CHAPTER 12- RADIATION PROTECTION

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

CHAPTER 12- RADIATION PROTECTION

LIST OF TABLES

LIST OF TABLES (cont'd)

CHAPTER 12- RADIATION PROTECTION

LIST OF FIGURES

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]

PARAMETERVALUE 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

[redacted]

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

Specific design features for maintaining personnel exposures ALARA are discussed in this section. These are used in addition to, and in conjunction with, the more generalized design features described in Section 12.1.2.

12.3.1.1 Common Equipment and Component Designs for ALARA

This section describes the design features used for several general classes of equipment and components. Since these classes of equipment are common to many of the plant systems, the features employed in each system to minimize exposures are similar and can be discussed generically by equipment type.

Filters: Filters that accumulate radioactivity are supplied with the means either to backflush/recharge the filter remotely or to perform cartridge replacement with semiremote tools (i.e., long-handled tools). For cartridge filters, adequate space is provided to allow removing, loading, and transporting the cartridge to the solid radwaste storage area.

Demineralizers: Demineralizers in radioactive systems are designed so that spent resin can be remotely transferred to spent resin tanks and fresh resin can be remotely loaded into the demineralizers. The demineralizers and piping can be flushed with condensate or demineralized water to remove any accumulations of resin. The equipment and floor drain demineralizers are typical examples; these are shown on drawings M-62 and M-63, respectively.

Evaporators: System abandoned (Section 11.2.2.1.3).

Pumps: Wherever practicable, pumps in radioactive areas are provided with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space and access for maintenance. Small pumps are installed in a manner that allows easy removal if necessary. All pumps in radioactive waste systems are provided with flanged connections for ease of removal. Pump casings have connections for draining the pump prior to maintenance. The use of base plates with drains connected to the floor drain system minimizes the spread of contamination resulting from leakage of continuously operating pumps.

Tanks: Whenever practicable, tanks are provided with sloped bottoms and bottom outlet connections. Overflow lines are lower than vent lines and are directed to the waste collection system to prevent an overflow from spreading contamination within plant structures. For tanks containing radioactive material, the tank and associated discharge piping can be flushed to reduce radiation levels if required for entry into the tank cells. Access is provided for removal, maintenance, or inspection of tank motor agitators and also for cleaning operations involving tank internals. The solid radwaste collection system (drawing M-66 and Figure 12.3-2) provides examples of these features.

Heat Exchangers: Heat exchangers are provided with corrosion-resistant tubes of stainless steel or other suitable materials with tube-to-tube sheet joints usually welded to minimize leakage. Impact baffles are provided and tube side and shell side velocities are limited to minimize erosive effects. Space is reserved for tube bundle removal, and provisions for flushing are supplied.

<u>Instruments</u>: Instrument devices are located in low radiation zones and away from radiation sources whenever practicable to reduce personnel exposure during maintenance. Primary instrument devices, which for functional reasons are located in high radiation zones, are designed for easy removal to a lower radiation zone for maintenance and calibration. Some instruments located in high radiation zones, such as thermocouples, are provided in duplicate to preclude the need for immediate entry in case of instrument failure and allow maintenance to be performed at a later time when radiation levels may be lower. Transmitters and readout devices are located in low radiation zones. Instrument and sensing lines are provided with flushing capability and are routed to minimize radioactive gas buildup. Backflushing capability exists for reactor vessel sensing lines. Tanks containing two-phase fluids are fitted with probe-type instruments.

Whenever practicable, diaphragm seals are provided on instrument sensing lines on process piping that contains highly radioactive solids to reduce radiation exposures during servicing of the instrument. Instrument and sensing line connections are typically located in or above the piping midplane to avoid corrosion product buildup.

<u>Valves</u>: To minimize personnel exposures from valve operations and maintenance, motor-operated, air-operated, or other remotely actuated valves are used to the maximum extent practicable where the reduction in operational exposure is greater than the expected increase in maintenance exposure associated with the remote operator.

Whenever practicable, valves are located in valve galleries and are shielded separately from the major components that accumulate radioactivity. Long runs of exposed radioactive piping are minimized in valve galleries. In areas where manual valves are used on frequently operated radioactive process lines, either reach rods or radiation shielding is provided to minimize personnel exposure.

For equipment located in high radiation zones, remote actuators are provided for frequently operated valves associated with system operation. Examples of remote actuators can be seen on drawing M-66. All other valve operations are either infrequent or performed when the equipment is not operating. To the maximum extent practicable, these valves are provided with straight reach rods to allow operators to feel whether or not the valves are tightly closed. Valves with reach rods are installed with their stems located in a horizontal position wherever possible, so that the reach rods are also horizontal but above head level to prevent restriction of access during maintenance.

Provisions are made in many radiation areas to drain adjacent radioactive components when maintenance is required on valves.

Wherever practicable, valves for clean, nonradioactive systems are separated from radioactive sources and are located in readily accessible areas. Vent, drain, and instrument root isolation valves on radioactive systems are located close to the process piping or equipment with which they are associated. This minimizes the lengths of piping carrying process fluids when these valves are closed.

Manually operated valves required for normal operation and shutdown are not located in filter and demineralizer valve compartments.

For large valves (2½ inches and larger) originally supplied and installed in lines carrying radioactive fluids, a double set of packing with a lantern ring is usually provided. Such valves,

except those having a packing gland which constitutes a direct communication path between primary containment atmosphere and the reactor building, are being systematically provided with an improved packing arrangement that eliminates use of the second packing set. Power-actuated valves so changed are also being fitted with automatic gland adjustment capability. A stuffing box is also provided with a leak-off connection that may be piped to a drain header. Flow from the leak-off is stopped by a plug or a manual stop valve. Full ported valves are used in systems containing radioactive solids.

