CHAPTER 12- RADIATION PROTECTION

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

12.1.1 POLICY CONSIDERATIONS

12.1.1.1 Management Policy

It is the licensee's policy to maintain occupational radiation exposure ALARA at LGS. This includes maintaining the annual dose to individuals working at the station ALARA, and keeping the annual integrated dose to station personnel ALARA. The management of this company is firmly committed to performing all reasonable actions to ensure that radiation exposures are maintained ALARA.

Sections 12.1.2 and 12.3 discuss the ALARA considerations that have been incorporated into the design of the LGS.

LGS will be operated and maintained in such a manner as to ensure that occupational radiation exposures are ALARA and that protection against radiation is in accordance with 10CFR20. The health physics provisions which accomplish this goal are described in Section 12.5.

12.1.1.2 Management Responsibilities

Chapter 13 describes the Limerick Generating Station operating organization and the licensee's corporate organization.

The President and Chief Nuclear Officer has the ultimate responsibility for operation of the Limerick Generating Station. This responsibility includes those activities necessary to ensure that radiation exposures are maintained ALARA. Responsibility is delegated to the Senior Vice President - Nuclear Operations and to the Vice President - Station Support (for all activities performed by the Station Support Department), and to the Vice President - Limerick Generating Station for all activities performed at the station. The Station Support Department provides technical support as needed to the Limerick Generating Station health physics organization in an effort to maintain the highest standards of exposure minimization. The Vice President - Limerick Generating Station delegates responsibility for the ALARA program through the Plant Manager to the Radiation Protection Manager who has responsibility for all actions required to maintain station exposures ALARA. The responsibilities of the Station Support Division staff, the Plant Manager, and Radiation Protection Manager in regard to ALARA are further described in Section 13.1.2.1.2. These responsibilities correspond to the applicable functions described in Regulatory Guide 8.8, Rev. 3 and Regulatory Guide 8.10, Rev.1.

12.1.1.3 Policy Implementation

The licensee provides the environment and support for the ALARA policy to function. The company's commitment to this policy is manifested in the plant design, established procedures, the provisions for review of procedures and plant design, provisions for subsequent procedure revisions and plant modifications, and the establishment of extensive varietal training programs.

General Employee Training, which encompasses radiological health aspects, is described in Sections 13.2.1.2. Such training enhances personnel awareness of licensee management's policy of maintaining exposures ALARA, of actual and potential problems, and of the need to develop proper attitudes. The training stresses the individual's responsibility to cooperate in maintaining exposures ALARA and the importance of adhering to approved procedures. The content of the training is adjusted in recognition of the duties, responsibilities, and anticipated radiation exposure of those receiving instruction and includes information on the biological effects of such exposure. The staff of the Radiation Protection Manager has well defined functions, responsibilities, and authorities to ensure proper supervision and implementation of health physics procedures. The Radiation Protection Manager has the authority to prevent unsafe practices and to direct steps to prevent any unnecessary radiation exposures. The plant staff health physics group assures that communication with supervisors of the various station service groups (e.g., maintenance, construction, instrument technicians, contractors) and operating supervisors occurs for the purpose of evaluating a course of action regarding ALARA for specific station activities. Such evaluation includes the review of appropriate procedures; the extent of existing or potential hazards, occupationally and to the general public; and the merits of applying special techniques to the performance of a job. Such communication with work forces is an effective means to respond to worker questions and concerns and to obtain information on actual working conditions, such as mobility, access, habitability, and necessary tooling, which can lead to future improvements.

The plant staff health physics group is knowledgeable of the origins of radiation exposure in the plant and its magnitudes. They recognize which jobs or locations cause the highest exposures. This information is obtained via area surveys, Radiation Work Permits (or equivalent), and personnel dosimetry. Analysis of these and other data for repeat or similar activities is performed to determine whether exposures are being decreased or at least prevented from increasing. There is prompt investigation of exposures of record which exceed expected values. Judicious application of dose extensions is exercised by review of prior data and analysis of the need for the exposure. The health physics procedures provide for appropriate documentation of reviews, surveys, analyses, and investigations such that corrective action or modification may be accomplished and subsequent data may be compared to the original data to verify effectiveness of the change. The need for modification to satisfy ALARA shall be based on consideration of the economics of equipment modification in relation to benefits to health and safety and other societal and socioeconomic considerations, including the utilization of atomic energy in the public interest.

To verify that the health physics operations at the station are functioning within the ALARA concept, a formal review shall be performed under the cognizance of the Station ALARA Council every three years (based upon the date of commercial operation of Unit 1). The review shall include review of applicable station procedures and practices, exposure records, the content of training programs which affect ALARA considerations, and consultation with the plant staff health physics group. The objective of the review is to evaluate the adequacy of the ALARA effort and, as appropriate, to determine means to lower exposures. The results of the review shall be documented, including identification of the procedures and records reviewed, the review team's evaluation, and any recommendations for improvements.

12.1.2 DESIGN CONSIDERATIONS

This section describes those considerations which are applied to the plant design for the purpose of incorporating features which provide for maintaining occupational radiation exposures ALARA.

Refer to Sections 12.3.1, 12.3.2, and 12.3.3 for details of design for maintaining personnel radiation exposures ALARA.

