sh. 1 of 2)

CVCS HEAT EXCHANGER INVENTORIES
Maximum Values
(Curies)
(0.25% Failed Fuel, No Gas Stripping, 2872 MWt Core Power)

NUCLIDE	LETDOWN	REGENERATIVE	Seal INJECTION
N-16	6.7E-01	1.5E+00	.0E+00
KR-85M	8.7E-02	5.9E-02	2.7E-03
KR-85	3.5E-01	2.4E-01	1.1E-02
KR-87	6.7E-02	4.3E-02	1.9E-03
KR-88	1.9E-01	1.2E-01	5.6E-03
XE-131M	3.8E-01	2.6E-01	1.2E-02
XE-133M	2.2E-02	1.5E-02	7.0E-04
XE-133	2.4E+01	1.6E+01	7.4E-01
XE-135M	4.9E-02	2.8E-02	1.2E-03
XE-135	4.7E-01	3.2E-01	1.4E-02
XE-137	1.1E-02	5.2E-03	2.2E-04
XE-138	4.1E-02	2.2E-02	9.8E-04
BR-84	1.6E-03	1.5E-04	4.2E-07
RB-88	1.9E-01	6.2E-02	2.4E-03
SR-89	2.0E-04	2.0E-05	1.3E-07
SR-90	1.0E-05	1.0E-06	6.4E-09
SR-91	3.5E-04	3.5E-05	2.2E-07
Y-91M	2.0E-04	1.2E-04	5.5E-06
Y-91	2.9E-05	2.0E-05	8.9E-07
Y-93	8.2E-06	5.6E-06	2.5E-07
ZR-95	9.5E-05	9.4E-06	5.9E-08
NB-95	3.2E-05	3.2E-06	2.0E-08
TC-99M	9.3E-03	9.2E-04	5.7E-06
MO-99	1.8E-02	1.8E-03	1.1E-05
RU-103	1.1E-05	1.1E-06	6.7E-09
RU-106	4.4E-06	4.3E-07	2.7E-09

110

TABLE 12.2-8 (Sh. 2 of 2)

NUCLIDE	LETDOWN	REGENERATIVE	Seal INJECTION
TE-129M	3.8E-04	3.5E-05	1.2E-07
TE-129	4.4E-04	4.0E-05	1.3E-07
I-131	1.5E-01	1.4E-02	4.8E-05
TE-131M	1.9E-03	1.7E-04	5.8E-07
TE-131	8.2E-04	7.5E-05	2.1E-07
TE-132	1.3E-02	1.2E-03	3.9E-06
I-132	4.9E-02	4.6E-03	1.5E-05
I-133	2.4E-01	2.2E-02	7.4E-05
I-134	3.2E-02	3.0E-03	9.0E-06
CS-134	1.7E-02	6.8E-03	2.7E-04
I-135	1.5E-01	1.4E-02	4.5E-05
CS-136	2.9E-03	1.1E-03	4.5E-05
CS-137	2.2E-02	8.6E-03	3.5E-04
BA-140	2.5E-04	2.5E-05	1.6E-07
LA-140	7.3E-05	7.2E-06	4.5E-08
CE-141	9.3E-06	9.2E-07	5.8E-09
CE-143	2.8E-05	2.7E-06	1.7E-08
CE-144	2.5E-05	2.5E-06	1.6E-08
CR-51	3.0E-03	2.6E-04	1.6E-08
MN-54	3.9E-04	3.4E-05	2.1E-09
FE-59	7.3E-05	6.5E-06	4.0E-10
CO-58	1.8E-03	1.6E-04	9.8E-09
CO-60	2.0E-04	1.8E-05	1.1E-09
BA-137M	4.3E-02	1.1E-02	3.6E-04

110

TABLE 12.2-9

CVCS HEAT EXCHANGER ACTIVITY

Maximum Values

(Gamma/sec)

(0.25% Failed Fuel, No Gas Stripping, 2872 MWt Core Power)

Energy Group	Letdown			Regenerative			Seal Injection		
MeV	SOLUBLE	CRUD	TOTAL	SOLUBLE	CRUD	TOTAL	SOLUBLE	CRUD	TOTAL
0.25	3.1E+10	8.4E+09	3.9E+10	1.7E+10	3.4E+10	5.1E+10	7.4E+08	2.5E+08	9.9E+08
0.50	1.6E+10	1.7E+10	3.3E+10	3.8E+09	7.2E+10	7.6E+10	1 3E+08	5.1E+08	6.4E+08
0.75	1.0E+10	5.9E+10	6.9E+10	2.1E+09	2.5E+11	2.5E+11	6.3E+07	1.8E+09	1.9E+09
1.00	3.9E+09	4.4E+08	4.3E+09	7.5E+08	1.8E+09	2.6E+09	2.1E+07	1.3E+07	3.4E+07
1.38	4.4E+09	1.3E+10	1.7E+10	6.3E+08	5.3E+10	5.4E+!0	1.3E+07	3.9E+08	4.0E+08
2.00	9.9E+09	3.4E+08	1.0E+10	4.9E+09	1.4E+09	6.3E+09	2.1E+08	1.0E+07	2.2E+07
3.00	7.0E+08	0.0E+00	7.0E+09	3.5E+08	0.0E+00	3.5E+08	1.5E+07	0.0E+00	1.5E+07
4.00	4.3E+07	0.0E+00	4.3E+07	1.3E+07	0.0E+00	1.3E+07	4.6E+05	0.0E+00	4.6E+05
6.00	-	1.7E+10	1.7E+10	-	3.8E+10	3.8E+10	-	0.0E+00	0.0E+00

110

TABLE 12.2-10 (Sh. 1 of 2)

CVCS ION EXCHANGER INVENTORIES
Maximum Values
(Curies)

(0.25% Failed Fuel, No Gas Stripping, 2872 MWt Core Power)

NUCLIDE	PURIFICATION	DEBORATING	PREHOLDUP	BORIC ACID CONDENSATE
N-16	2.0E-03	.0E+00	.0E+00	.0E+00
KR-85M	3.8E-01	3.8E-01	3.6E-01	4.3E-09
KR-85	1.5E+00	1.5E+00	1.5E+00	8.8E-07
KR-87	2.9E-01	2.9E-01	2.7E-01	9.5E-10
KR-88	8.2E-01	8.2E-01	7.6E-01	5.8E-09
XE-131M	1.7E+00	1.7E+00	1.6E+00	5.1E-07
XE-133M	9.9E-02	9.9E-02	9.3E-02	1.2E-08
XE-133	1.1E+02	1.1E+02	1.0E+02	1.8E-05
XE-135M	2.2E-01	2.2E-01	2.0E-01	1.5E-10
XE-135	2.0E+00	2.0E+00	1.9E+00	4.7E-08
XE-137	5.0E-02	5.0E-02	4.6E-02	8.1E-12
XE-138	1.8E-01	1.8E-01	1.7E-01	1.1E-10
BR-84	7.0E-02	6.3E-04	4.8E-05	7.7E-12
RB-88	2.8E+00	4.2E-01	4.4E-02	2.9E-09
SR-89	2.0E+01	1.8E-05	3.9E-02	6.1E-09
SR-90	6.6E+00	9.0E-07	1.4E-02	3.8E-10
SR-91	2.8E-01	3.1E-05	2.6E-04	1.9E-10
Y-91M	8.6E-04	8.6E-04	8.0E-04	2.0E-09
Y-91	1.3E-04	1.3E-04	1.2E-04	9.6E-08
Y-93	3.6E-05	3.6E-05	3.3E-05	1.0E-09
ZR-95	1.2E+01	8.4E-06	2.4E-02	3.0E-09
NB-95	2.2E+00	2.8E-06	4.2E-03	8.8E-10
TC-99M	4.7E+00	8.2E-04	4.2E-03	3.1E-09
MO-99	1.0E+02	1.6E-03	1.3E-01	8.1E-08
RU-103	8.5E-01	9.5E-07	1.6E-03	3.1E-10
RU-106	1.9E+00	3.8E-07	3.9E-03	1.6E-10

110

TABLE 12.2-10 (Sh. 2 of 2)

NUCLIDE	PURIFICATION	DEBORATING	PREHOLDUP	BORIC ACID CONDENSATE
TE-129M	2.6E+01	4.1E-02	4.4E-02	3.2E-06
TE-129	4.1E-02	3.8E-04	2.9E-05	9.9E-12
I-131	2.5E+03	1.3E+01	3.5E+00	1.4E-04
TE-131M	4.7E+00	4.2E-02	4.3E-03	3.6E-08
TE-131	2.8E-02	2.6E-04	2.0E-05	2.5E-12
TE-132	8.2E+01	6.4E-01	9.3E-02	1.9E-06
I-132	9.5E+00	8.6E-02	6.7E-03	4.6E-09
I-133	4.2E+02	3.8E+00	3.5E-01	2.1E-06
I-134	2.3E+00	2.1E-02	1.6E-03	4.3E-10
CS-134	5.6E+03	3.8E-02	1.0E+02	3.1E-07
I-135	8.2E+01	7.4E-01	6.1E-02	1.2E-07
CS-136	4.5E+01	6.4E-03	7.8E-01	2.9E-08
CS-137	8.6E+03	4.9E-02	1.5E+02	3.9E-07
BA-140	6.4E+00	2.2E-05	1.1E-02	4.5E-09
LA-140	2.4E-01	6.4E-06	2.8E-04	1.9E-10
CE-141	6.0E-01	8.2E-07	1.1E-03	2.5E-10
CE-143	7.6E-02	2.4E-06	8.3E-05	5.7E-11
CE-144	9.7E+00	2.2E-06	2.0E-02	9.0E-10
CR-51	1.6E+01	6.3E-02	3.0E-02	7.6E-09
MN-54	1.6E+01	9.2E-03	3.2E-02	1.4E-09
FE-59	6.6E-01	1.6E-03	1.3E-03	2.2E-10
CO-58	2.6E+01	4.1E-02	5.0E-02	5.8E-09
CO-60	1.2E+01	4.8E-03	2.5E-02	7.5E-10
BA-137M	8.6E+03	5.1E-02	1.5E+02	3.9E-07

110

TABLE 12.2-11 (Sh. 1 of 2)

CVCS FILTER INVENTORIES
Maximum Values
(Curies)
(0.25% Failed Fuel, No Gas Stripping, 2872 MWt Core Power)

Nuclide	SEAL INJECTION	REACTOR DRAIN	BORIC ACID	RCS LETDOWN PURIFICATION	REACTOR MAKEUP WATER
N-16	.0E+00	.0E+00	.0E+00	8.5E-04	.0E+00
KR-85M	1.2E-03	1.2E-03	.0E+00	4.5E-03	1.3E-13
KR-85	5.0E-03	1.5E-02	1.5E-08	1.8E-02	2.6E-09
KR-87	8.9E-04	8.9E-04	.0E+00	3.5E-03	8.6E-15
KR-88	2.6E-03	2.5E-03	.0E+00	9.7E-03	1.2E-13
XE-131M	5.4E-03	1.3E-02	1.3E-09	2.0E-02	9.2E-10
XE-133M	3.2E-04	4.9E-04	7.5E-14	1.2E-03	4.7E-12
XE-133	3.4E-01	6.6E-01	5.7E-09	1.2E+00	1.6E-08
XE-135M	5.5E-04	6.5E-04	.0E+00	2.6E-03	.0E+00
XE-135	6.6E-03	7.0E-03	.0E+00	2.4E-02	3.1E-12
XE-137	1.0E-04	1.5E-04	.0E+00	5.9E-04	.0E+00
XE-138	4.5E-04	5.3E-04	.0E+00	2.1E-03	.0E+00
BR-84	1.9E-07	2.1E-05	.0E+00	8.2E-05	.0E+00
RB-88	1.1E-03	2.5E-03	.0E+00	9.8E-03	6.2E-15
SR-89	5.7E-08	8.0E-06	1.5E-08	1.0E-05	2.9E-11
SR-90	2.9E-09	4.3E-07	7.2E-09	5.3E-07	4.6E-12
SR-91	9.9E-08	5.3E-06	.0E+00	1.8E-05	1.3E-14
Y-91M	2.5E-06	2.6E-06	.0E+00	1.0E-05	1.2E-14
Y-91	4.1E-07	1.1E-06	2.8E-07	1.5E-06	5.0E-10
Y-93	1.2E-07	1.2E-07	.0E+00	4.2E-07	7.3E-14
ZR-95	2.7E-08	3.8E-06	9.9E-09	5.0E-06	1.6E-11
NB-95	9.2E-09	1.2E-06	1.3E-09	1.7E-06	3.4E-12
TC-99M	2.6E-06	1.3E-04	.0E+00	4.8E-04	1.3E-13
MO-99	5.2E-06	4.2E-04	1.7E-10	9.4E-04	3.8E-11
RU-103	3.1E-09	4.2E-07	5.5E-10	5.6E-07	1.3E-12

110

TABLE 12.2-11 (Sh. 2 of 2)

Nuclide	SEAL INJECTION	REACTOR DRAIN	BORIC ACID	RCS LETDOWN PURIFICATION	REACTOR MAKEUP WATER
RU-106	1.2E-09	1.8E-07	2.0E-09	2.3E-07	1.6E-12
TE-129M	5.4E-08	1.5E-05	1.4E-08	2.0E-05	3.6E-12
TE-129	5.8E-08	5.8E-06	.0E+00	2.3E-05	.0E+00
I-131	2.2E-05	4.7E-03	1.9E-07	8.0E-03	2.3E-10
TE-131M	2.7E-07	3.5E-05	1.2E-14	9.7E-05	6.3E-14
TE-131	9.8E-08	1.1E-05	.0E+00	4.2E-05	.0E+00
TE-132	1.8E-06	3.0E-04	2.7E-10	6.5E-04	3.2E-12
I-132	6.8E-06	6.7E-04	.0E+00	2.6E-03	8.0E-15
I-133	3.4E-05	4.1E-03	8.9E-15	1.2E-02	3.7E-12
I-134	4.1E-06	4.2E-04	.0E+00	1.7E-03	.0E+00
CS-134	1.2E-04	7.4E-04	4.7E-06	9.1E-04	3.4E-09
I-135	2.1E-05	2.1E-03	.0E+00	7.6E-03	2.1E-13
CS-136	2.1E-05	9.9E-05	9.1E-09	1.5E-04	5.6E-11
CS-137	1.6E-04	9.4E-04	7.4E-06	1.2E-03	4.8E-09
BA-140	7.2E-08	8.4E-06	1.4E-09	1.3E-05	8.6E-12
LA-140	2.1E-08	1.5E-06	1.3E-14	3.8E-06	5.3E-14
CE-141	2.7E-09	3.6E-07	3.5E-10	4.8E-07	9.2E-13
CE-143	7.9E-09	5.3E-07	.0E+00	1.4E-06	1.3E-14
CE-144	7.2E-09	1.1E-06	1.0E-08	1.3E-06	8.5E-12
CR-51	9.2E-02	2.5E-01	2.0E-05	1.5E+02	5.3E-07
MN-54	8.8E-02	2.7E-01	2.8E-04	1.5E+02	2.1E-06
FE-59	3.7E-03	1.1E-02	1.7E-06	6.0E+00	3.2E-08
CO-58	1.4E-01	4.2E-01	1.2E-04	2.3E+02	1.7E-06
CO-60	6.7E-02	2.1E-01	3.3E-04	1.1E+02	2.0E-06
BA-137M	1.6E-04	1.2E-03	7.4E-06	2.3E-03	4.8E-09

110

TABLE 12.2-12 (Sh. 1 of 2)

CVCS TANK INVENTORIES

110

TABLE 12.2-12 (Sh. 2 of 2)

110

TABLE 12.2-13

SHUTDOWN COOLING SYSTEM (SCS) SPECIFIC SOURCE STRENGTHS

Maximum Values

(MeV/gram-sec)

Decay Time (hr)	Energy (MeV) 0.3	0.63	1.10	1.55	1.99	2.38	2.75	3.25	3.70
1	3.3(+4)*	2.4(+5)	6.7(+4)	1.9(+4)	4.7(+3)	3.4(+2)	1.6(+2)	9.9(+1)	1.2(+2)
10	2.5(+4)	1.2(+5)	2.9(+4)	7.5(+3)	2.2(+3)	2.9(+1)	6.7(-1)	6.2(-1)	8.9(-3)
100	1.8(+4)	4.4(+4)	6.3(+3)	2.4(+3)	3.5(+2)	2.2(+1)	2.7(-2)	8.7(-3)	-

110

* Numbers in parentheses denote powers of ten.