Valve designs with minimum internal crevices are used where crud trapping could become a problem, such as in piping carrying spent resin. Globe valves two inches and smaller in radioactive systems are Y-pattern-type to facilitate rodding if plugged.

Piping: The piping in pipe chases is designed for the lifetime of the unit. The number of valves or instruments in the pipe chases has been reduced to the maximum extent practicable. Piping layout is discussed in Section 12.3.1.2.

Floor Drains: Floor drains and sloped floors are provided for each room or cubicle that has serviceable components containing radioactive liquids. Whenever practicable, drain lines are embedded in concrete floors which provide shielding. If a radioactive drain line must pass through a radiation zone lower than that at which it will terminate, proper shielding is provided. Local gas traps or porous seals are not used on radwaste floor drains. Gas traps are provided at the common sump or tank. Wherever practicable, provisions exist for removing plugging in drain piping. Floor drains are designed to handle potential backflooding.

Lighting: Multiple electric lights are provided for each cell or room containing highly radioactive components so that the burnout of a single lamp does not require entry and immediate replacement of the defective lamp; sufficient illumination still remains available. Lighting in a radioactive area is actuated from outside the area and long-life bulbs are used. Section 9.5.3 describes the lighting system.

HVAC: The HVAC system is designed to minimize radioactive buildup and provides for easy access and fast replacement of the filter elements. Filter banks and components are separated from adjacent banks and components. Section 12.3.3 provides additional description of the HVAC radiation protection design features.

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

Clean Services: Wherever practicable, clean services such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipe-ways. In addition, active components of these clean services are located outside high radiation areas wherever possible to minimize any radiation exposure associated with maintenance of clean systems.

12.3.1.2 Common Facility and Layout Designs for ALARA

This section describes the design features used for standard-type plant processes and layout situations for radioactive systems and for potentially radioactive systems. These features are

used in conjunction with the general equipment designs described in Section 12.3.1.1 and include the details discussed in the following paragraphs:

Valve Galleries: Valve galleries are provided with shielded entrances for personnel protection. In many cases, the valve galleries are divided by shielding or distance into subcompartments that service only two or three components so that personnel are only exposed to the valves and piping associated with a few components at any given location. Process isolation valves are located close to wall penetrations. Floor drains are provided to control radioactive leakage, and curbs are supplied as necessary. To facilitate decontamination in valve galleries, concrete surfaces are covered with a smooth surfaced coating that allows easy decontamination.

Piping: Each piping run is analyzed to determine the potential radioactivity level and maximum expected surface dose rate. Radioactive pipes are routed separately from nonradioactive pipes to minimize personnel exposure. Pipes carrying radioactive materials are routed through controlled access areas zoned for a corresponding level of activity. Where radioactive piping must be routed through corridors or other low radiation areas, shielded pipe-ways are provided. Valves and instruments are not normally placed in radioactive pipe-ways. Wherever practicable, each equipment compartment is used as a pipe-way only for those pipes associated with equipment in the compartment. Doing so minimizes exposure due to the operation of one system while maintenance is being performed on another system that is shut down.

Piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Where possible, thermal expansion loops are raised rather than dropped. In radioactive systems, the use of nonremovable backing rings in the piping joints is minimized to eliminate a potential crud trap for radioactive materials. Wherever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe. Line size changes are typically made by eccentric reducers. Orifices are placed in vertical lines wherever possible.

Piping carrying resin slurries is run with large radius bends wherever possible instead of elbows, and horizontal runs are minimized. To prevent possible crud buildup, flow control valves and orifices are not normally used unless they are required for system operation. Large diameter piping is typically used with a minimum number of pipe fittings to reduce crud accumulation.

Field Run Piping: All routing of radioactive process piping, large and small, is reviewed by the engineering office to ensure that the radiation zone routing is proper and that the above principles are being employed.

Penetrations: Penetrations are normally located with an offset between the source and the accessible areas to minimize radiation streaming. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. If neither of these two methods is used, then alternative means are employed, such as using baffle shield walls or radiation shielding in the area around the penetration.

Contamination Control: Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. Equipment vents and drains from radioactive systems are normally piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All-welded piping systems are used for radioactive systems to the maximum extent practicable to reduce system leakage and crud buildup at joints. The

valves in some radioactive systems are provided with leak-off connections piped directly to the collection system.

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

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

Systems that become highly radioactive, such as the radwaste slurry transport system, are provided with flush and drain connections. Certain systems have provisions for chemical cleaning prior to maintenance.

Equipment Layout: In systems where process equipment is a major radiation source (such as fuel pool cleanup, radwaste, condensate demineralizer, etc.), pumps, valves, and instruments are separated from the process component. This allows servicing and maintenance of items in reduced radiation zones. Control panels are located in the lowest dose radiation zones. Redundant equipment can be separated from each other, and shielding is provided between the equipment to allow maintenance concurrent with system operation.

Except for HVAC components, major components (such as tanks, demineralizers, and filters) in radioactive systems are located in individual shielded compartments. For highly radioactive components (such as filters and demineralizers), completely enclosed shielded compartments with hatch openings are provided. Provision is made for some major plant components for removal to lower dose radiation zones for maintenance. Large HVAC filter plenums with multiple filter cartridges are individually shielded.

Labyrinth entrance-way shields or shielding doors are provided for each compartment or plenum from which radiation could stream to access areas and exceed the radiation dose limits for those areas. Adequate space for removal of components is provided.