Experiences and data from operating plants are evaluated to decide if and how equipment or facility designs can be improved to reduce overall plant personnel exposures. During plant design, operating reports and data such as that given in WASH-1311, NUREG-75/032, NUREG-0109 and Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants (References 12.1-1 through 12.1-4, respectively), are reviewed to determine which operations, procedures or types of equipment were most significant in producing personnel exposures. Methods to mitigate such exposures have been implemented wherever possible and practicable.

12.1.2.1 General Design Considerations

The objectives of the plant design for ALARA purposes are: to minimize the need for, and duration of, personnel access into high radiation areas, and to establish radiation levels as low as practicable in routinely occupied areas.

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

In addition to equipment and facility designs, system designs are considered to ensure that exposures are maintained ALARA. For example, both the primary coolant system and the condensate system are provided with cleanup capability to reduce the inventory of circulating corrosion products. This is one of the methods employed to minimize both the activation of these corrosion products and their subsequent deposition on the interior surfaces of piping and equipment.

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

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

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

12.1.2.2 Equipment Design Considerations

Considerations for equipment design include:

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

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

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

d. Redundant equipment to reduce the urgency for immediate repairs, thus providing time for planning repairs with ALARA in mind.

e. Isolation, draining, flushing, or decontamination of systems to reduce crud deposition and thus reduce radiation levels.

f. Isolation of components from contaminated process fluids.

g. Use of high quality components, such as valves, which minimize or preclude the leakage of radioactive material.

h. Use of closed drain systems for contaminated process fluids to preclude the creation of airborne contamination that is due to spillage.

12.1.2.3 Facility Design Considerations

Considerations for facility design include:

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

b. Use of valve stem extensions, articulated if necessary, to operate valves from behind shield walls.

c. Transport of contaminated components for service in lower radiation areas or for reuse in other parts of the plant.

d. Separation of sources of radiation such as pipe runs, storage tanks, and filters from normally occupied areas.

e. Use of permanent shielding between sources of radiation and access and service areas.

f. Maintaining ventilation flow paths from clean areas to contaminated areas.

g. Use of surface coatings to facilitate decontamination.

h. Use of labyrinth entrances to shielded cavities.

12.1.2.4 ALARA Design Review

For the operating plant, procedures provide for review of subsequent plant design and modifications by the radiation protection group, when applicable, and documentation of an ALARA design review. Procedures are described in Section 12.1.3. The following material describes the ALARA design review for the initial plant design and construction and is historical in nature.

Bechtel, as agent for the licensee, has developed an "In-Plant Radiation Exposure ALARA Review Specification". This specification was applied during all construction phases to perform multidisciplinary reviews of the plant to assure a design that:

a. Maintains the annual operating and maintenance radiation doses to individuals ALARA

b. Keeps the annual integrated radiation doses to all personnel ALARA.

This specification defined the purpose of the review; established the project ALARA review team; and described the discipline ALARA review process, the extent and format of ALARA review meetings, and the method of noting and resolving ALARA design changes. This specification is consistent with design guidelines given in Regulatory Guide 8.8.

The Bechtel Project Mechanical/Nuclear Group was responsible for the overall coordination of the ALARA design review and the interaction between the various Bechtel project disciplines, Bechtel Construction, the Bechtel Staff Radiation Protection Group, and the licensee.

The groups involved in the ALARA review program and their relationships are shown on Figure 12.1-1.

An experienced engineer from the Mechanical/Nuclear Group served as the project ALARA coordinator and was responsible for overall coordination of the ALARA reviews. Duties included maintaining the ALARA Review Specification and the ALARA documentation file, expediting the ALARA reviews, and resolving any ALARA inconsistencies. The project ALARA coordinator obtained input and expertise from the licensee, the Radiation Protection Group, and other Bechtel groups, as required.

An engineer was assigned from each engineering discipline to serve as the discipline ALARA representative. The discipline ALARA representatives coordinated and conducted the ALARA reviews within the discipline, interfaced between the discipline and the project ALARA coordinator, and expedited any design changes that resulted from the ALARA review.

Engineers from the field were assigned as construction ALARA representatives. They were responsible for reviewing field designs and field modifications to existing designs for consistency with the ALARA Review Specification. The construction ALARA coordinator coordinated the ALARA reviews in the field and interfaced between construction and engineering.

Experienced radiation protection specialists from the staff of the chief nuclear engineer provided health physics and radiation protection design input to the project. A radiation protection specialist was assigned to support the project design and ALARA review and was responsible for interfacing between the group and the project ALARA coordinator. The radiation protection specialist represented the staff radiation protection group during the ALARA reviews.

The licensee’s ALARA coordinator was a licensee engineer who represented the licensee during the ALARA reviews. One of the responsibilities was to advise the project ALARA coordinator of potential ALARA problems based on the licensee's experience from operating plants, by receiving input from the licensee’s Health Physics and Operations groups. The licensee’s ALARA coordinator also reviewed and provided input to the ALARA Review Specification, reviewed

exceptions to the ALARA design criteria, and provided licensee input for resolving ALARA inconsistencies that were identified during the ALARA review.

To facilitate documentation and coordination of the ALARA review effort, a scheduled system and activity design review approach was followed. Maximum review effort was expended on those systems and activities as identified in Table 12.1-1 that in the past had resulted in the highest occupational radiation exposures in operating plants. Each system and activity was divided into plant areas. The ALARA design review was conducted by reviewing each of these areas.