TABLE 12.2-14(sh. 1 of 2)

DESIGN BASIS(0.25% FAILED FUEL) RADIONUCLIDE CONCENTRATION IN THE SECONDARY SYSTEM

(μCi/gm)

110

TABLE 12.2-14(sh. 2 of 2)

110

110

TABLE 12.2-15(sh. 1 of 2)

SHIELDING DESIGN BASIS ACTIVITIES FOR SECONDARY SYSTEM COMPONENTS

110

TABLE 12.2-15(sh. 2 of 2)

TABLE 12.2-16 (Sh. 1 of 2)

FISSION AND CORROSION PRODUCT ACTIVITIES IN THE SPENT FUEL AND REFUELING POOLS BASED ON 0.25% FAILED FUEL

110

TABLE 12.2-16 (Sh. 2 of 2)

110

TABLE 12.2-17 (Sh. 1 of 2)

SFPCCS SHIELDING DESIGNE BASIS (0.25% FAILED FUEL)
ACTIVITIES (Ci)

110

TABLE 12.2-17 (Sh. 2 of 2)

SFPCCS SHIELDING DESIGNE BASIS (0.25% FAILED FUEL)
ACTIVITIES (Ci)

110

TABLE 12.2-18 (Sheet 1 of 5)

LIQUID RADWASTE SYSTEM SOURCE TERMS

TABLE 12.2-18 (sh. 2 of 5)

A. Liquid Radwaste System Process Stream Concentrations ($\mu Ci/cm^3$)*

110

TABLE 12.2-18 (sh. 3 of 5)

A. Liquid Radwaste System Process Stream Concentrations ($\mu Ci/cm^3$)*

110

TABLE 12.2-18 (sh. 4 of 5)

B. Liquid Radwaste System Inventories (Curies)*

110

TABLE 12.2-18 (sh. 5 of 5)

B. Liquid Radwaste System Inventories (Curies)*

TABLE 12.2-19

RADIONUCLIDE INVENTORIES IN GASEOUS
RASWASTE SYSTEM COMPONENTS
CHARCOAL DELAY BEDS (ACTIVITIES IN CURIES)

110

TABLE 12.2-20 ... 1 of 3)

AIRBORNE RADIOACTIVE CONCENTRATIONS

Primary Auxiliary Building

ELEVATION	AREA	EXHAUST AIR FLOW RATE[(1)]	AIRBORNE CONCENTRATIONS[(2)] Kr, Xe	AIRBORNE CONCENTRATIONS[(2)] I	AIRBORNE CONCENTRATIONS[(2)] H-3	FRACTION OF MPC[(3)] Kr, Xe	FRACTION OF MPC[(3)] I	FRACTION OF MPC[(3)] H-3
	Charging pump room	1050	1.89(-06)[(4)]	2.97(-11)	6.17(-10)	2.10(-01)	1.30(-03)	6.17(-05)
	SC HX. valve room	1750	1.02(-08)	2.50(-11)	6.21(-11)	1.14(-03)	1.08(-03)	6.21(-06)
	SC HX. room	750	2.13(-08)	5.20(-11)	2.44(-10)	2.37(-03)	2.25(-03)	2.44(-05)
	SI recirculation area	1050	1.83(-08)	4.46(-11)	1.11(-10)	2.03(-03)	1.93(-03)	1.11(-05)
	Pipe tunnel	1000	3.97(-08)	2.61(-15)	2.41(-10)	4.41(-03)	1.13(-07)	2.41(-05)
	Encapsulation tank room	1050	3.05(-09)	7.43(-12)	1.85(-11)	3.39(-04)	3.22(-04)	1.85(-06)

(1) CFM

(2) μCi/cc

(3) Fraction of MPC = $\sum_i \frac{(\text{Airborne Concentration})_i}{\text{MPC}_i}$

(4) Numbers in parentheses denote powers of ten.

TABLE 12.2-20 (Sh. 2 of 3)

AIRBORNE RADIOACTIVE CONCENTRATIONS

Secondry Auxiliary Building

ELEVATION	AREA	EXHAUST AIR FLOW RATE	AIRBORNE CONCENTRATIONS Kr, Xe	AIRBORNE CONCENTRATIONS I	AIRBORNE CONCENTRATIONS H-3	FRACTION OF MPC Kr Xe	FRACTION OF MPC I	FRACTION OF MPC H-3
	Spent resin pump room	1200	1.07(-05)	1.01(-10)	5.58(-09)	1.19(+00)	4.39(-03)	5.58(-04)
	Process radiation monitor room	700	6.64(-08)	9.02(-12)	1.69(-11)	7.38(-03)	3 91(-04)	1.69(-06)
	Reactor drain pump room	1300	4.48(-06)	3.86(-09)	3.01(-09)	4.51(-01)	2.20(-01)	3.01(-04)
	EDT room	1500	4.71(-08)	1.09(-10)	1.45(-09)	4.75(-03)	5.35(-03)	1.45(-04)
	SG lay-up wet recirc. pump room	1500	9.92(-13)	1.81(-13)	4.52(-11)	1.10(-07)	8.62(-06)	4.52(-06)
	Valve room	1000	1.75(-06)	4.27(-09)	1.06(-08)	1.95(-01)	1.85(-01)	1.06(-03)
	Valve room	1050	2.47(-05)	1.40(-11)	8.08(-10)	2.87(+00)	6.11(-04)	8.08(-05)
	Boric acid concent. room	2000	2.31(-10)	8.17(-13)	4.01(-10)	2.30(-05)	7.86(-05)	4.01(-05)
	Letdown valve room	2000	2.16(-06)	5.27(-09)	2.11(-08)	2.40(-01)	2.29(-01)	2.11(-03)

TABLE 12.2-20 (Sh. 3 of 3)

AIRBORNE RADIOACTIVE CONCENTRATIONS

Radwaste Building

ELEVATION	AREA	EXHAUST AIR FLOW RATE	AIRBORNE CONCENTRATIONS Kr, Xe	AIRBORNE CONCENTRATIONS I	AIRBORNE CONCENTRATIONS H-3	FRACTION OF MPC Kr Xe	FRACTION OF MPC I	FRACTION OF MPC H-3
	GRS header drain tank room	600	1.45(-06)	4.87(-10)	2.35(-08)	2.98(-01)	2.11(-02)	2.35(-03)
	LRS demineral. feed pump room	550	4.25(-07)	1.04(-11)	4.86(-10)	4.72(-02)	4.49(-04)	4.86(-05)
	Low act. spent resin sluice pump	1000	1.37(-07)	9.66(-09)	0.00(+00)	1.55(-02)	9.21(-01)	0.00(+00)
	Sludge decon. pump room	550	1.23(-06)	3.00(-11)	1.41(-09)	1.37(-01)	1.30(-03)	1.41(-04)
	Evaporation feed pump room	550	8.28(-07)	2.02(-11)	9.48(-10)	9.21(-02)	8.75(-04)	9.48(-05)
	Recycle release pump room	500	1.26(-06)	3.07(-11)	1.44(-09)	1.40(-01)	1.33(-03)	1.44(-04)
	Chemical waste pump room	500	6.77(-07)	1.65(-11)	7.75(-10)	7.53(-02)	7.16(-04)	7.75(-05)
	GRS inlet skid room	550	2.32(-06)	7.77(-10)	3.75(-08)	4.74(-01)	3.36(-02)	3.75(-03)
	Valve room	750	4.05(-07)	9.87(-12)	4.63(-10)	4.50(-02)	4.28(-04)	4.63(-05)
	Valve room	600	6.52(-07)	3.43(-09)	2.19(-09)	7.25(-02)	1.49(-01)	2.19(-04)

12.3 RADIATION PROTECTION DESIGN FEATURES

Radiation protection design features were provided to reduce direct radiation, control airborne radioactivity, identify radiation areas, decontaminate personnel and equipment, calibrate radiation monitors, and maintain personnel radiation exposure as low as reasonably achievable (ALARA) during plant operations including normal operation, anticipated operational occurrences, and design basis accident.

12.3.1 Facility Design Features

In this subsection, specific design features for maintaining personnel exposures ALARA are discussed. The radiation protection guidance given in Paragraph C.3 of Regulatory Guide 8.8 was 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 are of a low activity level and do not require special handling equipment. Unsealed sources and radioactive samples are handled in conventional hoods, which are part of the laboratory HVAC system, described in Section 9.4. 110

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.

12.3.1.1 Balance-Of-Plant (BOP) Equipment and Component Designs for ALARA

This subsection describes the radiological 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 subsections.

12.3.1.1.1 Filters

All filters in the plant are cartridge type. All filters except ventilation filters and GRS filters, which accumulate significant quantities of radioactivity, are designed for cartridge replacement with semiremote and/or remote handling devices. Adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste area.

Liquid systems containing radioactive filters except the low-activity blowdown filters are provided with a filter handling system for the removal of spent radioactive filter cartridges from their housings and for their transfer to the radwaste building for packaging and shipment from the site for disposal. When in use, the handling system is placed over the filter in the space normally occupied by its concrete hatch. The cartridge can then be lifted into a shielded cask. The radioactive filtering unit that contains the cartridge being removed has shielding walls separating it from other high-level radiation sources. Therefore, the operator will not be exposed to unattenuated radiation.

A specially designed vehicle powered by battery is used to transport casks, each containing a cartridge, from the power block to the radwaste building. The process is accomplished so 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. The removable hatches are provided with radiation probe access holes for radiation surveying.

12.3.1.1.2 Demineralizers

Demineralizers for highly radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks before solidification and so that fresh resin can be remotely loaded into the demineralizer. The demineralizers and piping are designed with provisions for being flushed with demineralized water. Strainers are installed in the downstream lines to prevent spent resin from being carried out. The removable hatches are provided with radiation probe access holes for radiation surveying. Each demineralizer is shielded from surrounding high-level radiation sources so that maintenance exposure will be ALARA.

12.3.1.1.3 Evaporators

Evaporators are provided with chemical addition connections to allow the use of chemicals for descaling operations. Large evaporator components are mounted on skids and separately installed. Provisions 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. Most of the valves in radioactive lines are located on or are operable from the accessible side of the shield wall. Alternatively, the lines containing the valves are flushable before access for valve lineup.

12.3.1.1.4 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 that allows easy removal, whenever it is practical. All pumps in highly radioactive systems are provided with flanged connections to permit easy removal when permitted by all other design considerations. Pump casings and bedplates are provided with draining capabilities.

12.3.1.1.5 Tanks

Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to control any contamination within plant structures. The overflow for tanks located outside structures, and which contain significant quantities of radioactivity, is directed to the liquid radwaste system. All tanks have high-level alarms to alert the operator before an overflow occurrence.

12.3.1.1.6 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.

12.3.1.1.7 Instruments

The following ALARA design considerations are itemized for clarity:

a. Output devices such as instrument readouts, pressure switches, electrical bistable devices, electric converters, and control devices, are installed and located so as to minimize plant personnel exposure to radiation.

b. Instrumentation displays and controls are installed in the lowest practicable radiation zone. Instrument readouts are located and positioned in areas that will result in the lowest personnel exposures, if such areas are consistent with other requirements such as instrument accuracy and precision.

c. Instrument readouts are designed, located, and positioned to minimize the time and exposure required to take a reading. The followings are considered in locating and positioning instrument readout devices to ensure ALARA exposures:

1. Locate in readily accessible areas.

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

3. Face readout in a direction convenient for reading.

4. Provide easily readable indicator numbers and easily observable indicator/dial pointers and needles.

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

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

d. In locating instruments and instrument readouts, the possibility of local hot spots due to streaming radiation or from the accumulation of radioactivity in lines, ducts, filters, and equipment is considered.

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

f. Diaphragm seals are provided on instrument sensing lines for process piping that may contain highly radioactive solids to reduce the servicing time required to free the lines of solids. Instrument and sensing line connections are located so as to avoid corrosion product and radioactive gas buildup.

12.3.1.1.8 Valves

To the extent practicable, all valves servicing radioactive or potentially radioactive equipment are located in shielded valve aisles, apart from the (adjacent) equipment being serviced. Walk-in valve aisles are used where practicable. Additional ALARA considerations related to valves are the following:

a. Long runs of radioactive pipe to and from valves located in valve aisles are minimized to reduce the amount of radioactive material in valve aisles.

b. All radioactive or potentially radioactive manually operated valves (and associated piping) are shielded from the valve operating area, to the extent practicable.

   1. When equipment in a Zone 6 area (defined in Table 12.3-1) 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.

   2. To the maximum practicable extent, simple straight reach rods are 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.

   3. 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 are not expected to enter the valve gallery during spent resin or cartridge transfer operations. The valve gallery shield walls are designed to minimize personnel exposure during maintenance within or adjacent to the gallery and to protect personnel who remotely operate the valves.