Wherever practicable, lubrication of equipment in radiation areas is achieved with the use of tube-type extensions to reduce exposure during maintenance.

Figures 12.3-1 to 12.3-7 provide typical layout arrangements for demineralizers, liquid and particulate filters, waste sludge tanks, offgas recombiners, sample stations, charcoal beds, and their associated valve compartments or galleries.

Exposure from routine in-plant inspection is controlled by locating inspection points in shielded low background radiation areas wherever possible. Radioactive and nonradioactive systems are normally separated 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 shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is normally provided. In high radiation areas where routine surveillance is required, remote viewing devices are provided as needed. Typically, equipment and valves in high radiation areas are made easily accessible by providing permanent access platforms, easily removable insulation, etc. Equipment man-ways are readily accessible. Equipment lay-down area requirements are considered in the layout, and adequate space is provided where necessary.

Facilities for Handling Sealed and Unsealed Radioactive Material: As discussed in Section 12.2.1.8, special materials used in the radiochemistry laboratory require the design of special handling equipment. For unsealed materials, the following are provided:

a. Exhaust hoods that exhaust to the ventilation system are located in areas such as sample stations and the radiochemistry laboratory.

b. Decontamination facilities, the radiochemical laboratory, controlled zone shop, and instrument repair shops are situated at various locations in the plant and are described in Section 12.5.

c. An area for the repair and maintenance of removed CRDs is provided in the reactor enclosure in close proximity to the CRD removal hatch.

12.3.1.3 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yards is regulated and controlled. Each high radiation area (as defined in 10CFR20) is provided with a personnel alert barrier. Each high radiation area with a dose rate greater than 1 Rem/hour and very high radiation areas (as defined in 10CFR20) require additional controls per Health Physics Supervision. Section 12.5 describes the control of ingress and egress of plant operating personnel to controlled access areas and the procedures employed to ensure that personnel exposure is within the limits prescribed by 10CFR20.

All plant areas are categorized into radiation zones according to expected radiation levels. Each radiation zone defines either the highest component dose rate in the area or the radiation level to which the aggregate of all contributing sources must be attenuated by shielding, whichever is higher. Each room, corridor, and pipe-way of every plant structure is evaluated for potential radiation sources during normal operation, including anticipated operational occurrences and shutdown. The radiation zone categories used, and their descriptions, are given in Table 12.3-1, and the specific zoning for each plant area is shown in 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-139, N-140, N-141, N-142, and N-143. All frequently used areas such as corridors are shielded for Zone I or Zone II access.

The locations of airborne radioactivity and area radiation monitors are described in Section 12.3.4. Unit 1 and common area radiation monitors are shown on 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-139, N-140, N-141, N-142, and N-143, and Table 12.3-7. Unit 2 area radiation monitors are shown on Table 12.3-7.

12.3.1.4 Control of Activated Corrosion Products

To minimize the radiation exposure associated with the deposition of activated corrosion products in reactor coolant and auxiliary systems, the following steps have been taken:

a. The reactor coolant system consists mainly of austenitic stainless steel, carbon steel, and low alloy steel components. The nickel content of these materials is low, and it is controlled in accordance with applicable ASME material specifications. A small amount of nickel base material (Inconel 600) is employed in the reactor vessel internal components. Inconel 600 is required where components are attached to the reactor vessel shell, and the coefficient of expansion must match

the thermal expansion characteristics of the low alloy vessel steel. Inconel 600 is selected because it provides the proper thermal expansion characteristics and adequate corrosion resistance, and it can be readily fabricated and welded.

b. Materials employed in the reactor coolant system are purchased to ASME material specification requirements. No special controls on levels of cobalt impurities are specified.

c. Hard-facing and wear materials having a high percentage of cobalt are restricted to applications where no satisfactory alternative materials are available.

d. A high temperature filtration system is not employed in the RWCU. The reasons for this decision include:

   1. Lack of quantitative data on the removal efficiency for insoluble cobalt by the high temperature filter.

   2. Uncertainty in the deposition model, including the relative effectiveness of cobalt removal on deposition rate.

   3. Doubtful cost-effectiveness in an area where other methods under study (such as decontamination) may prove better at reducing dose rates while also being more cost-effective.

e. Items a, b, and c above also apply to valve materials in contact with reactor coolant. Valve packing materials are selected primarily for their properties in the particular environment.

f. Sections 12.1.2.2, 12.3.1.1, and 12.3.1.2 describe the various flushing, draining, testing, and chemical addition connections that have been incorporated into the design of piping and equipment that handle radioactive materials. If decontamination is to be performed, these connections are used for that purpose.

g. The plant is designed with a powdered resin, pressure precoat cleanup system for the primary coolant in the reactor, and a full flow condensate cleanup system for the feedwater. These systems are described in Sections 5.4.8 and 10.4.6, respectively.

h. A chemistry control program to reduce crud buildup has been developed based upon studies performed by GE. This program will be implemented at LGS.

12.3.2 SHIELDING

In this section the bases for radiation shielding and shielding configurations are discussed.

12.3.2.1 Design Objectives

The basic objective of the plant radiation shielding is to reduce personnel exposures, in conjunction with a program of controlled personnel access to and occupancy of radiation areas, to levels that are within the dose regulations of 10CFR50 and are ALARA within the dose regulations

of 10CFR20. Shielding and equipment layout and design are considered in ensuring that exposures are kept ALARA during all anticipated personnel activities in all areas of the plant containing radioactive materials.