The design review was conducted in the home office by the discipline ALARA representatives, project ALARA coordinator, and the radiation protection specialist. The major tools used in the design review were the ALARA design review considerations checklist, which identified design features that were judged to be generally cost effective with respect to maintaining occupational radiation exposures ALARA, and the ALARA design review considerations sign-off sheet. The sign-off sheet documented the review of the design by the responsible discipline(s). The discipline ALARA representatives and the radiation protection specialist verified that the design considerations assigned to their groups were met.

The ALARA review of field designs and modifications was conducted by the construction ALARA representatives. The discipline responsible for the new design or modification was responsible for documenting that it had been reviewed for ALARA. The project ALARA coordinator coordinated any reviews that affected other disciplines or groups.

Compliance with the ALARA design review considerations was documented by the reviewer on the design review considerations sign-off sheets, which are kept on file. Any exceptions to the considerations were reviewed on a case-by-case basis.

12.1.2.5 ALARA Design Considerations for Decommissioning

The design features necessary to maintain radiation exposures ALARA during decommissioning operations are, in general, the features that have been implemented to keep exposures ALARA during the operational life of the plant. These are discussed in Sections 12.1.2.1, 12.1.2.2, 12.1.2.3, and 12.3. Some of these features that are especially applicable to decommissioning are discussed below.

a. Flushing and draining connections will provide for removal of radioactive fluids, allow rinsing to reduce residual activity, and provide an entry point for introduction of decontamination solutions.

b. Ventilation systems to minimize the spread of airborne radioactivity will be particularly useful in preventing exposures to internal radioactivity during decommissioning when large quantities of airborne radioactive particulates can be generated by cutting, sawing, and demolition.

c. The space envelopes reserved around equipment to facilitate maintenance will also allow for more rapid dismantling because cutting machines and other decommissioning equipment can be quickly installed with correspondingly lower exposure time.

d. The use of flanged connections on pumps in radioactive waste systems, and the removal provisions built into some major plant components, will reduce exposures to personnel during removal of these items.

e. Separation of radioactive from nonradioactive systems, and location of active components of nonradioactive systems in low radiation areas, will permit dismantling of normally clean systems with minimal exposure to personnel.

f. The availability of a complete, shielded radwaste facility will allow efficient low dose processing of residual fluids and decontamination solutions, as well as the packaging and shipping of solid radioactive materials. Other existing facilities, such as access control stations and decontamination areas, will also perform their intended functions in helping to keep exposures ALARA.

g. The use of liners and protective coatings will lower exposures by minimizing decontamination times and by reducing the quantities of materials that must be handled as radioactive waste.

12.1.3 OPERATIONAL CONSIDERATIONS

This section describes the development and implementation of operating procedures, including procedures for radiation protection and the ALARA program.

Plant procedures are further described in Section 13.5. Health physics operations are described in Section 12.5.

12.1.3.1 Procedure Development

Various procedures are written for the different activities associated with plant operations. These include procedures for operating, maintenance, surveillance testing, fuel handling, emergencies, radiation protection, and administration. The development of each variety of procedure is described in the administrative procedures.

The different procedures are prepared by personnel having experience and expertise in that particular area. The experience gained from operation of the PBAPS units and other plants is incorporated into procedure development. Each procedure is reviewed and approved per Administrative procedure process. Health physics personnel review procedures for activities which can affect radiation exposure.

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

12.1.3.2 Exposure Reduction Procedures

The ALARA concept is first and foremost practiced on the job by communication between health physics technicians or health physics supervisors and the workers. The health physics representatives are aware of the radiation conditions and advise the workers accordingly.

Specific radiation procedures describe the techniques to determine the radiological conditions in an area. Once determined, procedures describe the actions necessary to incorporate exposure reduction techniques as required.

12.1.3.3 General ALARA Techniques

The predominant exposure dose is received during outages. Although ALARA considerations are not limited to outage work, the activities conducted during outages are the most significant for dose reduction.

During the outage planning stages, an ALARA representative is designated by Health Physics supervision to review the jobs planned and evaluate the need for an ALARA effort. Examples of some techniques that apply to an ALARA evaluation are:

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

b. Installing permanent or temporary shielding if the net result is reduction in man-rem.

c. Training workers to improve proficiency, thus reducing stay time in the radiation area.

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

e. Establishing control points in low radiation areas.

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

g. Using special tools for remote handling of components.

h. Planning and preparing techniques and tools needed to accomplish the job before the job is started.

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

j. Using remote monitoring/alarming dosimeters in high radiation areas to maintain close checks on personnel exposures.

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

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

m. Construction of contamination containment devices such as glove boxes and tents.

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

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

p. Establishing auxiliary ventilation/filtration systems.

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

r. Contingency planning to account for known personnel hazards or accidents which may occur.

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

The ALARA representative communicates with other disciplines (maintenance, operating, instrument and control, etc.) as appropriate to discuss the implementation of ALARA techniques. The cooperation of more than one discipline is usually required for most jobs.

The dosimetry program consists of DLRs or equivalent badges processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited contractor for official data of record. Additionally, direct reading dosimeters are used as the unofficial dosimetric device. The computerized dosimetry record system and the ability to process the unofficial dosimetric device onsite provide a powerful tool for maintaining surveillance and managing each worker's dose accumulation. The direct reading dosimeters provide immediate readings and are available to obtain estimated exposures whenever there is concern. Workers' assignments to radiation areas can then be limited if administrative guidelines or regulatory limits are approached.