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

d. To the extent practicable, all motor-operated valves and pneumatic-operated valves (air-operated valves) that are in radioactive or potentially radioactive service are located in areas shielded from the adjacent components or items of equipment that the valves serve. Locating these valves (which typically have higher maintenance requirement than manually operated valves) in shielded areas minimizes potential radiation exposures to personnel during valve maintenance and inservice inspection.

e. For valves located in radiation areas, provisions are made to drain adjacent radioactive components. For valve maintenance, provisions are made for draining and/or flushing the valve and associated connecting lines of radioactive fluids so that radiation exposures can be minimized. Wherever practical, spaces are provided for temporary shielding.

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

g. 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 that have demonstrated satisfactory performance in nuclear plant operation are used. 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.

12.3.1.1.9 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. ALARA considerations related to pipe routing are discussed in greater detail in Subsection 12.3.1.3.

12.3.1.1.10 Equipment Draining and Flushing

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

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

To the extent practicable, drain piping is of welded construction and welded in a manner (e.g., using consumable inserts) to minimize crevices that might collect radioactive material. Use of backing rings in the welds or use of socket welds may be acceptable if they are for 2-inch (5 cm) and under piping that does not handle radioactive sludge.

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

The design of the radwaste filters are checked to ensure that the filters can be drained and flushed before filter element replacement.

Draining capability is provided

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

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

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

Where practicable, flushing of the radwaste tank interior is accomplished by an installed sparger (where justified) or by providing a recirculation line to the bottom of the tank for the pump servicing the tank so that spraying can be used to suspend settled deposits. For manual flushing, adequate capability is provided by water connections near the tank cubicles.

12.3.1.1.11 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. Normally, incandescent lights, which require less time for servicing, are provided, and hence, the personnel exposure is reduced. Fluorescent lights, which are used in some areas, do not require frequent service due to the increased life of the tubes.

12.3.1.1.12 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 from the housings. The design features are described in detail in Subsection 12.3.3. ALARA considerations for duct routing are discussed in Subsection 12.3.1.3.2.

12.3.1.1.13 Hydrogen Recombiners

The locations of the hydrogen recombiners and hydrogen recombiner control panels were selected to ensure adequate shielding from highly radioactive components. Hydrogen recombiner areas and hydrogen recombiner control panel areas require irregular access, not continuous occupancy, during a post-LOCA condition. Post-LOCA access is available to bring the hydrogen recombiners into the plant and to the locations for hydrogen recombiner installation.

12.3.1.1.14 Sample Stations

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

12.3.1.1.15 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. Traversing radioactive cubicles are evaluated case by case, but generally when a shield wall is penetrated, the clean service terminates in that cubicle.

12.3.1.2 NSSS Component Design Considerations

Following are some of the specific design features used to ensure that occupational radiation exposure due to operations and maintenance of the YGN 3 & 4 NSSS will be ALARA. Demineralizers are addressed under the heading of ion exchangers. 110

a. Pumps

1. Most pumps and associated piping are flanged to facilitate ease of removal to a low radiation area for maintenance or repair. Pump internals can then be removed to a low radiation area for maintenance.

2. All pump casings are provided with drain connections to facilitate decontamination.

b. Ion Exchangers

1. Ion exchangers are designed for complete drainage.

2. Spent resin is designed to be removed remotely by hydraulically flushing the resin from the vessel to the solid radwaste system.

3. The fresh resin inlet is designed to extend into a low radiation area above the shielded compartment housing the ion exchanger.

4. Ion exchangers are designed with a minimum of crevices in order not to accumulate radioactive crud.

c. Liquid Filters

1. Filter housings are provided with vent connections and designed for complete drainage.

2. Filter housings are designed with a minimum of crevices in order not to accumulate radioactive crud.

3. Filter housings and cartridges are designed to permit remote removal of the filter elements.

d. Tanks

1. Tanks are designed to be isolated for maintenance and provisions are made for complete drainage.

2. Tanks are provided with at least one of the following means of cleaning the tank internals for decontamination purposes:

a) Ample space is provided to facilitate cleaning from the tank manway.

b) Internal spray nozzles are provided on potentially highly contaminated tanks for internal decontamination.

c) The ability to hydraulically backflush or drain inlet screens is provided (on tanks or vessels with these screens) to facilitate decontamination.

3. All tanks are vented to either the gas collection header or the gas surge header, which facilitates removal of potentially radioactive gases during maintenance.

4. Nonpressurized tanks are provided with overflows, routed to a floor drain or other suitable collection point, to avoid radioactive fluids spilling to the floor or ground. 110

5. Tanks are designed with a minimum of crevices in order not to accumulate radioactive crud.

e. <u>Package Units</u>

1. Each package unit is skid mounted with all motors and pumps located on the periphery of the skid for free access and for quick removal to a low radiation area for maintenance or repair.

2. Space is provided on the skid for placement of portable shielding.

3. All package components are provided with provisions for flushing, draining, and chemical cleaning.

4. Heat exchangers are readily accessible for maintenance.

5. Controls are remotely mounted and the package can be remotely monitored. As many control elements as possible are mounted remotely from the components.

6. Components are designed with a minimum of crevices in order not to accumulate radioactive crud.

7. Radioactive gas is collected and sent to the gaseous radwaste system.

f. Valves

1. Radiation-resistant seals, gaskets and elastomers are employed when practical to extend the design life and reduce maintenance requirements. 110

2. Power-operated valves in the primary system are provided with double packing, a lantern gland, and stem leakoffs to collect leakage and to direct radioactive fluid away from access areas. All valve packing glands have provisions to adjust packing compression to reduce leakage.

3. Valves are designed so that they may be repacked without removing the yoke or topworks.

4. Remotely operated valves are utilized where practical and necessary.

5. Valve-wetted parts are made of austenitic stainless steel or other corrosion-resistant material.

6. Low leakage valves with backseats are employed wherever possible. Packless diaphragm valves are employed in highly contaminated systems.

g. Piping

1. Radiation source terms are provided to ensure that field-routed piping carrying radioactive material is either routed in shielding pipe cases or within shielded cubicles.

2. Whenever possible, pipe runs are sloped to prevent accumulation of, and to assist in the removal of, radioactive corrosion deposits.

3. The number of elbows, tees, etc., is minimized to reduce corrosion deposits. Where elbows are required, they are a long radius type for minimization of deposits. Pipe dead legs are avoided if possible. 110

h. Heat Exchangers

1. Heat exchangers are designed to accommodate the requirements of inservice inspection and for ease of access during maintenance to reduce the time operators are required to spend in a radiation environment.

2. Materials are selected to minimize the need for replacement, and corrosion-resistant materials are employed to reduce the frequency of maintenance.

i. Material Selection

Material is selected as described below to reduce exposures by reducing maintenance frequencies and by providing less circulating crud as a source of exposure where maintenance is necessary.

1. Materials of construction for components containing radioactive materials are selected with consideration for potential release of activated corrosion products from these materials. Materials contacting primary reactor coolant prohibit cobalt (Co) and antimony (Sb) as applicable.

2. Radiation exposure levels are considered when selecting materials for 40 year service. 110

3. Material selection is made with consideration given to other fluid conditions which could lead to premature material failure.

4. Other material considerations are discussed in Subsection 5.2.3.

j. Reactor Vessel Head Vent

A vent nozzle and line are provided on the reactor vessel head. This design feature allows a reduction of exposure during the head removal process by minimizing the gases discharged directly to the containment atmosphere while the head is being removed.

k. Reactor Coolant System Leakage Control

Exposures from airborne radionuclides to personnel entering the containment are minimized by controlling the amount of reactor coolant leakage to the containment atmosphere. Examples of such controlled leakage follow:

1. Pressurizer safety valve leakage is directed to the reactor drain tank, as discussed in Subsection 5.2.2.

2. Valves larger than 2 inches (5.08 cm) in diameter are provided with a double-packed stem with an intermediate lantern ring with a leakoff connection to the reactor drain tank.

3. Instrumentation is provided to detect abnormal reactor coolant pump seal leakage. The reactor coolant pumps are equipped with two stages of seals plus a vapor or backup seal as described in Section 5.4. The vapor or backup seal prevents leakage to the containment atmosphere and allow sufficient pressure to be maintained to direct the controlled seal leakage to the volume control and reactor drain tanks. The vapor seal is designed to withstand full reactor coolant system pressure in the event of failure of any or all of the two primary seals. 110

l. Refueling Equipment

1. All spent fuel transfer and storage operations are designed to be conducted underwater to ensure adequate shielding and to limit the maximum continuous radiation levels in working areas.

2. The equipment is designed to prevent the fuel from being lifted above the minimum safe water depth, thereby limiting personnel exposures and avoiding fuel damage.

3. The equipment is designed to limit the possibility of inadvertent fuel drops which could cause fuel damage and personnel exposures.

4. The refueling equipment is designed to facilitates the transfer of new and spent fuel at the same time to reduce overall fuel handling time and, therefore, personnel exposures during refueling.

5. Underwater cameras are used to facilitate safe handling and visual control, thus minimizing errors and potential exposures.

6. Portable hydraulic cutters are provided to cut expended control element assemblies and in-core instrumentation leads. The cutters allow underwater handling of these items. 110

7. Equipment is provided to allow for the underwater determination of leaking fuel elements.

m. Inservice Inspection Equipment

Inspection of the reactor coolant pressure boundary is done with remote equipment to keep personnel exposures to a minimum.

n. Remote Instrumentation

All systems containing radioactive fluids are designed to be controlled remotely to the maximum extent practical. This allows personnel radiation exposures from the normal operation of these systems to be minimized.

o. Inservice Inspection of Reactor Vessel Nozzle Welds

Welds joining the reactor vessel nozzle to reactor coolant pipe are designed to permit inservice inspection to be accomplished from the

Inner Diameter of the reactor vessel. Automated equipment normally used for reactor vessel pressure boundary inspections can be utilized in this area.

If inservice inspection of this area is performed from the outside, the removable sections of insulation for the reactor vessel and reactor coolant piping are utilized to facilitate access. These removable sections shall be lightweight and shall be held in place mainly by quick actuation-type buckle fasteners. After the necessary panels are removed, remote equipment can be utilized to perform the required inspections.

12.3.1.3 Common Facility and Equipment Layout Designs for ALARA Exposures

This subsection 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 Subsections 12.3.1.1 and 12.3.1.2 include the features discussed in the following subsections. 110

12.3.1.3.1 Valve Galleries and Aisles

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 that will allow easy decontamination. Valves are discussed in greater detail in Subsection 12.3.1.1.8.

12.3.1.3.2 Ducting and Piping

All potentially radioactive process lines are evaluated to determine proper routing and shielding requirements, based on minimizing radiation exposures to plant operating and maintenance personnel. Each duct and pipe run is individually analyzed to determine its potential radioactive impact on the plant. Duct routing is reviewed to ensure that airflow is from areas of low potential airborne radiation contamination to areas of higher potential airborne radiation contamination. Ventilation duct penetrations are located, evaluated, and shielded to satisfy the penetration requirements of Subsection 12.3.2.5.5. Because piping usually handles much higher levels of radioactivity than ducts, pipe routing and construction need to be more precise. The following piping guidelines are incorporated, to the extent practicable, into the design:

a. Pipes carrying radioactive material are routed in shielded pipe tunnels and chases or in areas where the radiation field due to the pipe is consistent with the radiation zone for the area being crossed. The routing of radioactive lines in low radiation zones is avoided to the extent practicable.

b. Whenever practicable, valves and instruments are not placed in radioactive pipeways (chases), and the pipeways are only used to route radioactive lines from pipe tunnels and adjacent radiation areas to the intended equipment.

c. Penetration through a pipe tunnel shield wall is avoided to the extent practicable, unless it is for a line that utilizes the tunnel for line routing.

d. Radioactive and nonradioactive piping are separated, to the extent practical, to minimize personnel exposure. Potentially radioactive (or intermittently radioactive) piping is located in the appropriate zoned and restricted areas. Branch lines having little or no flow during normal operation are connected above the horizontal mid-plane of the main pipe.

e. If maintenance is required, provisions are made for isolating and draining radioactive piping and associated equipment. 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 have pipe slopes that will permit thorough draining. Piping runs are sloped to promote drainage and to minimize crud and particulate deposition.

f. To aid in preventing crud buildup in process piping, sharp bends, dead ends, and other obvious crud traps are minimized, especially in pipes that handle resin slurries and sludges. Frequent and severe piping direction changes are minimized. In general, socket welds and welds employing backing rings are minimized on pipe sizes that are greater than 2 inches (5 cm) in diameter. These welds contribute to radioactive crud accumulation; therefore, slurry and sludge piping is greater than 2 inches (5 cm).

g. Slightly radioactive lines are routed so as to minimize radiation exposure to plant operating and maintenance personnel. Slightly radioactive lines in low-radiation zones are, to the extent practicable, routed at a minimum elevation above the finished floor of 10 feet. To the extent practicable, slightly radioactive lines are routed neither near normally traveled passageways, nor near galleries or other elevated work areas. To the extent practicable, radioactive or potentially radioactive sample lines used for grab samples are routed so that grab samples can be taken in low-radiation areas.

h. Routing and shielding of all radioactive process pipe of larger diameter than 2 inch (5.1 cm) are performed in the design office. Field routing of smaller radioactive lines which are not safety category 1 is done by providing field engineering with criteria for routing these lines.

For field routing of radioactive pipe, the guidelines listed below are followed.

1. Piping is installed at as high an elevation as is practicable but, in no case, below 10 feet (3 m) from the finished floor level in general access areas, nonsource cubicles, and hallways.

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

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

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

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

12.3.1.3.3 Penetrations

To minimize radiation streaming through penetrations, large penetrations are specially located and designed 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 shields or grouting the area around the service conveyance with material(s) of adequate density to satisfy radiation protection design. Subsection 12.3.2.5.5 specifies requirements for penetrations passing through radiation shield walls.

12.3.1.3.4 Contamination Control

Equipment vents and drains from highly radioactive systems are piped directly to the collection system instead of allowing contaminated liquid 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 that intrinsically leak, the spread of contamination is prevented by monitoring, detection, or collection. Leakage from components in high- radiation zones can be monitored by local or building sump level indicators and alarms or by area and process radiation monitors.

System process instrumentation can provide an indication of system leakage. Additional information on design features that permit monitoring and control of leakage is presented in Subsections 5.2.5, 9.3.3, and 12.3.4.

a. Floor and Sink Drains

Adequate floor drainage is provided for each room or cubicle housing component that contains, or may contain, radioactive liquids. Floors are properly sloped to the floor drain to facilitate floor drainage and prevent water puddles. Smooth surface coatings on concrete floors are used to facilitate draining and improve decontamination.