The plant conditions considered in radiation shielding design are normal operation including anticipated operational occurrences at full power, and plant shutdown. The shielding design objectives are as follows:

a. To ensure that radiation exposure to plant operating personnel, contractors, administrators, visitors, and proximate exclusion area boundary occupants are ALARA and within the limits of 10CFR20.

b. To ensure sufficient personnel access and occupancy time to allow normal anticipated maintenance, inspection, and safety-related operations required for each plant equipment and instrumentation area.

c. To reduce potential equipment neutron activation and mitigate radiation damage to materials.

The control room is sufficiently shielded so that the direct dose plus the inhalation dose (calculated in Chapter 15) in the event of DBAs does not exceed the limits of GDC 19.

12.3.2.2 General Shielding Design

Shielding is provided to attenuate direct radiation through walls and penetrations, and to attenuate scattered radiation to less than the upper limit of the radiation zone for each area shown in 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, which show the minimum shielding requirements for all plant areas. Shielding design criteria and shielding design source terms for specific plant equipment are presented in Section 12.2.

The material used for most of the plant shielding is ordinary concrete with a minimum bulk density of 140 lb/ft^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 shielding characteristics of the concrete block and associated grout fill material selected. As discussed in Section 1.8, Regulatory Guide 1.69 is not employed for LGS. Further discussion of concrete standards is contained in Section 3.8. Water is used as the primary shield material for areas above the spent fuel transfer and storage areas.

The guidance provided in Regulatory Guide 8.8 was followed for LGS. ALARA design features are discussed in Sections 12.1.2 and 12.3.1. Specific features pertaining to shielding are described below.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions, as described in Section 12.2. The thickness of each shield wall surrounding radioactive equipment is determined by approximating the actual geometry and physical condition

of the source or sources. The isotopic concentrations are converted to gamma ray sources using data from standard References 12.3-1 through 12.3-5.

The geometric model assumed for shielding evaluation of pipes, tanks, heat exchangers, filters, demineralizers, and evaporators (abandoned) is a finite cylindrical volume source. In cases where radioactivity is deposited on surfaces, such as inside pipe, the source is treated as a cylindrical annulus. Typical computer codes that are used for original design shielding analysis are listed in Table 12.3-2, which includes References 12.3-6 through 12.3-15. Shielding attenuation data used in those codes include gamma mass attenuation coefficients (Reference 12.3-16), gamma buildup factors (Reference 12.3-17), neutron/gamma multigroup cross-sections (Reference 12.3-18), and albedos (Reference 12.3-19). Additional sources of information pertaining to shielding techniques include References 12.3-20 through 12.3-27. This list of publications and computer programs are as applied in the original plant design and are not meant to be all-inclusive.

Shielding design for plant changes use programs and references which are industry standards at the time of the change. Examples of standards include those developed by the American Nuclear Society's ANS-6 Radiation Protection and Shielding Division, and the International Commission on Radiological Protection.

Computer programs used include those distributed by the Oak Ridge National Radiation Safety Information Computational Center, or commercially developed programs implementing established methods and parameters in industry standards.

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 below this maximum dose and therefore below the radiation zone upper limit.

Where shielded entry-ways to compartments containing high radiation sources are necessary, labyrinths or mazes are designed so that the scattered dose rate plus the direct dose rate through the shield wall from all contributing sources is below the upper limit of the radiation zone specified for each plant area.

12.3.2.4 Turbine Enclosure Shielding Design

Radiation shielding is provided as necessary for the following systems in the turbine enclosure and control structure, excluding the control room, to ensure that zone access requirements (drawings N-110, N-111, N-112, N-113, N-115, N-125, N-126, N-127, N-128, and N-130) are met for the surrounding areas:

a. Main steam system

b. Condensate system

c. Feedwater system

d. Air removal system

e. Sealing steam system

f. Condensate cleanup system

g. Gaseous radwaste recombiner system

h. Turbine enclosure HVAC system

i. Control room HVAC system.

12.3.2.5 Control Room Shielding Design

Drawings N-113, N-115, N-116, N-117, N-118, N-119, N-120, N-121, N-122, N-124, N-128, N-130 N-131, N-132, N-133, N-134, N-135, N-136, N-137, N-139, N-140, N-141, N-142, N-143 and Figure 12.3-26 represent layout and isometric drawings of the control room, showing its relationship to the reactor enclosure.

The design basis LOCA dictates the shielding requirements for the control room. Shielding is provided to permit access and occupancy of the control room under LOCA conditions with radiation doses limited to 5 rem whole body from all contributing modes of exposure for the duration of the accident, in accordance with GDC 19.

The design basis LOCA is described in Section 15.6.5. The direct radiation from airborne fission products inside the reactor enclosure contributes less than 50 mRem to personnel inside the control room for the 30 day period following a LOCA, based on the radioactivity sources described in Section 12.2.

The assumptions used to determine control room habitability are listed in Regulatory Guide 1.3 and are discussed in Section 15.0.

For isotopes that escape from the drywell to the reactor enclosure, credit is taken for shielding by the internal structures in the reactor enclosure. Shielding credit is taken for the reactor enclosure and control structure walls. For all isotopes that remain within the drywell, shielding credit is taken for the drywell wall.

12.3.2.6 Reactor Enclosure and Refueling Area Shielding Design

During reactor operation, the steel-lined, reinforced concrete drywell wall and the reactor enclosure walls protect personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and in associated equipment within the reactor enclosure. The reactor vessel shield wall, drywell wall, and various equipment compartment walls, together with the reactor enclosure walls, reduce the radiation levels in the yard area outside the reactor enclosure to less than Zone I maximum dose rates, for those areas that are normally accessible.

Where personnel and equipment removal hatch openings or penetrations pass through the drywell wall, additional shielding, such as labyrinths or doors, is provided to attenuate the radiation to below the required level defined by the radiation zone outside the drywell wall.