This same system produces tabulated lists of workers' exposures-to-date for each supervisor. These lists are provided routinely so the supervisors can know the dose accumulation of their workers, observe trends, and assign work duties more efficiently. Changes to the dosimetry program, such as contractor service versus onsite processing, frequency of badge changes, and types of dosimetric devices are made periodically to enhance the value of the program. Annual exposure reviews will be performed by the Engineer - Health Physics and Chemistry in order to identify groups with the highest exposure.

12.1.3.4 Operating Experience

Experience gained during operation of PBAPS Units 2 and 3 serves as a basis for procedures, techniques, and administrative controls for LGS. Exposure data from specific jobs previously performed at PBAPS have caused design changes during construction of LGS. The LGS Hot Maintenance Decontamination Shop design incorporates lessons learned from PBAPS.

12.1.4 REFERENCES

12.1-1 T.D. Murphy, WASH-1311, UC-78, "A Compilation of Occupational Radiation Exposure from Light-Water-Cooled Nuclear Power Plants 1969-1973", NRC, Radiological Assessment Branch, (May 1974).

12.1-2 T.D. Murphy, et.al., NUREG-75/032, "Occupational Radiation Exposure at Light-Water-Cooled Power Reactors 1969-1974", NRC, Radiological Assessment Branch, (June 1975).

12.1-3 T.D. Murphy, et. al., NUREG-0109, "Occupational Radiation Exposure at Light-Water-Cooled Power Reactors 1968-1975", NRC, Radiological Assessment Branch, (August 1976).

12.1-4 C.A. Pelletier, et. al., National Environmental Studies Project, "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants", Atomic Industrial Forum, (September 1974).

Table 12.1-1

SYSTEMS AND ACTIVITIES INCLUDED IN ALARA REVIEW

1. Waste Management Systems
   a. Liquid Waste Management System
   b. Solid Waste Management System
   c. Gaseous Waste Management System
2. Reactor Water Cleanup System
3. Fuel Pool Cooling and Cleanup System
4. Condensate Demineralizer System
5. Reactor Coolant System
6. Residual Heat Removal System
7. Main Steam System
8. Air Removal System
9. Feedwater System
10. High Pressure Coolant Injection System
11. Reactor Core Isolation Cooling System
12. Maintenance Activities
    a. Drywell Area Activities
    b. Refueling Area Activities
    c. Main Condenser Area Activities
    d. Inservice Inspection Activities
    e. TIP Maintenance Activities
    f. CRD Removal and Maintenance Activities
    g. Local Leak Rate Testing Activities

Table 12.1-1 (Cont'd)

h. Hot Maintenance Shop Activities

i. Snubber Inspection Activities

13. Miscellaneous

a. Radiation Zone I and II Areas

12.2 RADIATION SOURCES

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

12.2.1 CONTAINED SOURCES

The shielding design source terms are based on a noble gas fission product offgas release rate of 0.35 Ci/sec (after 30 minutes of decay) and the corresponding fission, activation, and corrosion product concentrations in the primary coolant. The guidance provided in ANSI N237 was not used to determine the shielding design source terms for LGS. The specific alternate methods used for calculating source term magnitudes are described in Section 11.1. The shielding design source terms in the primary coolant are listed in Tables 12.2-1 through 12.2-5. Throughout most of the primary coolant system, activation products, principally nitrogen-16, are the primary radiation sources for shielding design.

Basic reactor data and core region descriptions used for this section are listed in Tables 12.2-6 through 12.2-8.

The shielding design source terms are presented by enclosure location and system. Locations of the equipment discussed in this section are shown on the shielding and radiation zoning drawings, drawings N-110, N-111, N-112, N-113, N-115, N-116, N-117, N-118, N-119, N-120, N-121, N-122, N-124, N-125, N-126, N-127, N-128, N-130 N-131, N-132, N-133, N-134, N-135, N-136, N-137, N-140, N-141, N-142, and N-143. Detailed data on source descriptions for each shielded area are presented in Tables 12.2-11 through 12.2-13.

Shielding source terms presented in this section and associated tables are based on conservative assumptions about system and equipment operations and characteristics to provide reasonably conservative radioactivity concentrations for shielding design. For all systems transporting radioactive materials, conservative allowance is made for transit decay while at the same time providing for daughter product formation. Assumptions from NUREG-0016 (Reference 12.2-1) were also used where applicable. Therefore, the shielding source terms are not intended to approximate the actual system design radioactivity concentrations.

12.2.1.1 Primary Containment

12.2.1.1.1 Reactor Core

The radiation within the drywell during full power operation includes neutron and gamma radiation resulting from the fission process in the core. Tables 12.2-9 and 12.2-10 list the multigroup neutron and gamma ray fluxes at the outside surfaces of the RPV and the primary shield at the core midplane. The gamma fluxes include those resulting from capture or inelastic scattering of neutrons within the RPV, the core shroud, and the primary shield, as well as gamma radiation resulting from prompt fission and fission product decay. The largest radiation sources after reactor shutdown are the decaying fission products in the fuel. Table 12.2-14 lists the fuel assembly source terms. Secondary sources include the structural material activation of the RPV, its internals, and the piping and equipment located in the primary containment; and the activated corrosion products accumulated or deposited in the internals of the RPV, the primary coolant piping, and other process system piping in the primary containment.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the RCS used for shielding design are fission products estimated to be released from fuel and activation and corrosion products that are circulated in the reactor coolant. These sources are listed in Tables 12.2-1 through 12.2-5. The N-16 concentration in the reactor coolant, with HWC, is bounded at 4.8×10^{-5} Ci/gm of coolant at the reactor recirculation outlet nozzle.