All floor drains considered to be radioactive are directed to appropriate liquid radwaste storage tanks. Sumps are used as intermediate collection points. Such sumps and tanks are appropriately shielded or appropriately located within radiation areas. Shielding of radwaste drain piping is discussed in Subsection 12.3.2.

To the extent practicable, radiation area floor drains are segregated from nonradiation area floor drains to protect against backflow of radioactive liquids into nonradiation areas, if drainage is blocked or if a large radioactive spill occurs.

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

Ventilation for floor and sink drains is considered to minimize the spread of radioactive gases and vapors into nonradiation and low-radiation areas.

b. Venting of Equipment

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

Radioactive sumps are evaluated to determine whether these sumps should be vented to a high-radiation area, such as to within the cubicle the sump is located, if it is a high-radiation cubicle, or to a filtered vent header. Venting of radwaste sumps can be important to control the concentrations of radioactive contaminants normally released to the air from potentially contaminated water held in the sumps. For sumps in shielded cubicles that vent to the cubicle, cubicle ventilation rates are designed to ensure adequate control over expected airborne concentrations of radioiodine. If venting to other areas is required, the sump covers have air inleakage and have no special provisions for sealing so that the sump can maintain a slightly negative pressure with respect to the area in which the sump is located. A small amount of air in-leakage to the sump is desirable to maintain airflow through the vent line.

Direct venting to a ventilation filtering unit was evaluated for the following sumps:

1. Primary auxiliary building sump
2. Secondary auxiliary building sump
3. Fuel building normal sump
4. Fuel building decontamination sump
5. Radwaste building normal sump
6. Radwaste building decontamination sump

c. Barriers for Radioactive Fluids

In addition to the radiation contamination control design features discussed above, the following design features specifically related to decontamination are incorporated into the radiation protection design.

1. Curbs

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

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

2. <u>Protective Surface Coatings</u>

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

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

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

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

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

d. Radiation Monitoring

Area radiation monitors (ARMs) and process radiation monitors (PRMs) are used to aid in the detection and control of radioactive contamination. Monitoring locations are selected to achieve this goal.

12.3.1.3.5 Equipment Layout

In those systems where process equipment is a major radiation source (such as chemical and volume control, radwaste, and boron recycle), 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 lower radiation zones.

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

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

Labyrinth entranceway shields are provided for each compartment from which radiation could stream to access areas and exceed the radiation zone dose limit for those areas. For potentially high radiation components (such as filters and demineralizers), completely enclosed shielded compartments with hatch openings are provided to the extent practicable.

Figure 12.3-1 provides typical layout arrangements for valves, pumps, major components and panels.

Exposure from routine in-plant inspection is controlled by locating, wherever possible, inspection points in properly shielded low background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on adequate space and ease of motion in a properly shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is considered.

12.3.1.3.6 Station Support Facilities

a. Equipment Decontamination Facilities

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

The plant's main machine shop and equipment decontamination facility are located in the radwaste building and secondary auxiliary building, respectively. Their purpose is for cleaning, repairing, and decontaminating tools and small pieces of equipment. The rooms are equipped with shelves, tables, hooded sinks, welding equipment, a lathe, a saw, a drill press, and tanks. For larger pieces of equipment, cleaning and decontamination are assumed to take place in the spent fuel shipping cask decontamination pit in the fuel handling building. Maintenance of large equipment that is impossible to do in the plant building can be performed in YGN maintenance working shop. 425

b. Personnel Decontamination Facility

A personnel decontamination facility is incorporated into the design of the restricted area access control station located in the access control building. This room contains a shower, sink, radiation measuring equipment, and change areas.

c. Laboratory Complex

The plant's laboratory complex consists of a radiochemistry laboratory, a counting room(s) where liquld, gaseous, and smear radioactive samples can be analyzed and a storage where bulk quantities of chemicals and laboratory supplies are stored. 61

1. Radiochemistry Laboratory

The radiochemistry laboratory was designed to provide for the safe and efficient processing and analysis of radioactive and potentially radioactive samples. Such samples may be expected for such systems as the primary coolant loop, chemical and volume control, fuel handling and storage, steam generator blowdown, and radwaste, etc. It is located on the [redacted] of the access control building. 61

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

To minimize the accumulation and spread of surface contamination, floor coatings, surface coatings, workbench surfaces, fume hood interiors, and sink and drain pipe materials were chosen to minimize the adherence and ease the removal of contamination. To

minimize the spread of airborne radioactivity, fume hoods are provided for the storage and processing of volatile radioactive
61 samples, and the radiochemistry laboratory is kept at a negative pressure with respect to all adjacent areas. Air exhaust from this laboratory requires filtering before it is released to the environment.

61

2. Counting Rooms

The counting rooms are located near the radiochemistry laboratory on [redacted] of the access control building. These rooms are provided with computer-grade regulated electric circuits and nonfluorescent lighting to ensure the optimal performance of the counting equipment. The desired radiation level in the counting room will be below background (0.1 mrem/hr). To ensure that the counting room will not be affected by any in-plant airborne radioactivity, the rooms are maintained at a positive pressure with respect to all surrounding areas and ventilated with fresh filtered and conditioned air. The room HVAC is designed to maintain the temperature and humidity tolerances required by the detectors and their associated electronics and computer equipment. The use of thick concrete walls for shielding the counting rooms is precluded because of natural radiation emanating from the concrete itself.

The equipment being considered for the counting rooms includes a gamma-ray spectroscopy system, an automatic low background alpha/beta counting system, and standard portable alpha/beta counting equipment.

Localized radiation shielding will be provided for the counting equipment when needed.

d. Laundry Facility

The plant laundry facility is located on the basement floor of the access control building. It is designed to receive, decontaminate, store, and distribute the radiological protective clothing used in the plant. The major equipment includes dry cleaning units, wet washers, conventional clothes dryers, a surface contamination monitor for detection of contaminated cloth, a laundry chute where incoming contaminated apparel will be dropped off, a sorting table with local ventilation, shinks, and a folding and monitoring table. The floor, surface coatings, and equipment in the laundry were chosen to minimize the build-up and ease the removal of surface contamination. The laundry is kept at negative pressure with respect to all surrounding areas to minimize the spread of airborne contamination originating from the handling of contaminated protective equipment. Air within the laundry flows from the clean folding area to the potentially dirty sorting area. 386

The laundry also includes a mask (respirator) cleaning facility. This facility includes space and equipment for collecting, cleaning, inspecting, and temporary storage of respiratory protective equipment.

e. Calibration Room

Each unit has an instrument calibration room located in the basement of the access control building. This room is designed to provide a location where radiation protection instrumentation can be calibrated, stored, serviced, and decontaminated by vacuuming when necessary. Adequate radiation shielding was provided to protect adjacent general access areas.

12.3.1.4 Radiation Zoning and Access Control

12.3.1.4.1 Normal Operation

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 shall normally gain access to radiation controlled areas through the access control building. The flow of personnel is shown in Figure 12.3-2. For the personnel who need routine access to the power block, but do not perform any radiation work, such as main control room operators, a clean pathway is provided that is separated from the potentially contaminated pathway used by radiation workers.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposure ALARA and within the standards of Korean AEA and related regulations. Each room, corridor, and pipeway of every plant building has been evaluated for potential radiation sources during normal operation and shutdown. Radiation zones are determined on the basis of maximum radiation level and occupancy requirements. Descriptions of radiation zone categories to be employed in normal operating condition are given in

Table 12.3-1, and the specific zoning for each plant area is shown in Figures 12.3-3 through 12.3-7. The radiation zone for Hanbit 3&4 Temporary Storage is described in Figure 12.3-14. All frequently accessed areas, such as corridors, are shielded for Zone 1 or Zone 2 access. 766

Ingress or egress of plant operating personnel to controlled access areas is to be controlled by the plant health physics staff to ensure that radiation levels and allowable working time are within the limits prescribed by Korean AEA and related regulations.

Cubicle entrances are designed to satisfy the shielding design criteria given in Subsection 12.3.2 and permit the appropriate cubicle access. The most frequently specified entrance into radiation cubicles is a labyrinth structure. Figure 12.3-8 shows the important features of this structure. The next most frequently used cubicle entrance is a removable hatch or plug. A typical floor hatch is shown on Figure 12.3-9. Removable concrete shield wall sections are specified to accommodate maintenance and equipment removal, where necessary. Each labyrinth roof has been evaluated individually to determine the amount of radiation shielding that the roof requires to protect stairs, galleries, and other occupied areas.

12.3.1.4.2 <u>Postaccident Conditions</u>

Another function of the plant radiation shielding system is to ensure access to and occupancy of vital areas during a design basis accident in order to mitigate or recover from the accident.

General Design Criteria (GDC) 19 to 10 CFR 50, Appendix A requires the following:

> "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its

equivalent to any part of the body, for the duration of the accident."

NUREG-0737 gives additional guidance for establishing post-design-basis accident radiation exposure limits for vital areas.

a. Post-Design-Basis Accident Zone Classifications

Radiation zone maps have been prepared to illustrate the post-accident radiation environment. These zone maps are based on the source term and scenario of design basis accident (i.e., large LOCA) and the layout of piping, equipment, and shielding within the plant.

They have been prepared for several times after the onset of the accident. The number of times should be adequate to depict the temporal behavior of the radioactivity. Times of 1 hour, 1 day, and 1 week are typical and are important for evaluating the near-term emergency situation. The post-design-basis accident radiation zone designations used to classify the radiation zones and radiation zone maps for the containment and primary auxiliary building resulting from the design-basis accident are shown on Table 12.3-2 and in Figures 12.3-10 through 12.3-12, respectively.

b. Vital Area Definition and Identification

NUREG-0737, II.B.2 defines a vital area as any area that will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. It states that the control room, technical support center (TSC), sampling station and sampling analysis area must be classified as vital areas. It further states that the evaluation to determine the necessary vital areas should include (but not be limited to) a consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be performed for these areas if they are determined not to be vital.

The vital areas for YGN 3&4 consists of the following:

1. Main control room complex

2. Technical support center (TSC) and satellite technical support center (STSC)

3. Postaccident sampling area

4. Postaccident sample preparation and analysis areas

5. Hydrogen recombiner control panel areas

6. Hydrogen recombiner areas

The shutdown cooling heat exchanger rooms, the low-pressure safety injection (LPSI) pump rooms (also used as shutdown cooling pumps), the containment spray (CS) pump rooms, and the high-pressure safety injection (HPSI) pump rooms are major ECCS equipment rooms that would be used during a design-basis accident. These rooms are not vital access areas, and there are no vital access areas in the immediate vicinity of these rooms. All safety-related equipment located within or just outside these rooms is qualified for the expected LOCA conditions (e.g., the qualification radiation dose within these rooms is 1.0×10^7 rads). These rooms are located on the lowest elevation of the primary auxiliary building

Vital areas and vital area access routes are identified on Figure 12.3-13.

c. Vital Area Design Criteria

The radiological design criteria for vital areas are dictated by the requirements of NUREG-0737, II.B.2, which states that the design dose rate for personnel in a vital area shall be within the guidelines of GDC 19 of 10 CFR 50, Appendix A. Specifically, GDC 19 states that personnel radiation exposures shall not exceed 5 rem whole body or its equivalent to any part of the body for the duration of the accident. While located in a vital area, occupancy requirements should be considered. For vital areas or vital activities that do not require continuous occupancy for the duration of the accident (e.g., post-accident sampling and analysis), the radiation exposure shall not exceed 5 rem whole body or its equivalent during the time required to access and be in the vital area or required to perform the vital activity. In addition to complying with the GDC 19 dose requirement, the post-LOCA dose rate to areas that require continuous occupancy (i.e., the main control room complex, TSC, and STSC) is not to exceed 15 mrem/hr averaged over 30 days (NUREG-0737, II.B.2).

The post LOCA dose rates and integrated doses to YGN 3&4 vital areas meet these requirements. The desired dose rates and doses are maintained primarily via radiation shielding (generally concrete) located between contained post-LOCA radiation sources and the vital area.

Additionally, for areas that require continuous occupancy during post LOCA situations, a ventilation system employing filters is provided to remove particulates and iodines from the air to limit internal dose (particularly the thyroid dose) due to inhalation of airborne radionuclides. Inhalation doses to vital areas that do not require continuous post-LOCA occupancy can be limited via the use of respirators or self-contained breathing apparatus.

NUREG-0718, Rev.2, II.B.2 states the following:

"Applicants shall (1) perform radiation and shielding design reviews around systems that may, as a result of an accident, contain TID-14844 source term radioactive material and (2) implement plant design modifications necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. Applicants shall, to the extent possible, provide preliminary design information..... Where new designs are involved, applicants shall provide a general discussion of their approach to meet the requirements by specifying the design concept selected and the supporting design bases and criteria."

Detailed post-LOCA dose calculations for all vital areas or vital area access routes show that those areas meet the dose rate or dose requirements.

d. Vital Area Accessibility

1. Main Control Room Complex

Because the main control room complex requires continuous occupancy during post-LOCA situations, the design dose rate shall not exceed 15 mrem/hr averaged over 30 days.

Access to the main control room complex is via the access control building. The access control building is next to the primary auxiliary building. It is further from the containment building than is the primary auxiliary building. According to detailed post-LOCA dose calculations for the main control room, this area meets the dose rate or dose requirements. The location of and the accessway to the main control room are shown on Figure 12.3-13.

2. Technical Support Center and Satellite Technical Support Center

Because the TSC requires continuous occupancy during post-LOCA situations, the design dose rate shall not exceed 15 mrem/hr averaged over 30 days.

The TSC is located in the access control building of YGN 3. As previously stated, the access control building is next to the primary auxiliary building and further from the containment building than is the primary auxiliary building. Therefore, the direct doses and direct dose rates in the TSC due to post-LOCA sources in the primary auxiliary building and the containment building are smaller than the comparative control room doses and dose rates from these sources. The major post-LOCA sources affecting the TSC are the radioactive sources in the airborne plume located outside the TSC and the airborne radioactive sources located within the TSC. Radioactive sources in the airborne plume outside the TSC originate from unfiltered containment leakage and ECCS equipment leakage, which is charcoal and HEPA filtered by the ECCS room HVAC system before its release to the atmosphere.

Airborne sources are pulled into the TSC from the external airborne plume. These sources are charcoal and HEPA filtered by the TSC ventilation system before entering the TSC. Concrete shielding is provided to shield the TSC from external plume sources. According to the detailed calculation, the post-LOCA dose rate to the TSC does not exceed 15 mrem/hr averaged over 30 days. The location of and the accessway to the TSC are shown on Figure 12.3-13.

The STSC is located in the unit 4 control room complex.