Inside Drywell Structure: Areas within the drywell are Zone V and are normally inaccessible during plant operation. The reactor vessel shield provides shielding for access in the drywell during shutdown and reduces the activation of and radiation damage to drywell equipment and materials.

Outside Drywell Structure: The drywell wall reduces radiation levels in accessible areas of the reactor enclosure from sources within the drywell to below the maximum level for Zone II. Shielding requirements for the personnel access and equipment removal and CRD removal hatch openings are shown on drawings N-119 and N-134 in the areas numbered 401, 405, and 409, respectively. Shielding for these areas includes high density concrete, because neutron shielding must be considered for these large openings in the drywell wall. Drywell piping and electrical penetrations are shielded by providing either lock shields within the penetration assembly or a shielded penetration room. Shielded piping penetration room locations and bulk shielding requirements are shown on drawings N-118, N-119, N-120, N-133, N-134, and N-135. These rooms, numbered 306, 307, 309, 501, 510, 522, 523, and 599 are designated radiation Zone V during reactor power operation and are provided with personnel access controls.

Radiation shielding is provided as necessary for the following systems in the reactor enclosure and refueling area to ensure that the radiation zone and access requirements (drawings N-116, N-122, N-131, N-137, and N-139) are met for surrounding areas:

a. RWCU system

b. FPCC system

c. Neutron monitoring system

d. HPCI system

e. RCIC system

f. RHR system

g. Core spray system

h. Reactor enclosure and refueling area HVAC systems.

Main steam lines are located within shielded structures from the drywell wall to the reactor enclosure wall.

Spent fuel is a primary source of radiation during refueling. Because of the extremely high activity of the fission products contained in the spent fuel assemblies and the proximity of Zone II areas, extensive shielding is provided for areas surrounding the fuel transfer canal and pool to ensure that radiation levels remain below zone levels specified for adjacent areas.

After reactor shutdown, RHR system pumps and heat exchangers are in operation to remove heat from the reactor water. The radiation levels in the vicinity of this equipment will temporarily reach Zone V levels due to corrosion and fission products in the reactor water. Shielding is provided to attenuate radiation from RHR equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas. Adequate shielding is also provided to maintain radiation zoning requirements during the hot standby operation of the RHR system.

The concrete shield walls surrounding the spent fuel cask loading, storage, and transfer areas, as well as the shield walls surrounding the fuel transfer and storage areas, are sufficient to limit

radiation levels outside the shield walls in all accessible areas to below Zone II maximum dose rates.

Water in the spent fuel pool provides shielding above the spent fuel transfer and storage areas. Radiation levels at the fuel handling equipment are calculated to be below Zone II maximum dose rates during normal operations.

Water is also used as shielding material above the steam dryer and separator storage area. Concrete walls and water in the pool are designed to provide Zone II dose rates in adjacent accessible areas during storage of the dryer and separator.

A portable refueling shield is provided to reduce radiation dose rates in the drywell that are due to the transfer of spent fuel assemblies from the reactor vessel to the spent fuel pool. During refueling, the lead and steel shield is located in the reactor well, between the reactor vessel and the spent fuel pool, which permits continuous personnel occupancy of the drywell.

In addition, a temporary local personnel alarming rate meter is expected to be used in the drywell, with local alarms. Access controls to the upper reaches of the drywell aid in ensuring satisfactory protection for personnel.

12.3.2.7 Radwaste Enclosure Shielding Design

Radiation shielding is provided as necessary for the following systems in the radwaste enclosure to ensure that the radiation zone and access requirements (drawings N-140 through N-143) are met for surrounding areas:

a. Liquid radwaste equipment drain subsystem

b. Liquid radwaste floor drain subsystem

c. Liquid radwaste chemical waste subsystem

d. Liquid radwaste laundry drain subsystem

e. Solid radwaste system

12.3.2.8 Offgas Enclosure Shielding Design

Radiation shielding is provided as necessary for the gaseous radwaste system in the offgas enclosure to ensure that zone access requirements (drawings N-140 through N-142) are met for surrounding areas.

12.3.2.9 Diesel Generator Enclosure Shielding Design

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

12.3.2.10 Miscellaneous Plant Areas and Plant Yard Areas

Radiation shielding is provided for all radiation sources located in plant enclosures so that radiation levels at accessible areas outside are maintained below Zone I levels. Plant yard areas that are frequently occupied by plant personnel are accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public.

12.3.3 VENTILATION

The plant ventilation system provides a suitable environment for personnel and equipment during normal operation and anticipated operational occurrences. Detailed HVAC system descriptions are provided in Section 9.4. Control room habitability is discussed in Section 6.4.

12.3.3.1 Design Objectives

The systems are designed to operate so that the in-plant airborne activity levels for normal operation (including anticipated operational occurrences) in the general personnel access areas are within the limits of 10CFR20. The systems operate to reduce the spread of airborne radioactivity during normal and anticipated abnormal operating conditions.

During postaccident conditions, the ventilation system for the plant control room provides a suitable environment for personnel and equipment and ensures continuous occupancy in this area. The plant ventilation systems are designed to comply with the airborne radioactivity release limits for offsite areas during normal operation.