12.2.1.1.3 Main Steam System

Radiation sources in the main steam system piping include activation gases, principally N-16, and the corrosion and fission products carried over to the main steam system.

The N-16 concentration without HWC in the steam is assumed to be 5.0×10^{-5} Ci/gm of steam leaving the reactor vessel at the nozzle. Fission product radioactivity corresponds to an offgas release rate of 0.35 Ci/sec at 30 minutes decay from the reactor nozzle. Partition factors for carryover of radioactivity into the main steam system are 100% for gases, 2% by weight for halogens, and 0.1% by weight for particulates. These partition factors are applied to the reactor water shielding source terms as given in Tables 12.2-1 through 12.2-5.

Hydrogen injection by the HWC system causes the reactor water chemistry to become less oxidizing which results in a re-distribution of the N-16 normally produced by radiolysis in the reactor core. Under HWC conditions more of the N-16 is carried over into the steam and less remains in the reactor water. Under HWC conditions the N-16 concentration in the steam is 2.50×10^{-4} Ci/gm.

12.2.1.1.4 Drywell Sumps

The concentrations of radioisotopes used for shielding design for the drywell equipment and floor drain sumps are listed in Table 12.2-15.

12.2.1.2 Reactor Enclosure and Refueling Area

12.2.1.2.1 Reactor Water Cleanup System

Radiation sources in the RWCU system consist of those radioisotopes carried in the reactor water. The activity inventory is based on component transit times. The radioisotopes for the RWCU recirculation pumps, regenerative and nonregenerative heat exchangers, filter/demineralizers, holding pumps, and the backwash receiving tank are the accumulated fission, activation, and corrosion products, based on the inlet reactor coolant concentrations given in Tables 12.2-1 through 12.2-5, allowing for decay due to transit time. Tables 12.2-16 through 12.2-23 provide the shielding design source terms for these components.

12.2.1.2.2 Residual Heat Removal System

The pumps, heat exchangers, and associated piping of the RHR system are carriers of radioactive materials. For plant shutdown, the RHR pumps and heat exchangers are the radiation sources, resulting from the radioisotopes carried in the reactor coolant after four hours of decay following shutdown. The source terms listed in Table 12.2-24 are used for the shielding calculations for this system.

12.2.1.2.3 Reactor Core Isolation Cooling System

Components of the RCIC system that contain radiation sources are the RCIC turbine and steam inlet and exhaust piping. The steam radioactivity, as discussed in Section 12.2.1.1.3, without decay correction for transit from the RPV steam nozzles, is used for the shielding calculations for this system and is listed in Table 12.2-25.

12.2.1.2.4 High Pressure Coolant Injection System

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

12.2.1.2.5 Core Spray System

The core spray system components, during testing, use condensate from the CST which contains very low radioactivity concentrations (Table 12.2-87); therefore, no shielding is required.

12.2.1.2.6 Spent Fuel Storage and Transfer

The predominant radiation sources in the spent fuel storage and transfer areas are the spent fuel assemblies. For shielding design purposes only, the spent fuel pool is assumed to contain 2862 fuel assemblies. These spent fuel assemblies are conservatively assumed to have 48 hours of decay. Shielding design source terms are shown in Table 12.2-14.

12.2.1.2.7 Fuel Pool Cooling and Cleanup System

Sources of radiation in the FPCC system result from the transfer of radioisotopes from the reactor coolant and crud deposits on spent fuel assemblies into the spent fuel pool during refueling operations. The shielding design source terms for the FPCC system are presented in Table 12.2-27. These source terms then undergo subsequent decay and accumulation on the FPCC filter/ demineralizers. Table 12.2-28 shows the FPCC filter/demineralizers shielding design source terms. The shielding design source terms for the FPCC heat exchanger are shown in Table 12.2-29.

12.2.1.2.8 Reactor Enclosure HVAC System

Components of the reactor enclosure HVAC system that contain sources of radioactivity are the equipment compartment exhaust air filters. Table 12.2-30 shows the reactor enclosure equipment exhaust air filters shielding design source terms.

12.2.1.2.9 Control Rod Drives

Shielding design source terms for the CRD mechanisms after removal from the RPV are shown in Table 12.2-31.

12.2.1.3 Turbine Enclosure

12.2.1.3.1 Primary Steam and Power Conversion Systems

Radiation sources for piping and equipment that contain primary steam are based on the radioisotopes carried over into the main steam from the reactor coolant and include fission

product gases and halogens, particulate fission and corrosion products, and gaseous activation products as discussed in Section 12.2.1.1.3. Steam density variations and steam transit times through equipment and pipes are factored into the shielding source term evaluation to account for volumetric dilution effects, radiological decay, and daughter product generation. Tables 12.2-32 through 12.2-36 show the shielding design source terms for the following components that use main steam; moisture separators, cross-around piping, feedwater heaters, steam seal evaporator, and the reactor feed pump-turbine.