3. <u>Postaccident Sampling, Preparation, and Analyses Areas</u>

The postaccident sampling, preparation, and analyses areas require irregular access, not continuous occupancy, during post-LOCA situations. They are located in the secondary auxiliary building and the access control building

Due to the amount of concrete, the dose rate and integrated dose contributions to these areas from post-LOCA sources in the containment building and primary auxiliary building are insignificant. The major dose rate and dose acquired during sample collection activities is from postaccident fluids in sample lines in the sampling room.

According to NUREG-0737, II.B.3, a sample must be obtained and analyzed within 3 hours after an accident without incurring a radiation exposure to any individual in excess of 5 rem to the whole body or 75 rem to the extremities. To ensure this, shielding is provided around these sample lines so that the dose rate in accessible areas in the sample room during post-LOCA sample taking will be no greater than 100 mrem/hr.

Shielding for sources in the sample sink is the responsibility of the sample vendor. Normally, lead shot is provided in the face of the sample sink for shielding. The dose rate criterion is 100 mrem/hr at the face of the sample sink from the worst expected postaccident source. A dose rate no greater than 100 mrem/hr from sample lines during sample taking ensures that the dose from sample taking is a small fraction of the 5 rem whole body dose limit allowed during sample acquisition, preparation and analysis.

The other major postaccident dose contributions during sample acquisition, preparation, and analysis are from sample handling and from exposure to a postaccident airborne source within the secondary auxiliary building and the access control building. The detailed analyses for dose contributions from these sources show that the total doses exposure during sample acquisition, preparation, and analysis does not exceed the dose requirements. The locations of and the access way to the postaccident sampling room and analysis areas are shown on Figure 12.3-13.

4. Hydrogen Recombiner Areas and Hydrogen Recombiner Control Panel Areas

Hydrogen recombiner areas and hydrogen recombiner control panel areas require irregular access, not continuous occupancy, during post-LOCA situations. The YGN 3&4 hydrogen recombiners are portable units (i.e., they are to be installed after initiation of the LOCA). To enable installation, post-LOCA access must be available to bring the hydrogen recombiners into the plant and to the locations for hydrogen recombiner installation.

The hydrogen recombiners are installed in the primary auxiliary building (outside the containment building wall) at [redacted]

The hydrogen recombiners are installed no later than 4 days after a LOCA. Installation time required is 20 hours.

After installation, the hydrogen recombiners can be operated from the control room rather than at the local control panels because the hydrogen recombiner control switches are also provided in the control room. Therefore, hydrogen recombiner control panel areas require access only at times when controlling hydrogen recombiners is impossible in the main control room. This is expected to be an infrequent occurrence.

The locations of and the access way to these areas are shown on Figure 12.3-13.

12.3.2 Shielding

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

12.3.2.1 General Shielding Design Criteria

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

The shielding design dose rates for YGN 3&4 meet Korean AEA and related regulations and 10 CFR 50, which are concerned with allowable radiation to individuals in restricted and unrestricted areas. The only shielding required to be safety-related is the control room and the primary containment shielding; this shielding satisfies the requirements stated in Criterion 19 of 10 CFR 50, Appendix A, and Korean AEA and related regulations

Radiation protection of personnel, equipment, and materials is largely dependent upon the adequacy of the design of the station shielding system.

Radiation shielding has the passive protection function of radiation attenuation and consists of material placed between radiation sources and personnel and/or equipment and materials needing protection from radiation. The shielding system is designed and constructed to assure that the station can be operated and maintained such that the resultant radiation level and doses are within the limitations of applicable regulations and are as low as is reasonably achievable (ALARA). Specific design dose rate limits recommended to achieve this objective are discussed in Subsection 12.3.1.4 and listed in Table 12.3-1.

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

a. Normal Operating Conditions

For shielding design, normal station operating conditions are considered to include conditions generally known as anticipated abnormal operational occurrences. The two modes of normal station operation are

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

2. normal shutdown of the reactor (including refueling operations).

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

b. Accident Conditions

Station shielding provides protection to plant operating personnel and the general public under postulated design-basis accident conditions as defined in Chapter 15.

1. Control Room Habitability

The main control room and associated areas are shielded such that, after a postulated design-basis LOCA, the whole body dose in the control room for the duration of the accident will not exceed 5 rem or its equivalent to any part of the body, including dose contribution during ingress and egress, as per requirements 10 CFR 50, Appendix A, Criterion 19. Subsection 6.4.2.5 describes control room shielding.

The radiation shielding protecting the main control room (and associated areas) is designed based on the anticipated radiation environment resulting from the postulated LOCA.

2. Direct Offsite Doses

Adequate plant shielding is provided to limit accident site boundary doses, due to direct and scattered radiation from contained sources within the plant, to the limits specified in 10 CFR 100 and Chapter 15 of the Standard Review Plan conditions.

c. Seismic and Safety Classification

Structural walls of the station are designed, as required, to meet seismic category requirements. Shielding walls are designed to Seismic Category I, II, or III criteria depending upon the particular design requirements other than radiation protection requirements (e.g., structural integrity and load bearing capacity) that the walls must meet.

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

d. Protection of Equipment

Appropriate shielding is provided, where needed

1. to limit radiation heating of structural concrete,

2. to reduce neutron activation of equipment, and

3. to limit the radiation dose to equipment and materials.

Protection from neutrons and from neutron-induced gamma rays is important around neutron sources such as the nuclear reactor core. The primary shield around the reactor vessel and any neutron shielding near the reactor vessel nozzles are examples of plant shielding designed to protect personnel and equipment against neutron radiation and neutron-induced gamma rays.

e. Maintenance, Inspection, and Testing Considerations

Appropriate shielding is provided to ensure personnel access and sufficient occupancy time to areas containing equipment that requires maintenance, inspection, and testing.

f. Additional Requirements

In addition to the radiation protection functions discussed above, the shielding systems have other functional requirements. Generally, these depend upon the location of the shield and the access requirements to equipment or areas beyond the shield. Thus, access to an area is through the shield itself; e.g., through removable shield walls. Removable shield walls, portable shields, and compensatory shielding are discussed in Subsection 12.3.2.5.2.

12.3.2.2 General Shielding Design

Shielding is provided to attenuate direct and scattered radiation to levels less than the upper limit of the radiation zone for each area. The shielding requirements for all plant areas are determined during the design phase that proceeds building construction. Design criteria for penetrations comply with the intent of Regulatory Guide 8.8 and are discussed in Subsection 12.3.2.5.5.

The material used for most of the plant shielding is ordinary concrete with a bulk density of 145 ± 5 lb/ft^3 (2.33 ± 0.8 g/cm^3). Whenever poured-in-place concrete has been replaced by concrete blocks or other material, design ensures 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 Section 1.8. Water is used as the primary shield material for areas above the spent fuel transfer and storage areas.

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, steam generator, and pressurizer compartment shield walls, reduces radiation levels outside the containment building to less than 0.5 mrem/hr.

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 Subsection 12.3.2.2.7).

Where personnel, emergency, 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.

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 (i.e., primary shield), which is approximately 7 feet (2.1 m) thick, and by the concrete secondary shield, which is 4 feet (1.2 m) 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 designed using the computer program [redacted] The [redacted] 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-23 CASK cross sections library (Reference 2) was used for the scattering data base. Layout details, such as air gap, concrete thickness, and core internals, are conservatively incorporated into the model.

Post-shutdown shielding considers the source strengths given in Tables 12.2-5 and 12.2-11. Shielding codes such as , and (see Subsection 12.3.2.3) were used for these shutdown shielding analyses.

The reactor cavity streaming shield is installed to limit neutron and gamma fluxes at the reactor vessel flange area, at the operating floor elevation, and to the areas outside the primary shield wall hot-and cold-leg piping penetrations. The reactor cavity streaming shield is composed of two shields (an upper straining shield and a lower streaming shield). The design description for the reactor cavity streaming shield is as follows:

a. Lower reactor cavity streaming shield

- Minimum shield thickness: 3.5 ft (1.1 m)
  (Top of shield is at elevation 102 ft)
- Air gap size between reactor vessel and shield wall: 7.5 in. (19 cm)

b. Upper reactor cavity streaming shield

- Minimum shield thickness: 3 ft (0.9 m)
  (Top of shield is at elevation 112 ft - 6 in.)
- Air gap size between reactor vessel and shield wall: 5 in. (12.7 cm)

Shielding calculation of the reactor cavity streaming was performed using the computer program (Reference 3). The calculational results show that the 164
operating floor dose rate is less than 100 mrem/hr.

Design considerations for the cavity to facilitate inservice inspection and adequate venting areas are incorporated.

The secondary shield is a reinforced concrete structure surrounding the reactor coolant equipment, including piping, pumps, and steam generators. This shield protects personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full power operation.

Several components of the CVCS are located in shielded compartments, which are normally Zone 6 areas, inside the containment. 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 regenerative heat exchanger and the reactor drain tank.

After shutdown, most of the containment is accessible for limited periods of time and all access is controlled.

Shutdown dose rates are expected to range from 0.5 to 1000 mrem/hr 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, shielding is provided for areas surrounding the refueling canal to ensure that radiation levels remain below the zone levels specified for adjacent areas. Water provides the shielding over the spent fuel assemblies during fuel handling.

Furthermore, substantial structural barriers to limit access in the vicinity of the fuel transfer tube during fuel handling operations are provided.

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, the shutdown cooling system, the spent fuel pool cooling and cleanup system, and the primary sampling system.

Shielding is provided as necessary around the following equipment in the auxiliary building to ensure the design radiation zone and access requirements are met for surrounding areas:

a. Letdown heat exchanger and piping
b. Purification, preholdup, and deborating ion exchangers
c. Volume control tank
d. Charging pumps and piping
e. Shutdown cooling heat exchangers
f. LPSI pumps
g. Equipment drain tank and reactor drain pumps
h. Chemical waste drain tank
i. CVCS filters
j. Spent fuel pool cleanup ion exchangers and filters
k. High activity spent resin tank and pump
l. Gas stripper
m. Seal injection heat exchanger
n. Boronometer

o. Process radiation monitor
p. Boric acid concentrator
q. Boric acid condensate ion exchangers

Shielding is based upon operation with maximum activity conditions as discussed in Sections 11.1, 11.2, 11.3, and 12.2.1.

Depending on the equipment in the compartments, the radiation zones vary from 2 through 6. Corridors are shielded so that they are Zone 2 areas, and operator areas for valve compartments are generally limited to Zone 3 access.

Whenever practicable, frequently operated valves in high-radiation areas are provided with remote actuators extending to design radiation Zone 2 or Zone 3 areas.

Removable sections of concrete slab and concrete plugs are utilized as necessary for equipment maintenance and spent filter cartridge replacement. Permanent or temporary shielding is considered between equipment in compartments with more than one piece of equipment in order to shield maintenance workers from adjacent radiation sources.

Following reactor shutdown, the shutdown cooling system (SCS) pumps and heat exchangers are in operation to remove heat from the reactor coolant system. The radiation levels near 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 SCS equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas.

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 values. The building external walls are sufficient to shield external plant areas to Zone 1 values.

Water in the spent fuel pool provides shielding above the spent fuel transfer and storage areas. Personnel radiation levels surrounding the fuel handling equipment areas are limited to less than 2.5 mrem/hr 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 (SFPCC) system (Subsection 9.1.3) shielding was based on the maximum activity discussed in Subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the SFPCC system to be shielded includes the SFPCC heat exchangers, pumps, and piping. (SFPCC filters and ion exchangers are located in the secondary auxiliary building.)

12.3.2.2.5 Radwaste Building Shielding Design

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

a. Radwaste holdup tanks and pumps
b. Monitor tanks
c. Radwaste evaporators
d. Waste solidification equipment
e. Waste drumming and storage areas
f. Concentrate tanks and pumps
g. Radwaste piping
h. Filters and demineralizers

i. Low activity spent resin tank and pump
j. Gaseous radwaste system (GRS) head drain tank
k. GRS charcoal guard beds and delay beds
l. Evaporator feed tanks and pumps
m. Demineralizer feed tank and pumps

Shielding is based on operation with maximum activity conditions as discussed in Chapter 11.

Depending on the equipment in the compartments, the radiation zones vary from 2 through 6. Corridors are shielded so that they are Zone 2 areas, and operator areas for valve compartments are generally limited to Zone 3 access, as shown on the radiation zone drawings of Figure 12.3-6.

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

The need for partial shield walls between equipment in compartments with more than one piece of equipment was evaluated for dose savings during maintenance operations.

12.3.2.2.6 Turbine Building Shielding Design

Radiation shielding is not required for most process equipment located in the turbine building. The condensate polishing system was analyzed to determine whether any shielding is required if substantial primary to secondary system leaks develop. All areas in the turbine building are normally classified as Zone 1 areas.

12.3.2.2.7 Control Room Shielding Design

Figure 12.3-3 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 for the duration of the accident with radiation doses limited to 5 rem whole body from all contributing modes of exposure. This is in accordance with 10 CFR 50, Appendix A, General Design Criterion 19.

The design basis LOCA is described in Subsection 15.6.5. The direct radiation from airborne fission products inside the containment, from fission products deliberately collected on HVAC filters, and from the radioactive cloud surrounding the control room was considered when calculating the dose to personnel inside the control room following a postulated LOCA. The shielding of the control room ensures compliance with 10 CFR 50, Appendix A, General Design Criterion 19.

In addition to the parameters of Regulatory Guide 1.4, the following was considered in the demonstration of the control room habitability:

a. Shielding by the containment outer wall
b. Shielding by control room walls and slabs
c. Radiological decay
d. Iodine species distribution and containment spray removal of iodine as discussed in Subsection 6.5.2

No credit is taken for shielding by the internal structures in the containment.

12.3.2.2.8 Miscellaneous Plant Areas and Plant Yard Areas

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 working shop including Old Reactor Vessel Head Storage Facility 689 containing radiation sources. Plant yard areas that 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. If necessary, radiation shielding was provided for outside tanks so that the surface dose rate does not exceed 0.5 mrem/hr.

12.3.2.2.8.1 The Shielding of Hanbit 3&4 Temporary Storage

The major radiation sources in the Hanbit 3&4 Temporary Storage are two old reactor vessel heads and four old steam generators. The shielding for the old steam generators and reactor vessel is performed under the postulation of the radiation sources which are a crud in the plenum and U-Tube of old steam generator and reactor vessel head. The crud thickness for the vessel head and 766 plenum, and U-Tube are considered to 1.0E-03 g/㎠ and 1.0E-04 g/㎠, respectively. The shielding for Hanbit 3&4 Temporary Storage is designed to meet the radiation zone designation and access basis for the neighboring rooms of the old steam generator storage room. The maximum concrete wall thickness is designed with 45cm to meet the radiation dose rate less than 0.001 mSv/hr.