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

a. During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in restricted areas (as defined in pre-1994 10CFR20) of the plant are ALARA and within the limits specified in 10CFR20. The average and maximum airborne radioactivity levels in unrestricted areas (as defined in pre-1994 10CFR20) of the plant during normal operation and anticipated operational occurrences will be ALARA and within the limits of 10CFR20.

b. During normal operation and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the exclusion area boundary will be ALARA and within the limits specified in 10CFR20 and 10CFR50.

c. The dose limits of 10CFR50.67 will be satisfied following those hypothetical accidents, described in Chapter 15, that involve a release of radioactivity from the plant.

d. The dose to control room personnel shall not exceed the limits specified in GDC 19 and 10CFR50.67 following those hypothetical accidents, described in Chapter 15, that involve a release of radioactivity from the plant.

12.3.3.3 Design Guidelines

To accomplish the design objectives, the following guidelines are followed whenever practicable.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

The following design guidelines describe equipment and layout features that minimize the formation of airborne radioactivity:

a. Equipment vents and drains are piped directly to a collection device connected to the collection system instead of allowing any radioactive fluid to flow across the floor to the floor drain.

b. All-welded piping systems are used on radioactive systems to the maximum extent practicable to reduce system leakage.

c. Decontaminatable coatings are applied to the concrete floors and walls of potentially radioactive areas to facilitate decontamination.

d. Radioactive equipment has design features that minimize the potential for airborne radioactive contamination during maintenance operations. These features include flush connections on pump casings for draining and flushing the pump before maintenance and flush connections on piping systems that could handle radioactive fluids.

e. Exhaust hoods are used in the laboratories, work areas, and sample stations to facilitate processing of radioactive samples by forcing contaminants away from the personnel breathing areas and into the ventilation and filtering systems.

f. Equipment decontamination facilities are ventilated to ensure control of released radioactivity and prevent the spread of radioactive contamination.

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

h. To minimize the amount of airborne radioactivity that results from valve packing leakage, most larger valves (2½ inches and larger) are supplied and installed with a double set of packing and lantern ring in lines carrying radioactive fluids. The stuffing box is provided with a leak-off connection which is usually plugged, but may be piped through a manual stop valve to a drain header. An improved packing arrangement is being systematically provided for such valves and eliminates use of the second packing set. The exceptions for those valves that have a packing gland, which constitutes a direct communication path between primary containment atmosphere and the reactor building. These valves are being fitted with an improved packing arrangement that uses two packing sets and a lantern ring to maintain packing gland testability. Power-actuated valves, as part of the improved packing arrangement, are fitted with automatic gland adjustment capability.

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

a. The airflow is directed from areas with lesser potential for radioactive contamination to areas with greater potential for radioactive contamination.

b. In building compartments with a potential for radioactive contamination, a greater volumetric flow is exhausted from the area than is supplied to the area to minimize the amount of uncontrolled exfiltration from the area, or makeup air for the exhaust from the area is infiltrated from surrounding less contaminated areas with no direct supply to the area provided.

c. Floor and equipment drain collection tank vents are piped to a collection header and processed by the ventilation system.

d. Air is supplied to each principal enclosure via separate supply intakes and duct systems.

e. Air being discharged from potentially contaminated areas of the reactor and turbine enclosures is passed through prefilters, HEPA filters, and charcoal filters before release (drawings M-75 and M-76). Air being discharged from the radwaste enclosure is passed through prefilters and HEPA filters as shown in Figure 9.4-3. In addition, to aid in minimizing the quantity of radioactive material discharged, means are provided to isolate the reactor enclosure and the control room upon indication of high airborne activity levels.

f. Suitable primary containment isolation valves are installed in accordance with GDC 54 and GDC 56, including valve controls, to ensure that the containment integrity is maintained.

g. Only exhaust duct-work serves potentially contaminated equipment rooms. The duct-work is at negative pressure so that any leakage is directed into the duct.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment

a. Ventilation ducts are designed to minimize the leakage of radioactive contamination into or out of ducts (as applicable) and the buildup of radioactive contamination within the ducts. Internal obstructions are avoided wherever practicable. Within the reactor enclosure and control structure, welded construction of duct sections is used, and flanged and gasketed joints are used to join duct-work segments. Seismic class II duct-work in the turbine and radwaste enclosures is constructed to Sheet Metal and Air Conditioning Contractors National Association standards modified to allow no more than 4% leakage. All duct-work is 100% leak tested after installation.

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

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

testing, and inspection. Filter adsorber units comply with the access and service requirements of Regulatory Guide 1.52 and Regulatory Guide 1.140, to the extent discussed in Sections 6.5.1 and 9.4, respectively.

2. Local cooling equipment servicing the normal building requirements is located in low radiation areas whenever practicable.

c. All filter systems in which radioactive materials could accumulate to produce significant radiation fields external to the duct-work are appropriately located and shielded to reduce exposure to personnel.

d. The HVAC system is designed to allow fast replacement of components. To simplify element handling, access to active elements is direct from working platforms. Space is provided on the platforms for accommodating safe personnel movement during replacement of components, including the use of necessary material handling facilities, and during any inplace testing operations.

e. The prefilters and HEPA filters are designed with replaceable 2 foot by 2 foot units that are clamped in place against compression seals. The filter housings are designed, tested, and proven to be airtight with bulkhead-type doors that are closed against compression seals. There are two types of charcoal filters: tray-type and rechargeable-type. The tray-type charcoal filter units are designed with replaceable trays clamped in place against compression seals. There are three trays for each 2 foot by 2 foot unit. The rechargeable charcoal filter units are designed with perforated metal beds filled with bulk charcoal. The bulk charcoal in the beds can be drained and refilled.

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

g. Filters in all systems are changed based on the airflow and the pressure drop across the filter bank. Charcoal adsorbers are changed based on the residual adsorption capacity of the bed as measured by the testing of carbon samples taken from the removable canisters located in the carbon bed. The testing of the carbon adsorbers and all other components is described in Sections 6.5 and 9.4.