12.2.1.3.2 Condensate and Feedwater Systems

The radiation sources in the condensate and feedwater systems are based on decayed main steam radioactivity (Section 12.2.1.1.3). Eighty percent of the N-16 and 100% of the noble gases are assumed to be removed from the condensate and feedwater systems by the main condenser air removal system. The gaseous radiation sources in the hotwell are shown in Table 12.2-37; they are negligible in the remainder of the condensate and feedwater systems. The hotwell is designed for a two minute holdup of condensate, and therefore N-16 radioactivity at the condenser outlet is negligible. Particulate fission products, activated corrosion products, and the particulate daughter products from the decay of fission product gases in transit through the turbine and condenser are the inlet radiation sources to the condensate system. These shielding design source terms, as shown in Table 12.2-38, are present in the condensate pumps and piping and accumulate on the condensate filter/ demineralizers. Table 12.2-39 provides the shielding design source terms for the condensate filter/demineralizer and the condensate backwash receiving tank. The shielding design source terms for the feedwater system are listed in Table 12.2-40.

12.2.1.3.3 Gaseous Radwaste Recombination System

Shielding design sources in the gaseous radwaste recombination system originate from noble gases and other noncondensable gases removed from the main condenser, and the radioactivity entering with the extraction driving steam to the SJAEs. The radioactivity entering is based on the primary steam radioactivity as described in Section 12.2.1.1.3, decayed for the expected transit time to the SJAEs. Eighty percent of the N-16 and 100% of the noble gases are assumed to be removed from the condenser by the SJAEs. The specific activities or quantities of radioactivity, including particulate daughters, in the SJAE condenser, mechanical vacuum pump, offgas pipe, preheater, recombiner, recombiner catalyst, aftercondenser, and H_2 analyzers, to be used for shielding design calculations, are shown in Tables 12.2-41 through 12.2-48.

12.2.1.3.4 Turbine Enclosure HVAC System

Components of the turbine enclosure HVAC system that contain sources of radioactivity are the equipment compartment exhaust air filters and the SGTS air filters. Tables 12.2-49 and 12.2-50 show the turbine enclosure equipment exhaust air filters and the SGTS air filters shielding design source terms.

12.2.1.4 Radwaste Enclosure

12.2.1.4.1 Liquid Waste Management System

Liquid radwaste is collected and processed as discussed in Section 11.2. The liquid waste management system shielding design sources are radioisotopes, including fission and corrosion products, present in the reactor coolant. The components of this system contain varying amounts of radioactivity, depending on the system and equipment design.

The concentrations of radioisotopes used for shielding design for pipes, tanks, filters, demineralizers, abandoned and unused evaporators (Section 11.2.2.1.3), and equipment and floor drain sumps may be derived from or are listed in Tables 12.2-51 through 12.2-69. Shielding for each component of the liquid waste management system is based on reactor coolant concentrations given in Tables 12.2-1 through 12.2-5.

12.2.1.4.2 Solid Radwaste System

The solid radwaste system collects, monitors, processes, packages, and provides temporary storage facilities for radioactive spent bead and powdered resins and dry solid wastes for offsite shipment and permanent disposal. The system is described in Section 11.4.

The high integrity containers used for packaging resin wastes are washed free of external surface contaminants and stored in concrete shielded compartments before shipment. The aforementioned operations are accomplished using remote container loading, transfer, and an overhead crane. Shielding design source terms for the solid radwaste system components are based on reactor coolant concentrations given in Tables 12.2-1 through 12.2-5 and are presented in Tables 12.2-70 through 12.2-77, and 12.2-101 through 12.2-103.

12.2.1.4.3 Radwaste Enclosure HVAC System

Components of the radwaste and offgas enclosures HVAC system that contain sources of radioactivity are the equipment compartment exhaust air filters. Table 12.2-78 shows the radwaste and offgas enclosure equipment exhaust air filters shielding design source terms.

12.2.1.5 Offgas Enclosure

12.2.1.5.1 Gaseous Radwaste Charcoal Treatment System

The gaseous radwaste charcoal treatment system as described in Section 11.3 is located in the offgas enclosure and receives effluent from the gaseous radwaste recombination system for further decay before release.

The shielding design terms for the charcoal treatment system components are based on the expected transit times for noble gases and the formation and accumulation of noble gas daughter products. These gases pass through the charcoal treatment system and out the turbine enclosure stack. The shielding design source terms for the piping, filters, and charcoal treatment system equipment are presented in Tables 12.2-79 through 12.2-82.

12.2.1.5.2 Offgas Enclosure HVAC System

See Section 12.2.1.4.3 for the offgas enclosure HVAC shielding design source terms.

12.2.1.6 Shielding Design Sources Resulting from Design Basis Accidents

The shielding design for the control room is based on radiation sources resulting from DBAs. Control room shielding design considers radiation sources from the primary containment, the reactor enclosure, the turbine enclosure and control structure, and the SGTS filters. The shielding design source terms for these areas are given in Tables 12.2-83 through 12.2-86.

With regards to the Independent Spent Fuel Storage Installation, a postulated accidental sealing failure can potentially produce a gaseous effluent. This effluent is limited by the requirements of 10CFR72.106, which does not require consideration of simultaneous contributions to dose from the plant.

12.2.1.7 Stored Radioactivity

The only sources of radioactivity not stored inside the plant structures are the refueling water storage tank, the CST, 10CFR20.2002 storage area, the radwaste storage pad and designated material laydown areas in the protected area.

During normal operation of the Independent Spent Fuel Storage Installation only direct radiation is emitted from loaded dry storage containers. The radiation dose is limited by the requirements of 10CFR72.104, which considers the direct dose from the storage containers in combination with the normal plant sources and effluents.

The CST contains low concentrations of radioisotopes. A dike is provided around the CST so that unrestricted access is limited to areas with a dose rate less than 2 mRem/hr. The CST source terms are shown in Table 12.2-87.