12.3.2.3 Shielding Calculational Methods

The radiation exposure of individuals, equipment, and materials is a function of the following basic parameters, which are given due consideration in the shielding design:

a. Source strength (type, intensity, energy)

b. Number of sources, source geometry, and self-absorption factors

c. Shielding material, geometery, and mass between source(s) and receptor

d. Distance between source(s) and receptor

e. Time that receptor is exposed

f. Allowed dose rate or dose

Where radioactive crud buildup sources were known, the source strength parameter was appropriately adjusted and the shielding designed to accommodate the effects of crud buildup for at least 10 years of reactor operation. Where

INTENTIONALLY BLANK

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

The provided shielding thicknesses ensure compliance with plant radiation zoning. To minimize plant personnel exposure, maximum equipment activities under the plant operating conditions described in Subsection 12.2.1, rather than annual average activities, are used to design the radiation shielding. 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 decay gamma spectrum from the assumed radionuclide sources are distributed over a fixed energy bin structure for dose rate calculations.

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.

The [redacted] computer program has been used for shielding analysis. This program calculates gamma dose rates from well-defined, fixed-source geometries ranging from a simple point or line to spheres, cylinders, or rectangular parallelepipeds. Point-and-line kernel integration techniques are combined with the infinite medium buildup factor methodology to obtain dose rates inside volume sources or outside laminar shields that surround the source region. Mass attenuation coefficient and buildup factor data for many shielding materials are included in the program.

For the shielding design of gamma sources given a more complex geometry, the [redacted] or [redacted] code has been used. [redacted] is a general purpose point kernel integration program for estimating the effect of gamma rays that originate in a volume distributed source. The three-dimensional geometry of the program is described by using quadratic equations to define surfaces that bound various source and shield regions. [redacted] is the same program as [redacted] except that this program uses the combinational geometries which are defined using a three-dimensional fixed geometry.

For the analysis of the primary shield, the computer code [redacted] was used. [redacted] is a multigroup one-dimensional discrete ordinates transport program that solves the one-dimensional Boltzmann transport equation for neutrons and gamma rays in slab, sphere, or cylindrical geometry. Using a finite-difference technique, [redacted] allows general anisotropic scattering via an L^{th} order Legendre expansion of the scattering cross-sections.

For design of the reactor cavity, a three-dimensional model was used to simulate radiation streaming from the reactor surface to the containment. This model was implemented using the [redacted] Monte Carlo program. 164

Descriptions of the above computer codes are included in Table 12.3-3.

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

Where shielded entrance ways to compartments containing high radiation sources are necessary, labyrinths or mazes are designed using the general purpose gamma-ray scattering code, G^3 (Reference 7). The labyrinths are so constructed that the scattering dose rate plus the transmitted dose rate through the shield wall from all contributing sources is below the radiation limit specified for the area outside the labyrinth.

12.3.2.4. Specific Shielding Design Criteria

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

Zone classification and dose rate categories for YGN 3&4 are summarized in Table 12.3-1. Figures 12.3-3 through 12.3-7 show the plant radiation zoning to be used for the design of the radiation shielding.

12.3.2.5 Specific Shielding Design

12.3.2.5.1 Shielding Materials and Construction Methods

Bulk shielding structures such as cubicle shielding walls, floors, and ceilings are mainly designed of ordinary concrete, either of (solid) block or poured-in-place construction. Concrete is a mixture of materials, the exact proportions of which may differ from application to application. Concrete for radiation shielding is classified as ordinary concrete. The design of concrete mixtures and forms, the construction of concrete radiation shielding structures, and the quality assurance provisions needed to verify that the desired quality of construction will be achieved are in accordance with accepted design criteria for concrete radiation shields.

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

Removable concrete slabs are provided where necessary to accommodate equipment installation, removal, and construction. Removable concrete slab installation is controlled to ensure as-built radiation attenuation characteristics similar to those expected from equivalent poured concrete.

For the primary shield around the reactor vessel, the expected nuclear heating will not be severe. Therefore, special designs (e.g., water cooling coils) for cooling the primary shield are not required.

Where a potential of leakage or spillage of radioactive material exists, effective features are incorporated into the design of the shielding to prevent the spread of contamination by seepage through walls. As discussed in Subsection 12.3.1.3.4, shielding surfaces shall be coated with a nonporous coating to permit effective decontamination.

12.3.2.5.2 Removable Shield Walls, Portable Shields, and Compensatory Shielding

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

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

Compensatory, portable, or temporary shielding is considered in plant design only as required where other more permanent shielding is not practicable.

Where compensatory shielding is necessary, provisions are made to accommodate such shielding in terms of space, structural loading, clearances, and equipment accessibility.

The plant shielding system uses two types of removable shield walls: stacked unmortared precast concrete panels and shield hatches and plugs. The primary reasons for a removable wall are equipment installation, inspection, maintenance, and removal.

The following paragraphs give guidelines for the design of removable shield walls. The term "major" requires the removal of a removable shield wall in addition to repairing and maintaining equipment. "Seldom" is defined in this section as once a year. "Often" is defined in this section to mean more frequent than once a year.

a. Stacked (Unmortared) Precast Concrete Panels

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

Removable stacked precast concrete panels are used in the shield design when a room contains equipment that seldom requires replacement or major maintenance. The shielded equipment that fits into this category includes heat exchangers, pumps, and radwaste tanks.

b. Removable Shield Hatches and Plugs

Removable shield hatches (or removable floor slabs) and plugs are used in the shield design when a room contains equipment that often requires replacement or maintenance.

In addition to all equipment that falls into the "often" category, shield hatches or plugs are used, whenever practicable, for access to equipment and piping that have, or are in radiation areas that have, a dose rate greater than 3 rem/hr.

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

A radiation detector probe access hole is provided in all filter and demineralizer removable shield hatches so that radiation levels of the contained equipment may be measured without removing the shield hatches. This access hole is a vertical stepped hole bored in the top of the shield hatch. The arrangement is pictured in Figure 12.3-9.

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

12.3.2.5.3 Inservice Inspection and Maintenance Requirements

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

During construction, shield walls are visually inspected for cracks and separations that might compromise the shield.

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

12.3.2.5.4 Shield Thicknesses

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

Shielding thickness is designed with consideration given to all sources in the area including localized hot spots or penetrations. Design parameters are listed in Subsection 12.3.2.3.

Computer codes used in shielding design account for energy spectra and source strengths for each nuclide (including daughter products), material cross sections or attenuation coefficients for each material or element comprising the shield, dose buildup factors, and other relevant parameters.

12.3.2.5.5 Shield Wall Penetrations and Streaming Ratios

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

Compensatory shielding is used where necessary to reduce radiation streaming due to penetrations and localized shield deficiencies (expected hot spots). Techniques used include increased wall thickness, provision for labyrinths or shadow shielding, provision for bends or directing the streaming path away from accessible areas, and use of higher density materials such as lead, steel, or lead wool.

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

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

a. Five times the design dose rate for "uncontrolled" access areas

b. Five times the design dose rate for penetrations located from 0 to 10 feet (0 to 3 m) above the floors in "controlled" access areas which have design dose rates ≤ 10 mrem/hr and

c. Ten times the design dose rate for penetrations located more than 10 feet (3 m) above the floor in "controlled" access areas

For "controlled" access areas having design dose rates of greater than 10 mrem/hr, specific streaming ratios for penetrations less than 10 feet (3 m) above the floor are area-dependent and may be more or less restrictive than those for "controlled" access areas of ≤ 10 mrem/hr.

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

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

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

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

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

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

d. The penetration is located as high above the floor as is practicable and not less than 10 feet (3 m).

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

f. The diameter of the penetration is as small as practicable. For electrical penetrations, use of sleeves or conduit having larger than 6 inch (15 cm) nominal diameter is avoided.

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

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

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

j. Offset penetrations are used when large lines or ducts penetrate shielding walls of cubicles that contain high levels or radiation, i.e., shield walls greater than or equal to 3 feet (91 cm) thick. HVAC ducts and openings are the most common penetrations that incorporate offsets, but in general, offsets are not used unless no other method will work.

12.3.3 Ventilation

The design of plant heating, ventilating, air conditioning (HVAC), and air cleaning systems protects plant operating and maintenance personnel and the general public from exposure to radiation from airborne sources under all operating conditions including refueling, maintenance, and anticipated operational occurences.

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 Korea nuclear regulations and 10 CFR 50, Licensing of Production and Utilization Facilities, as discussed in Subsection 12.3.3.2.

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

a. The dose equivalent received by members of the public due to station operation will be as low as reasonably achievable (ALARA).

b. The occupational dose equivalent received by onsite personnel during construction and operation will be as low ALARA.

c. The absorbed dose received by the station equipment will not compromise its availability or operability.

d. Stack releases will not cause offsite doses to exceed the limits set forth in 10 CFR 100 during accident conditions or the limits set forth in 10 CFR 50 Appendix I for normal operating conditions.

e. The above criteria will be met in a reasonable and prudent fashion so as to allow for the safe, efficient, and economical operation of station.

12.3.3.3 Design Guidelines

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

a. Keep areas with a potential for airborne contamination at a negative pressure with respect to surrounding areas.

b. Maintain flow direction from areas of less potential to areas having greater potential for airborne contamination.

c. Where practical, locate exhaust ducts in the vicinity of the sources of airborne contamination.

d. Provide for special filtration of air exhausted from areas with a potential for airborne contamination.

e. Ensure that the air exhausted from areas of potential airborne contamination is not used to ventilate other accessible areas.

f. Provide for continuous air monitors or fixed sampling equipment as appropriate.

g. Maintain a minimum particulate transport velocity of 1,500 feet per minute (457 m/sec) in main ducts carrying potential airborne radioactivity.

h. Maintain a minimum airflow velocity (25 ft/min) (7.6 m/min) through open penetration and doorways leading to areas having potential airborne contamination.

i. Provide cubicle airflows in potentially occupied cubicles sufficient to maintain airborne concentrations less than DAC(Derived Airborne Concentration) per the NSSC Notice 2014-34(방사선방호 등에 관한 기준), where practicable. 812

j. Don not recirculate exhaust air from areas where the potential for radioactive contamination may exist to the served spaces.

k. Ensure that isolation damper closure time is less than air travel time from detection point to isolation point for those dampers utilized for the isolating a portion of the system from the effect of airborne radioactivity.

l. Locate all air intakes at a sufficient distance from any gaseous or exhaust release points to preclude reentrainment under anticipated operating conditions.

m. Locate duct accessories (test point, etc.) required for testing outside contaminated areas.

n. Evaluate all potentially radioactive ducts to determine proper routing and shielding requirements, based on minimizing radiation exposure to station operating and maintenance personnel.

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 9.4.6.)
b. Primary auxiliary building (see Subsection 9.4.3.1)
c. Secondary auxiliary building (see Subsection 9.4.3.2)
d. Fuel building (see Subsection 9.4.2)

e. Radwaste building (see Subsection 9.4.4)
f. Emergency core cooling (ECCS) HVAC system (see Subsection 9.4.5.3)
g. Access control building (see Subsection 9.4.7.1)

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

Other structures (e.g., pump intake structures and the auxiliary boiler building 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 cleaning units ventilation systems. The exceptions to Regulatory Guide 1.52 and Regulatory Guide 1.140 are discussed in Section 1.8.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Area radiation and airborne radioactivity monitors 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 main 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 fuel accident monitors in the containment building and fuel building perform safety-related functions to actuate systems that mitigate the consequences of postulated fuel handling accidents.

The ARMS 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 Korean AEA and related regulations, 10 CFR 50, and Regulatory Guides 8.2 and 8.8. Selected ARMS monitors perform safety- related function, and provides signals to the BOP ESFAS for the generation of the containment purge isolation actuation signal (CPIAS) and fuel building emergency ventilation actuation signal (FBEVAS). The ARMS-monitored postaccident 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

The radiation monitoring system (RMS) will identify a fuel handling accident in the containment or fuel building and generates a CPIAS or FBEVAS for actuation of the appropriate BOP engineered safety feature (ESF) systems.

The RMS performing safety-related functions is designed in accordance with IEEE standards 279, 323, and 344 to meet the single failure criterion, separation, isolation, enviromental, and seismic qualification requirements. These monitors together with selected PERMS (process and effluent radiation monitoring system) monitor described in Section 11.5 for the BOP ESFAS. The PERMS monitors generate the BOP ESFAS control room emergency ventilation actuation signal (CREVAS).

The safety evaluation of the ESFAS function of these monitors is discussed in Section 7.3.

12.3.4.1.1.2 Power Generation Design Basis

a. Local detectors are designed to function properly in the ambient environment of their locations.

b. Local detectors in the auxiliary, radwaste, fuel, containment, and access control buildings operate at normal atmospheric pressures. Detectors located in the containment building are removed temporarily during containment pressure tests (these are not the containment high-range area monitors listed in paragraph f in Subsection 12.3.4.1.1.2).

c. 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.

d. 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.

e. 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.

f. All area radiation monitors are powered from the non-Class 1E instrumentation ac buses, with the exception of the CPIAS and FBEVAS monitors. These monitors are powered from the Class 1E instrument ac power system.

g. Areas that contain a liquid, gaseous, or particulate radiation source that potentially could produce a dose rate during normal operation greater than 2.5 mrem/hr 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 accidential 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 (e.g., the volume control 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.

h. 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 main control room, of radiation levels at selected locations within various plant buildings. The area monitors listed in Table 12.3-4 accomplish the following design objectives:

1. Assist plant operators in making decisions on deployment of personnel in the event of an accident resulting in a release of radioactive material in the plant.

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 monitoring for use in ensuring 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 warns 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 setpoints are determined by the normal background radiation at the detector location, the limits for personnel exposure in restricted areas, and the calculated levels for normal operating conditions in the area.

The CPIAS and FBEVAS monitors provide for fuel handling accident monitoring in the containment or fuel buildings perform an additional safety function. The FBEVAS closes the normal fuel building ventilation system and activates the fuel building emergency ventilation system. The CPIAS initiates containment building purge isolation.

The containment high range area monitors are qualified for operation under post-LOCA environment. These monitors provide postaccident indication of radiation level inside the containment. This capability complies with Item II.F.1 of NUREG-0718.

Criticality alarm monitors are not required based on the fuel storage and handling design features for YGN 3&4. This design is acceptable per Regulatory Guide 8.12, as discussed in Appendix 1A.

i. 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.

j. Each channel local control unit 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.

k. To minimize occupational radiation exposure to maintenance personnel, the ARMS is modularized as much as practicable for rapid replacement of components, calibration and installation.

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 main control room, of radiation levels at selected locations within various plant buildings where radioactive materials may be present, stored, handled, transported, or inadvertently introduced.

Infrequently accessed radiation areas that are not monitored continuously are monitored with portable devices before and during access.