12.3.3.4 Design Description

Portions of the ventilation systems serving the following enclosures are assumed to be potentially radioactive and are discussed in detail in Section 9.4.

a. Turbine enclosure

b. Control structure

c. Secondary containment

d. Radwaste enclosure

e. Chemistry Laboratory Expansion

f. Offgas enclosure

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

Ventilation system design parameters are given in Tables 12.3-3 through 12.3-6.

A typical layout of a potentially radioactive filter unit is given on Figure 12.3-7.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitors

Fixed area radiation monitors are mounted throughout the plant at selected locations. Each location contains a gamma-sensitive detector, local indicator, and local alarm (visual and audio). Indicators, alarms, and recorders are located in the control room, except for the nine local area monitors (Section 12.3.4.1.3). Alarm setpoints are adjusted from the auxiliary equipment room/control room except for local area monitors.

The location and range for each monitor are given in Table 12.3-7. Ranges specified are consistent with the potential dose rates for a given area. Selection of location is based on the equipment in the area and the need for personnel access. In some locations, the detector is mounted immediately inside a cavity while associated local indicators and alarms are mounted just outside the access door. Personnel are thereby alerted to unusual radiation levels within the cavity before initiating access. Experience at PBAPS Units 2 and 3 was used in determining locations.

Area monitors located at fuel storage areas comply with the requirements of 10CFR70.24. These and area monitors in the radwaste enclosure comply with requirements of GDC 63. The area radiation monitors, with associated alarms, indicators and recorders in the main control room provide the capability to alert supervisors and personnel in that area of abnormally high and unexpected radiation levels. Subsequent notification to appropriate plant personnel could avoid inadvertent unnecessary exposure. Unexpected increases in radiation levels may be due to changes in operations, deposition of crud, or transport of radioactive material (e.g., sources) within the plant. Recorded area radiation monitor readings serve to define trends that may result from build-up, spills, or contamination of a process fluid.

Control and calibration of radiation monitoring (fixed and portable) are provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program described in Section 17.2.

12.3.4.1.1 Design Bases

The purpose of the area radiation monitoring system is to provide personnel protection in accordance with the guidelines of 10CFR20, 10CFR50, 10CFR70, and Regulatory Guide 8.8. The area radiation monitoring system has no function related to the safe shutdown of the plant or to the quantitative monitoring of the release of radioactive material to the environment. Consistent

with this purpose, the area radiation monitoring system is designed to provide the following functions:

a. Warn of excessive gamma radiation levels in areas where nuclear fuel is stored or handled, in accordance with 10CFR70.24(a)(2).

b. Provide operating personnel with a record and indication in the control room of gamma radiation levels at selected locations within the various plant structures, in accordance with Regulatory Guide 8.8.

c. Supplement other systems in detecting abnormal migrations of radioactive material in or from process streams.

d. Provide local alarms at strategic locations throughout the plant, in accordance with Regulatory Guide 8.8, where a substantial change in radiation levels or loss of sensing capability might be of immediate danger to personnel in the area.

e. Contribute supervisory information to the control room so that correct decisions can be made with respect to deployment of personnel in the event of a radiation accident.

f. Assist in the detection of unauthorized or inadvertent movement of radioactive material in the plant including the radwaste enclosure.

g. Furnish information for conducting radiation surveys.

h. Area Radiation Monitors may be removed from service for maintenance, testing, or known evolutions which would cause an ARM to alarm. Additionally an ARM may be removed from service to clear control room annunciators so other ARM alarms will not be masked. ARM's may be removed from service if the appropriate compensatory actions, if required, are taken (i.e. HP surveys, access control) when the ARM is removed from service.

The above functions are performed under the following design conditions:

h. Environmental parameters shown in Table 12.3-8 are applicable to the design of the area radiation monitoring equipment except for the monitors located inside primary containment, where the detectors are designed for a normal operating temperature range of 64°F to 151°F (up to 340°F for accident conditions) and a pressure range of atmospheric to 2 psig.

i. Noise from any source in the operating environment should not disturb the meter indication by more than ±2% of equivalent full-scale.

j. The detector/indicator and trip unit should be responsive to gamma radiation over an energy range of 0.08 MeV to 7 MeV. The energy dependence should not exceed ±20% of the indicated scale reading for a dose rate of approximately 50 mRem/hr resulting from 0.1 MeV to 3 MeV gammas.

k. At the control room, the reading should be reproducible within ±10% of the local indicated point, and drift should not exceed ±0.2% of equivalent linear full-scale for a 24 hour period or ±2% for a 30 day period.

l. The range of the monitors is shown in Table 12.3-7. The ranges selected ensure readout of both the highest and lowest anticipated radiation levels, in accordance with Regulatory Guide 8.8.

m. Operate without effective change in performance when the ac supply voltage changes over a range of ±10% from nominal value or has a frequency variation of ±5% from nominal value.

n. Operate so that interruption or failure of the ac power supply will result in actuation of the trip circuit to produce an alarm, in accordance with Regulatory Guide 8.8.