The refueling water storage tank also has low concentrations of radioisotopes when water is returned from the refueling pool. The refueling water storage tank is surrounded by a dike which limits unrestricted access to areas with a dose rate less than 0.2 mRem/hr. The refueling water storage tank source terms are shown in Table 12.2-88.

Provisions have been made to recycle the water from both the condensate and refueling water storage tanks through the condensate filter/demineralizers.

The 10CFR20.2002 storage area is an area north of the radwaste storage pad and northwest of the spray pond. The area is approximately 1.5 acres in size and approved for storage of slightly contaminated soils, sediment and sludges as approved by the 10CFR20.2002 application. Annual and maximum limits for the amount of material and radioactivity concentration are established by the 10CFR20.2002 application for the material stored there. The offsite dose at the nearest residence will not exceed 0.101 mrem/year from the radioactive waste material stored at the 10CFR20.2002 storage area.

The radwaste storage pad is located outside the protected area, but within the site restricted area, southwest of the spray pond and northwest of the Unit 1 cooling tower. The waste material on the pad consists of low level waste being stored on a temporary (interim) basis, awaiting offsite shipment, and contaminated reusable material. Storage pad boundary dose rates are set such that exposure of personnel in unrestricted areas is in accordance with 10CFR20.1301 requirements. The normal offsite dose to any member of the public will not exceed 1.0 mR/year from the radioactive waste material on the storage pad and 1.0 mR/year from storage of contaminated reusable material.

Designated area within the protected area are used for temporary storage of very low level contaminated material. This material consists of reusable materials to support plant operations and maintenance activities, and materials awaiting processing or offsite shipment for subsequent recycling, recovery, or free release to the extent achievable. The normal offsite dose from this category of material is included in the 1.0 mr/yr limit from storage of contaminated reusable material on the radwaste storage pad.

No other radiation sources aside from the Independent Spent Fuel Storage Installation (ISFSI) are normally stored outside the plant structures. Spent fuel is stored in the spent fuel pool until it is placed in the spent fuel shipping cask for offsite transport or into the ISFSI Transfer Cask for placement at the ISFSI for interim storage. Space is provided in the radwaste enclosure for storage of spent filter cartridges and solidified spent resins. Radiation sources stored inside the plant structures are shielded to provide a dose rate of less than 2 mRem/hr for all areas outside plant structures.

12.2.1.8 Special Sources

Special materials used in the radiochemistry laboratory and sealed sources used for calibration require special handling equipment and are shielded accordingly. Unsealed sources and radioactive samples are handled in conventional hoods that exhaust to the ventilation system. Design features provided are discussed in Section 12.3.1.

The TIP system gamma detector and its drive cable become radiation sources following activation by neutrons in the reactor. The level of the radiation source depends upon the material compositions of the components, the irradiation history and decay time. The material composition of the gamma TIP is shown in Table 12.2-89. The radiation levels from the detector and cable are shown in Table 12.2-104 for a range of decay times after the TIP is retracted from the reactor vessel.

The reactor startup sources are shipped to the site in special shielded casks. The sources are transferred from the cask to the source holders. The source holders are then loaded underwater into the reactor. The sources are removed from the reactor and placed in the spent fuel pool or sent offsite.

12.2.2 Airborne Radioactive Material Sources

12.2.2.1 Sources of Airborne Radioactivity

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

12.2.2.2 Production of Airborne Radioactive Materials

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

12.2.2.3 Locations of Sources of Airborne Radioactivity

Practically all the sources of airborne radioactivity are found in the reactor, turbine, and radwaste and offgas enclosures. Within these structures the radioactivity is released in equipment cubicles, valve and piping galleries, sampling stations, radwaste handling and shipping areas, cleaning and decontamination areas, and repair shops.

12.2.2.4 Control of Airborne Radioactivity

Ventilation is an effective means of controlling airborne radioactive materials. Ventilation flow paths are such that air from low potential airborne areas flows into higher potential airborne areas. This flow pattern ensures that radioactivity released in the above mentioned source locations, which usually have low personnel access requirements, has little chance to escape into areas with a high personnel occupancy such as corridors, working aisles, and operating floors. Levels of airborne radioactivity are periodically checked by surveys of the plant by the radiation protection staff.

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

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

a. Estimate the total airborne releases (in curies per year) for each of the plant enclosures.

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

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

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

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

12.2.2.6 Estimation of Total Airborne Releases Within the Plant

The estimated quantities of airborne radioactive material produced in the plant enclosures are given in Table 12.2-93. These releases were based upon BWR-GALE (Reference 12.2-1), a computerized mathematical model for calculating the release of radiological materials in gaseous and liquid effluents. Assumptions applicable to the development of Table 12.2-93 from BWR-GALE are as follows:

a. The reactor enclosure releases are taken to be the sum of the auxiliary enclosure and containment enclosure releases calculated by BWR-GALE.

b. Turbine enclosure releases from BWR-GALE are assumed to include any airborne radioactive material produced in the control structure.

c. The radwaste enclosure releases from BWR-GALE are "per reactor" and consequently are doubled for LGS. Offgas enclosure releases are assumed to be included in the radwaste enclosure releases.

d. Tritium releases from BWR-GALE are divided equally between the reactor and turbine enclosures.