Each channel of the ARMS consists of a fixed-position Geiger-Mueller (G-M) tube and/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 provide local and remote alarms upon a high radiation level.

The radiation energy in the monitored area is detected by the G-M tube and/or ion chamber. The output from each detector is processed locally by a microprocessor and transmitted to radiation monitoring cabinets located in the computer room. The radiation monitoring system (RMS) is a microprocessor-based system. The radiation level is displayed at the local control unit, RMS CRTs in the control room, health physics room, TSC, and EOF. The range and location of the ARMS monitors are indicated in Table 12.3-4. Abnormal radiation levels are annunciated by visual and audible alarms, both locally and in the main control room. Monitor alarm setpoints are selected on the basis of the normal background radiation at the detector location, on the limits for personnel exposure in restricted areas, and on the calculated levels for normal operating conditions in the area.

Each channel is checked for trend changes in each shift, safety-related channels 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.

Location of ARMS monitors is provided in Table 12.3-4.

The CPIAS and FBEVAS monitors perform safety functions. A high radiation signal in the fuel building initiates closure of the normal fuel building ventilation system and activates the fuel building emergency exhaust system. A high radiation signal in the containment initiates containment building purge isolation.

The ARMS meets the requirements of Regulatory Guide 1.97. The high range containment area radiation monitors are provided as discussed in Item II.F.1 of NUREG-0718.

The ARMS for Hanbit 3&4 Temporary Storage conforms to ANS 6.8.1. 766

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 Unit Panel

The local unit panel contains the following:

a. Monitor on-off switch
b. High radiation level light
c. Monitor available light
d. High-high (interlock) radiation level light
e. Radiation level display
f. High radiation alarm
g. Interlock switching function to process control devices (if required)
h. Instrument failure alarm light

The safety-related CPIAS and FBEVAS monitor local units provide Class 1E isolated contact output for the BOP ESFAS.

12.3.4.1.2.3 Control Room Readout

The system CRTs located in the main control room, health physics room, computer room, TSC located in Unit 3, STSC located in Unit 4 computer room, and EOF display all system data. The keyboards in the health physics room and computer room are fully interactive and able to enter changes to the channel file. Other keyboards are capable of display control only, with no ability to enter changes to the channel file.

12.3.4.1.2.4 System Outputs

A safety-related divisionalized cabinet provides Class 1E analog output for the recorders located in this cabinet.

Radiation monitoring system cabinets provide a contact for high-radiation common alarm on the main control board, and individual alarms are displayed on the RMS CRT.

12.3.4.1.2.5 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 12.3-4.

12.3.4.2 Airborne Radiation Monitoring System

The HVAC system in-line airborne radioactivity monitors are as follows:

a. Primary auxiliary building HVAC exhaust monitor
b. Secondary auxiliary building vent exhaust monitor
c. Fuel building exhaust monitor
d. Containment purge exhaust monitor
e. Main control room outside air intake monitor
f. Containment air monitor
g. Radwaste area vent exhaust monitor
h. Condenser vacuum pump exhaust monitor
i. Gland steam exhaust monitor
j. Combined gaseous radwaste system cubicle exhaust monitor
k. TSC air intake monitor
l. Miscellaneous process monitors

These monitors are discussed in Section 11.5 and are tabulated in Table 11.5-1.

12.3.4.2.1 Design Basis

12.3.4.2.1.1 Safety Design Basis

The safety design basis of the airborne radioactivity monitors is presented in Subsection 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 Subsection 11.5.1.2.

12.3.4.2.2 <u>System Description</u>

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

12.3.5 References

TABLE 12.3-1

RADIATION ZONE DESIGNATION

TABLE 12.3-2

POSTACCIDENT RADIATION ZONE DESIGNATION

* RL = radiation level

TABLE 12.3-3

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

TABLE 12.3-4 (Sh. 1 of 4)

TABLE 12.3-4 (Sh. 3 of 4)

TABLE 12.3-4 (Sh. 4 of 4)

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

TYPICAL LAYOUT ARRANGEMENTS
FOR VALVES, PUMPS, MAJOR
COMPONENTS, AND PANELS

Figure 12.3-1

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

FLOW OF PERSONNEL WITHIN
RADIATION CONTROLLED AREA

Figure 12.3-2

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 1 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 2 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 3 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 4 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 5 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 6 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 7 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 8 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
PRIMARY AUXILIARY/CONTAINMENT
BUILDING
(Sheet 9 of 9)
Figure 12.3-3

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
SECONDARY AUXILIARY BUILDING
(Sheet 1 of 4)

Figure 12.3-4

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
SECONDARY AUXILIARY BUILDING
(Sheet 2 of 4)

Figure 12.3-4

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
AUXILIARY BUILDING
(Sheet 3 of 4)

Figure 12.3-4

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
SECONDARY AUXILIARY BUILDING
(Sheet 4 of 4)

Figure 12.3-4

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
FUEL BUILDING
(Sheet 1 of 2)

Figure 12.3-5

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
FUEL BUILDING
(Sheet 2 of 2)

Figure 12.3-5

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
RADWASTE BUILDING
(Sheet 1 of 4)

Figure 12.3-6

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
RADWASTE BUILDING
(Sheet 2 of 4)

Figure 12.3-6

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
RADWASTE BUILDING
(Sheet 3 of 4)

Figure 12.3-6

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
RADWASTE BUILDING
(Sheet 4 of 4)

Figure 12.3-6

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

RADIATION ZONES (NORMAL)-
ACCESS CONTROL BUILDING
(Sheet 1 of 4)

Figure 12.3-7

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

RADIATION ZONES (NORMAL)-
ACCESS CONTROL BUILDING
(Sheet 2 of 4)

Figure 12.3-7

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) - ACCESS CONTROL BUILDING
(Sheet 3 of 4)

Figure 12.3-7

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (NORMAL) -
ACCESS CONTROL BUILDING
(Sheet 4 of 4)

Figure 12.3-7

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

SAMPLE SKETCH OF
LABYRINTH ENTRANCE

Figure 12.3-8

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

TYPICAL ARRANGEMENT OF
SHIELD HATCH

Figure 12.3-9

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 HOUR AFTER ACCIDENT
(Sheet 1 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 HOUR AFTER ACCIDENT
(Sheet 2 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 HOUR AFTER ACCIDENT
(Sheet 3 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 HOUR AFTER ACCIDENT
(Sheet 4 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 HOUR AFTER ACCIDENT
(Sheet 5 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 HOUR AFTER ACCIDENT
(Sheet 6 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 HOUR AFTER ACCIDENT
(Sheet 7 of 7)
Figure 12.3-10

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 DAY AFTER ACCIDENT
(Sheet 1 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 DAY AFTER ACCIDENT
(Sheet 2 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 DAY AFTER ACCIDENT
(Sheet 3 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 DAY AFTER ACCIDENT
(Sheet 4 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 DAY AFTER ACCIDENT
(Sheet 5 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 DAY AFTER ACCIDENT
(Sheet 6 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 DAY AFTER ACCIDENT
(Sheet 7 of 7)
Figure 12.3-11

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 WEEK AFTER ACCIDENT
(Sheet 1 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 WEEK AFTER ACCIDENT
(Sheet 2 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY
AUXILIARY/CONTAINMENT BUILDING -
1 WEEK AFTER ACCIDENT
(Sheet 3 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 WEEK AFTER ACCIDENT
(Sheet 4 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 WEEK AFTER ACCIDENT
(Sheet 5 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 WEEK AFTER ACCIDENT
(Sheet 6 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

RADIATION ZONES (ACCIDENT) - PRIMARY AUXILIARY/CONTAINMENT BUILDING - 1 WEEK AFTER ACCIDENT
(Sheet 7 of 7)
Figure 12.3-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

LOCATION OF AND ACCESS WAY
TO VITAL AREAS
(Sheet 1 of 6)

Figure 12.3-13

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

LOCATION OF AND ACCESS WAY
TO VITAL AREAS
(Sheet 2 of 6)

Figure 12.3-13

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

LOCATION OF AND ACCESS WAY
TO VITAL AREAS
(Sheet 3 of 6)

Figure 12.3-13

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

LOCATION OF AND ACCESS WAY
TO VITAL AREAS
(Sheet 4 of 6)

Figure 12.3-13

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

LOCATION OF AND ACCESS WAY
TO VITAL AREAS
(Sheet 5 of 6)

Figure 12.3-13

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

LOCATION OF AND ACCESS WAY
TO VITAL AREAS
(Sheet 6 of 6)

Figure 12.3-13

KOREA HYDRO & NUCLEAR POWER CO., LTD.
HANBIT 3&4
FSAR

RADIATION ZONES (NORMAL) –
HANBIT 3&4 TEMPORARY STORAGE EL. 0.00 m
FIGURE 12.3-14

12.4 DOSE ASSESSMENT

12.4.1 Estimated Occupancy of Plant Radiation Zones

It is difficult to predict the average occupancy of any one zone by week, much less by function. To estimate the radiation exposure to operating personnel, the assumed occupancy of plant radiation zones given in Table 12.4-1 were used.

12.4.2 Estimates of Inhalation Doses

Small airborne radioactivity concentrations of radionuclides are expected within the various plant structures. Implementation of the health physics program minimizes any significant inhalation doses to plant personnel.

Because radionuclides may be present and released in significant quantities from fluids, maximum design-basis radioiodine concentrations were calculated in the buildings most susceptible to airborne contamination. The model and assumptions used in calculating these airborne radioactivity concentrations are presented in Subsection 12.2.3. The resultant design-basis concentrations and the thyroid dose acquired from their inhalation were calculated with the design-basis concentration.

12.4.3 Objectives and Criteria for Design Dose Rates

The objectives and criteria for design dose rates in various areas are discussed in Section 12.3.

12.4.4 Estimated Annual Occupational Exposures

Many ALARA features, such as the selection of materials to reduce crud levels and the separation of radiation areas, were incorporated into the design of

YGN 3&4. The actual dose rates experienced in most areas are expected to be lower than the respective design dose rates. Thus, a calculation of occupational exposures based on design dose rates overestimates these exposures in most cases. Both actual and design dose rate calculations require a knowledge of occupancy factors.

The most reliable prediction of occupational exposures is based on data from the operating history of PWRs having a CE NSSS. Tables 12.4-2 and 12.4-3 show the total annual man-Sv reported for CE PWRs in recent years. From Tables 12.4-2 and 12.4-3, an average annual dose is estimated to be approximately 3.31 man-Sv/unit-yr. 446

Table 12.4-4 shows the distribution of personnel and man-Sv doses according to work function for CE PWRs between the years 1984 and 1987.

For YGN 3&4, dose due to each work function is expected to be reduced to the values of Table 12.4-5 (Reference 1), since the following design and equipment improvements were applied:

a. Establishment of permanent working platforms for maintenance and repair of steam generator and reactor coolant pump

b. Improvement of design criteria for mechanical reach rod installation

c. Use of multiple stud tensioners for reactor vessel head and steam generator manway cover opening/closing

d. Use of WEPA equipment for refueling cavity decontamination

e. Use of automatic equipment for steam generator inspection and repair (e.g., ETC, tube plugging), etc.

Table 12.4-1 gives the estimated occupancy times in plant radiation areas and gamma doses to plant personnel on the basis of the expected average dose rate in each radiation zone shown in Table 12.4-6.

The annual doses to plant personnel are expected to be lower than the above estimated doses because YGN 3&4 will be operated under an aggressive KHNP operations and maintenance ALARA plan. 446

12.4.5 Estimated Annual Dose at the Exclusion Area Boundary (EAB)

12.4.5.1 Direct Radiation Dose Estimates

The direct radiation from the containment, auxiliary, radwaste, and turbine buildings is negligible compared with that from outside storage tanks. The principal source of radioactivity not stored in plant structures is from the holdup tank.

This tank is expected to contain concentrations of radionuclides that yield a dose rate at the tank shielded surface of 0.005 mSv/hr or less. The estimated annual direct dose at the EAB, based on 8760 hours of occupancy, is 0.002 mSv/unit. 446

12.4.5.2 Dose Estimates Due to Airborne Radioactivity

The estimated dose at the site boundary due to released activity is given in Subsection 11.3.3.

12.4.6 Estimated Annual Dose to Construction Workers

The estimated annual dose received by construction workers on YGN 3&4 due to the operation of YGN 1&2, and that of construction workers on YGN 4 due to the operation of YGN 1, 2, and 3 is within the limits of Korean AEA and related regulations for exposure to individuals in unrestricted areas.

12.4.6.1 Direct Radiation Dose Estimates

The estimated dose rates are the sum of the direct radiation from the operating units' containment, auxiliary building, turbine building, radwaste building, and the outside storage tanks. The annual individual dose and the total man-Sv dose were calculated by using occupancy times based on the projected construction manpower requirements and the following assumptions: 446

a. Total construction period is 6 years and YGN 3 is in operation during the last year of YGN 4 construction.

b. During the first year of construction, 100% of the construction manhours are spent outside. During years 2, 3, and 4, 50% of the construction manhours are spent outside. During the last year of construction, 20% of the construction manhours are spent outside of structures.

c. Construction workers are assumed to be on site for 2,500 hr/yr (50 hr/week for 50 week/yr).

d. The number of construction workers employed annually for each unit is as listed in Table 12.4-7.

e. The distances from the YGN 2 containment surface to YGN 3 workers and from the YGN 3 containment surface to YGN 4 workers are 236 m and 236m, respectively.

f. Once the YGN 4 major plant buildings and structures are completed, workers inside them or behind them with respect to YGN 1, 2, and 3 sources receive negligible direct radiation because of the shielding provided by the concrete and steel in those structures.

Based on the above assumptions, the estimated annual direct dose per each construction worker during construction of YGN 3&4 from YGN 2 and the annual direct dose for each YGN 4 construction worker during the last year of YGN 4 construction is given in Table 12.4-8.

The estimated annual dose (man-Sv) to construction workers of YGN 3&4 during 446
construction is given in Table 12.4-9.

12.4.6.2 Exposures due to Airborne Radioactivity

12.4.7 References

1. "Design Improvement Studies on the Standardization of Nuclear Power Plants," Vol.17, KOPEC, 1987.8.

2. B. G. Brooks, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1984," NUREG-0713 Vol. 6, U.S. NRC, 1987.

3. B. G. Brooks, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1985," NUREG-0713 Vol. 7, U.S. NRC, 1988.

4. B. G. Brooks, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1986," NUREG-0713 Vol. 8, U.S. NRC, 1989.

5. B.G. Brooks, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1987," NUREG-0713 Vol. 9, U.S. NRC, 1990.

6. "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I", Regulatory Guide 1.109, U.S. NRC, Rev.1, 1977.

TABLE 12.4-1 (Sh. 1 of 3)

ESTIMATED OCCUPANCY TIMES IN PLANT RADIATION AREAS
AND GAMMA DOSES TO PLANT PERSONNEL

TABLE 12.4-1 (Sh. 2 of 3)

TABLE 12.4-1 (Sh. 3 of 3)

TABLE 12.4-2

OCCUPATIONAL RADIATION EXPOSURE DATA
FROM OPERATING CE PWR PLANTS

TABLE 12.4-3

YEARLY AVERAGES AND GRAND AVERAGES FOR NUMBER OF PERSONNEL AND MAN-SV DOSES FOR SELECTED PLANTS

TABLE 12.4-4

DISTRIBUTION OF PERSONNEL AND MAN-SV DOSES FOR PLANTS

TABLE 12.4-5

DISTRIBUTION OF MAN-SV DOSES OF YGN 3&4

TABLE 12.4-6

MAXIMUM AND EXPECTED AVERAGE DOSE RATES
IN THE PLANT RADIATION ZONES

TABLE 12.4-7

ESTIMATED NUMBER OF CONSTRUCTION

WORKERS AT THE YGN 3&4 SITE

TABLE 12.4-8

ESTIMATED ANNUAL DOSE PER WORKER

TABLE 12.4-9

ESTIMATED WHOLE-BODY MAN-SV DOSES
OF CONSTRUCTION WORKERS

12.5 HEALTH PHYSICS PROGRAM

12.5.1 Organization

12.5.1.1 Program Organization

The site health physics program applies to all four units, i.e., YGN 1, 2, 3 & 4. The YGN 3&4 organization is given in Section 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 ensuring that the plant operation meets the radiation protection requirements of the Korean Atomic Energy Act (AEA) and related regulations that are applicable to the health physics program. The corporate 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 report to the manager, Radiological Control Section, are experts in implementing the radiation protection program. They establish and control the radiation work procedure to keep the radiation exposure ALARA.

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

A more detailed discussion of the responsibilities and authority of the supervisory positions mentioned above and the required training and qualifications for the personnel presently holding these positions is presented in Subsections 13.1.2 and 13.1.3, respectively.

12.5.1.2 Program Objective

The radiation protection program has these objectives:

a. To provide administrative control of persons on the site to ensure that personnel exposure to radiation and radioactive materials is within the guidelines of NSSC Notice 2014-34(방사선방호 등에 관한 기준) and that such exposure is kept ALARA. 735

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

Limits are developed consistent with NSSC Notice 2014-34(방사선방호 등에 관한 기준) 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. 735

b. To provide administrative control over any plant effluent releases to ensure that these releases are below the limits of NSSC Notice 2014-34 (방사선방호 등에 관한 기준). 735

12.5.1.3 Radiation Protection Program

The plant radiation protection program will be officially initiated at YGN 3 and later at YGN 4, when radioactive material licensed to Yonggwang is first brought into the respective unit, 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 consistent with the recommendations of Korea AEA and related Regulations.

The radiation protection program ensures that the following tasks are performed:

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 radiation exposure.

d. Radiation areas are segregated and appropriately posted to limit internal uptake of radioactive materials.

e. Instruments and equipment are 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 while safely supplying a reliable source of power to the public.

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

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

12.5.2 Equipment, Instrumentation, and Facilities

12.5.2.1 Controlled Area

The station design establishes a controlled area for each unit. 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 stated in Table 12.3-1. The radiation zoning planned for YGN 3&4 are shown in the radiation zone drawings, Figures 12.3-3 through 12.3-7.

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 access control building (see Figure 12.5-1) and will be controlled by the radiation protection personnel. All other potential access points to the controlled area are kept locked or sealed. Temporary controlled access areas may be established in the clean area of the plant and are 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 access requirements may be extended to include portions of the backyard area (i.e., the restricted area) to allow access for trucks to the fuel and radwaste buildings. The exclusion area boundary is maintained by fencing, rope, or other continuous barricades. The backyard areas are not used to store contaminated materials. Personnel requiring entry into the restricted area, such as truck drivers, are provided with protective clothing and continually escorted. Radiation dosimetry devices are provided to such personnel if they require access into a 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 that 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 that prevents unauthorized access. Personnel entering high-radiation areas receive radiation protection instructions from a health physicist when the RWP is issued.

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 monitors.

c. A counting room(s) where radioactive samples are analyzed for isotopic composition and activity levels. It is shielded and air-conditioned to reduce background levels and data fluctuations. Additional design considerations are given in Subsection 12.3.1.3.6.

d. A storage area for bulk quantities of chemicals and laboratory supplies.

e. Health physics (radiation protection) office and access control check-point.

f. A personnel locker and changing room.

g. Personnel decontamination facility.

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 before leaving the area. The access control station includes portal monitors and frisking probes.

k. An equipment and instrument decontamination facility (or facilities) with equipment handling facilities and a hot instrument shop to provide an area for repair and maintenance of contaminated instruments.

12.5.2.3 Radiation Protection Design

Access to areas with the potential for continuous radiation levels in excess of 1 mSv/hr is restricted by a lockable door, constructed to allow rapid 446 entry or exit in case of an emergency. Special materials, such as lead brick or sheet, are provided to shield 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). Nonroutine handling of radioactive materials are planned on a case-by-case basis, and special shielding or tools are utilized to the extent practical to limit radiation exposure to personnel. The above items are covered in more detail in Subsections 12.3.1 and 12.3.2.

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

b. Liquid scintillation counter

c. Alpha counter

d. Beta counter

e. Hyper-pure germanium multichannel analyzer

f. NaI (Tl) multichannel analyzer

g. Two-pi gas flow counter

h. Pulse generator

I. Air flow calibrator

j. Gamma source calibrator

k. Source isotope calibrator

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 includes the following:

a. Wide range beta-gamma survey meter
b. Medium range beta-gamma survey meter
c. Low range beta-gamma survey meter
d. Alpha survey meter
e. Beta survey meter
f. Neutron survey meter
g. Portable surface contamination monitor
h. Beta particulate air monitor

12.5.2.4.3 Portable Air Sampling Instrumentation

The portable air sampling instrumentation includes the following:

a. Battery operating air sampler
b. High volume air sampler
c. Regulated air sampler

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 before entry by operations or maintenance personnel. The continuous air monitors (CAMs) 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 Exposure Monitoring Instrumentation

The radiation monitoring instrumentation includes the following:

a. Neutron pocket dosimeters
b. Auto dose monitoring system
c. Thermoluminescent dosimeters (TLD)
d. Portal monitor
e. Whole body counter
f. Thyroid monitor

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 to ensure their proper functioning. Portable emergency instrumentation includes the following:

a. A wide-range GM survey meter
b. A low-level contamination detection instrument
c. A portable air sampler
d. Self-reading dosimeters
e. Various respirators

12.5.2.4.6 Calibration of Radiation Protection Instrumentation

Radiation protection instruments are tested and calibrated semiannually and after each repair by a radiation protection technician. 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 and calibration are performed by trained instrument and control technicians using the proper calibration facility and approved procedures. Radiation protection instruments include the following. 165

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, auxiliary 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 decontaminating 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 decontaminating the outside of drums containing solidified waste. In the hot machine shop, a centralized decontamination facility is provided for decontaminating tools and equipment. The central decontamination facility contains spray, ultrasonic bath, and chemical bath. The decontamination facility in the auxiliary building is used for washing down items before they are transported to the radwaste or access control building for maintenance.

Typically, process system components are decontaminated in place before maintenance or equipment removal. Each process component, e.g., pumps, heat exchangers, and filters containing a significant quantity of radioactive fluid, is provided with flush and drain connections. These components will be flushed in place before maintenance. If necessary, a portable pump and drum of decontamination chemicals may be used to circulate the decontaminant through the component using the flush and drain connections. Components requiring further decontamination will be processed through the auxiliary building facility or enclosed in poly bags and transported to the central decontamination facility.

Decontamination of large components is treated case by case because of the varied configurations and locations of equipment. Where necessary, temporary curtains are placed around components requiring in-place decontamination. Components in potentially radioactive areas are coated with decontaminable coatings to facilitate cleanup. If necessary, solvents or sandblasting will be employed to completely remove coatings for maximum decontamination.

12.5.3 Procedures

12.5.3.1 Radiation and Contamination Surveys

12.5.3.1.1 Policy

The procedures for radiation and contamination surveys will be established before the initial core loading. These procedures will specify the conditions, requirements, and area for the routine radiation survey and nonroutine special survey. The purpose of the survey will be to collect and to check the radiation and/or contamination level of various plant areas. 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 conduct routine surveys and keep records. The plant health physicist reviews such survey results and identifies appropriate actions. The manager, Radiation Management Section, has 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 are conducted as needed. Certain radiation conditions in the working area may require continuous survey during the maintenance work.

b. Contamination

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

c. Air

An airborne radioactivity survey, which is an evaluation of the concentration of airborne radioactivity present in any area, is made periodically 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

Procedures for access control to radiation areas or potential radiation areas are developed by the Radiation Management Section staff, and all radiation workers are well acquainted with them. The radiation exposure of individual workers is carefully investigated and logged to ensure that radiation exposure is as low as reasonably achievable. The plant health physicist establishes the ALARA program.

12.5.3.3 Access and Stay Time Control

12.5.3.3.1 General

Persons not thoroughly familiar with the controlled area procedures are escorted by health physics technicians or are instructed with proper protection guidelines to ensure adequate radiation protection. Upon termination of work within the controlled area, workers leave immediately. Certain controlled areas will be 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 levels, of the protective clothing requirements, and of other requirements to perform their job safely.

12.5.3.3.2 Entry into the Controlled Area

The following are requirements for entry into a radiological control area:

a. TLD badges, dosimeters, protective clothing and other required personal monitors are worn as specified by the appropriate RWP.

b. Entry into the controlled area with open wounds is prohibited. All open wounds are sealed with a waterproof bandage before 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 is normally through the area or cubicle access control point. Entrance to the controlled area via any other route must be authorized by the plant manager, or his designee, and included in the RWP.

12.5.3.3.3 Exit from the Controlled Area

The following procedure will be followed upon leaving 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 before leaving the area.

e. Exit during an emergency condition is in accordance with the station's emergency plan.

12.5.3.4 Contamination Control

12.5.3.4.1 Facility Contamination Control

Contamination of general plant areas 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 before entering another area of the plant. The final checkpoint for all personnel leaving all restricted areas of the plant is the main 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 using several types of protective clothing when entering contaminated areas.

a. A lab coat is worn by laboratory personnel during radioactive sample analysis.

b. A 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.

e. Plastic suits are worn over cloth coveralls in areas where the potential exists for liquid contamination of personnel.

f. A cloth cap is worn for low-level dry contamination, cloth hoods are worn for high level dry contamination, and plastic hoods are worn 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's general areas are accessible to personnel without protective clothes. As a result, and to minimize the area in which protective clothing is required, temporary change areas will be set up adjacent to the work areas for special maintenance jobs. Also, permanent change areas will be established for areas routinely requiring protective clothing. If, at any time, the number of 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 the main access control point shall become the main change area for the entire restricted part of the plant.

12.5.3.5 Airborne Activity Control

The proper respiratory equipment as specified by the health physicist staff on the RWP is required before entering the airborne contamination area. It is the responsibility of the health physics group to survey the area and to determine 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., when 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 are required to prevent internal exposure in an 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 this procedure shall mean thermoluminescent dosimetry and pocket dosimeters. All radiation workers will be issued a TLD monitor, and they must wear this monitoring device as specified while within the radiation controlled area.

12.5.3.6.2 Plant Personnel Exposure

12.5.3.6.2.1 External Dosimetry

Thermoluminescent dosimeter (TLD) badges will be issued to all personnel entering the controlled area of the plant. Each badge has its own identification number, and badge is assigned to each person. Badges must be worn at all times while within the controlled area and are to be placed in a designated badge rack.

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

Film badges are maintained in reserve by the health physicist staff for use by personnel where conditions of the job necessitate this, and they serve as a backup in situations where there is a lack of TLD badges.

All personnel having occasion to enter the controlled area or anyone else designated by the manager, Radiation Management Section, or his designated alternate, is issued a self-reading pocket dosimeter. This is identified and stored in a manner similar to the TLD badge. All regularly assigned self-reading 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 starting work. All other personnel who 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 areas of the plant will be evaluated by urinalysis. Urinalysis is performed, as a minimum, once per year on all plant personnel or as deemed necessary by the manager, Radiation Management Section.

The whole-body counting system is also used for the internal dose assessment. Plant procedures will require all radiation workers to take a whole-body scan at least once a year. Special nonroutine scanning may be requested by the plant health physicist whenever significant internal radiation exposure is suspected. Any overexposure will be reported to the Plant Nuclear Safety

Committee (PNSC), who will initiate a further detailed investigation.

12.5.3.7 Radioactive Materials Safety Program

The storage, handling, transportation, and disposal of radioactive materials will be described in the plant procedure, which ensures compliance with all applicable regulations so that personnel are not exposed unnecessarily to radiation.

12.5.3.7.1 Receiving Radioactive Material

Whenever an order for radioactive material is initiated by 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. Shipments 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 boundary, the plant health physicist technicians will conduct an external radiation survey, then the package will be placed in a suitable location and properly labeled. The manager, Radiation Management Section, is responsible 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 the health physicist. Only the radioactive material package with the permissible radiation and contamination levels will be permitted to be transported into the site area.

12.5.3.7.2 Storing Radioactive Material

All radioactive materials are stored in restricted area(s) as designated by the Health Physics Group. All radioactive materials entering and leaving a radioactive materials storage 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 for radioactive sources complies with the Korean Atomic Energy Act (AEA) & Enforcement regulations 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 will not be 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.

Whenever the possibility of personnel exposure is deemed as significant, all radioactive material will be tagged and/or labeled properly on the container before transfer.

12.5.3.7.4 Fuel Handling, Storage, and Shipment

The receipt, inventory (including location), disposal, and transfer of all fuel, new and spent, will be in accordance with the Korean Atomic Energy Act (AEA) & Enforcement regulations.

The Health Physics Group is responsible for surveys, both radiation and contamination, of all fuel before and during unpacking and storage. Surveys of all shipping containers 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. The training program is designed to cover essential station operation and radiation protection subject matter in the depth required for various technical positions. 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, Radiation Management Section, is responsible for the radiation protection training of YGN 3&4 employees and other individuals assigned to the station. It will be his responsibility to ensure that all personnel assigned to YGN 3&4 and personnel working with radioactive materials are adequately trained. A record will be kept of all individuals trained.

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

ACCESS CONTROL POINT FOR
INGRESS AND EGRESS FROM
THE CONTROLLED AREA

Figure 12.5-1