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

12.2.2.7 Distribution of Airborne Releases Within the Plant

In the approach taken to determine the anticipated distribution of gaseous effluents, it is assumed that all airborne radioactive material originates only within the equipment areas of the plant. It is further assumed that a major percentage of the release is generated within a few specific areas of each enclosure, with the remainder coming from other equipment areas. Eighty percent of each enclosure's release is distributed as described below among the major contributing areas, and 20% is assigned to the "other equipment areas" category. Releases are assumed to be generated continuously throughout the year except for the drywell, where a 30 day release period is used.

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

Table 12.2-94 lists the major airborne contributors in each enclosure and the percentage of the total enclosure release assigned to each. Tables 12.2-95 through 12.2-97 provide the specific equipment areas of the plant associated with the major contributors and the applicable exhaust air flow rates. Note that only those equipment areas that have a significant potential for airborne radioactive material releases were included in the
"other equipment areas" category.

12.2.2.8 Estimated Airborne Radioactive Material Concentrations Within the Plant

The airborne radionuclide concentrations for each equipment area were calculated using the following methodology. For a specific area, the appropriate enclosure release (Table 12.2-93) was multiplied by the applicable release percentage for the area (Table 12.2-94) and divided by the area's annual exhaust flow (Table 12.2-95, 12.2-96, or 12.2-97). The resultant concentrations are presented in Tables 12.2-98 through 12.2-100, which also include the fractions of the maximum permissible concentrations in air as defined in 10CFR20 Appendix B, Table I (pre-1994).

12.2.2.9 Changes to Source Data Since PSAR

Airborne radioactive material sources were not specified in the LGS PSAR. Section 12.2.2 has been added in compliance with the "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Regulatory Guide 1.70.

12.2.3 REFERENCES

12.2-1 NUREG-0016 (Revision 0), "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Boiling Water Reactors," Office of Standards Development, NRC, Washington, D.C. (April 1976).

12.2-2 EPRI NP-495, "Sources of Radioiodine at Boiling Water Reactors," Project 274-1, Final Report, (February 1978).

Table 12.2-6

BASIC REACTOR DATA[1]

PARAMETER		VALUE USED IN MODEL
Reactor rated thermal power		3527 megawatts
Overall average core power density		53.8 watts/cc
Core power peaking factors:		
At core center		
$\frac{Pmax}{Pave_Z}$	(axial)	1.5
$\frac{Pmax}{Pave_R}$	(radial)	1.4
At core boundary		
$\frac{Pmax}{Pave_Z}$	(axial)	0.5
$\frac{Pmax}{Pave_R}$	(radial)	0.7

Core volume fractions:

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

Average water density between core and vessel and below the core	0.74 g/cc
Average water-steam density above core:	
In the plenum region	0.23 g/cc
Above the plenum (homogenized)	0.6 g/cc
Average steam density	0.036 g/cc

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

Table 12.2-7

CORE REGION DESCRIPTION TO DETERMINE
RADIAL FLUX DISTRIBUTION AT REACTOR CORE MIDPLANE

REGION DESCRIPTION	REGION THICKNESS (cm)	CUMULATIVE THICKNESS (cm)	MATERIAL
Active fuel zone	237.60	237.60	Core
Water	20.37	257.97	Water
Core shroud	5.08	263.05	Stainless steel
Water	58.26	321.31	Water
Pressure vessel liner	0.476	321.786	Stainless steel
Pressure vessel	16.354	338.14	Carbon steel
Air	51.76	389.89	Air
Steel liner (primary shield)	1.27	391.16	Carbon steel
Concrete (primary shield)[1]	48.26	439.42	Magnetite/ ilmenite
Steel liner (primary shield)	3.81	443.23	Carbon steel
Air gap	418.78	862.01	Air
Primary containment	195.58	1057.59	Ordinary concrete

(1) Ilmenite/magnetite aggregate mixture

Table 12.2-8

MATERIAL COMPOSITION TO DETERMINE RADIAL FLUX DISTRIBUTIONS AT REACTOR CORE MIDPLANE

ELEMENT	MATERIAL (10^{24} atoms/cc)						
	CORE	WATER	STAINLESS STEEL-304	CARBON STEEL	CONCRETE[1]	AIR	ORDINARY CONCRETE
H	1.83×10^{-2}	4.89×10^{-2}			3.66×10^{-3}		7.84×10^{-3}
O	2.26×10^{-2}	2.44×10^{-2}			4.51×10^{-2}	1.12×10^{-5}	4.41×10^{-3}
Mo	5.93×10^{-4}						
U-235	1.24×10^{-4}						
U-238	5.46×10^{-3}						
Mg					9.69×10^{-4}		1.43×10^{-4}
A					1.32×10^{-3}		2.39×10^{-3}
Si			1.69×10^{-3}	4.19×10^{-4}	5.49×10^{-3}		1.57×10^{-2}
Ca					4.00×10^{-3}		2.91×10^{-3}
Ti							2.17×10^{-3}
Mn			1.73×10^{-3}	8.59×10^{-4}	4.86×10^{-4}		
Fe			5.76×10^{-2}	8.40×10^{-2}	1.16×10^{-2}		3.09×10^{-4}
C			3.18×10^{-4}	1.00×10^{-3}			
Cr			1.73×10^{-2}				
Ni			8.06×10^{-3}				
N						4.17×10^{-5}	
Na							1.05×10^{-3}
K							6.91×10^{-4}
S							5.30×10^{-5}

[1] Magnetite/ilmenite mixture