12.3.4.1.2 Design Details

The area radiation monitoring system is shown in the typical functional block diagram of Figure 12.3-27. Each channel consists of a combined sensor/converter unit, a local auxiliary unit (readout with visual and audible alarm), a combined indicator/trip unit, a shared power supply, and a shared multipoint recorder. The locations of the radiation sensors are indicated in Table 12.3-7. Further design details of the area radiation monitoring system are as follows:

a. Each indicator and trip unit is provided with one upscale trip continuously adjustable over the entire scale and one downscale trip adjustable over the lower decade. Provision is made to permit the upscale trip to be set and checked for accuracy with respect to the indicator.

b. Detectors are wall-mounted and suitable for operation in the anticipated plant environment with no additional protection.

c. Panel locations of the remote equipment are shown in Table 12.3-9.

d. The following features are provided for components located in the auxiliary equipment room:

    1. Radiation level indicator (meter)

    2. High radiation alarm light (amber)

    3. Downscale alarm light (white)

    4. Alarm reset (push button)

    5. Meter zero adjust (on the amplifier)

    6. Alarm level adjust

    7. Trip check push button

    8. Power supply switch and "power-on" light (clear lens)

    9. Indicators to show power supply voltages

    10. Annunciator outputs

11. Recorder outputs

e. The radiation monitors are calibrated at regular time intervals in accordance with station procedures. Calibration methods are covered in detail in the equipment procedures manual.

f. The following annunciators are located in the control room to alert the operator:

1. Reactor enclosure area, high radiation (Units 1 and 2)
2. Refueling floor area, high radiation (Units 1 and 2)
3. Turbine enclosure area, high radiation (Units 1 and 2)
4. Turbine enclosure common area, high radiation
5. Radwaste enclosure common area, high radiation
6. Reactor enclosure common area, high radiation
7. Admin. bldg. area, high radiation
8. Unitized area, Rad monitors downscale
9. Common area, Rad monitors downscale

12.3.4.1.3 <u>Local Area Monitors</u>

In addition to the area radiation monitors described above, ten local area monitors are provided, located on each of the two refueling bridges, the three turbine enclosure crane cabs, the east and west unit 1 and 2 turbine enclosure deep bed vessel areas, and the health physics and chemistry source storage and calibration room. The essential differences between these monitors and those area monitors described above are as follows:

a. No outputs to the control room are provided.

b. Alarms are local only.

c. No recorders are provided.

d. Local power (battery) packs are provided in the event of external power cutoff, except for the local area radiation monitors in the health physics and chemistry source storage and calibration room, and east and west Unit 1 and 2 turbine enclosure deep bed vessel areas.

The power for the turbine enclosure cranes is turned off except when the cranes are being used or tested. The battery backup for the crane monitors is not available when the crane power is off.

A portable alarming rate meter will be used in the drywell to warn plant personnel if a core component is dropped during fuel transfer operations. The meter will have local alarms to ensure adequate personnel warning. Administrative controls will be used to prevent access to the upper drywell and unnecessary high exposures to personnel.

12.3.4.1.4 Postaccident Area Radiation Monitors

In the event of an accident, access will be required to certain locations outside secondary containment to perform sampling, analysis, and monitoring tasks. The identified areas include the counting room and chemistry laboratory, the main control room, the turbine enclosure near the control room exit, the north stack instrument room, the postaccident sampling station. The operational support center utilizes a portable arm. The area monitors associated with these locations have been designated as postaccident area radiation monitors. In addition to having dose rates available through the shared multipoint recorder in the control room, instantaneous and stored data are available on demand for display and trending in the control room, TSC via the RMMS computer system. Data from PMS is transmitted to the EOF via the emergency plan display system (EPDS). The ranges of these area monitors are specified in Table 12.3-7 and incorporate the highest anticipated postaccident dose rates for these locations. The design details and operating conditions are as described in Sections 12.3.4.1 and 12.3.4.2. Compliance with Regulatory Guide 1.97 (Rev 2) is discussed in Section 7.5.

12.3.4.2 Airborne Monitoring

Airborne radioactivity monitoring is accomplished by use of the following:

a. Monitoring ventilation ducts in key locations throughout the plant.

b. Continuous air monitors which are portable, cart-mounted, and monitor particulate activity.

c. High and low volume portable samplers capable of attaching filters and charcoal cartridges for particulate and iodine monitoring.

The ventilation system monitors are located at positions which provide representative air concentrations and a rapid indication of abnormal conditions. Those systems which require HEPA filtration have monitors upstream of the filters. Both the in-line GM tube and beta scintillator, and off-line particulate, iodine, and noble gas monitoring configurations can be utilized. Readout and annunciation are provided in the main control room. The off-line monitors provide the capability to obtain particulate and iodine samples for isotopic analysis. Emergency dc power is provided in the event of a LOOP. The detectors are calibrated routinely and after any maintenance work is performed on the detector.

CAMs are located in freely accessible areas where airborne radioactivity is most likely to exist. These CAMs are mobile and can be moved from area to area as deemed necessary by plant conditions or maintenance operations. CAMs incorporate either fixed or movable filters for the collection of particulate activity, which is monitored directly by a detector. Readout is recorded in counts per minute. The filters can be removed for further analysis using counting room instrumentation. Audible and visual alarms indicate when setpoint levels have been exceeded. The detectors are calibrated routinely and after any maintenance work is performed on the detector.

The CAM's primary function is to indicate trends and sudden changes in airborne activity. Typical locations are solid waste handling areas, spent fuel pool areas, and the reactor operating floor and turbine building. The monitoring system is capable of detecting particulate, iodine and noble gas radioactivity. A flexible hose can be attached to the monitor intake and inserted into a cavity

or work area to detect the presence of localized airborne activity. Conformance to Regulatory Guide 8.2 is discussed in Section 12.5.1.

Ventilation monitors and CAMs are used as trending devices and will indicate areas and times needing special samples taken.

Alarm setpoints are set at low levels to ensure close respiratory controls. CAMs, however, cannot account for inversion conditions or properly identify isotopic content of the air. When a setpoint is reached, the monitor alarms. The reason for the alarm is evaluated and when necessary, grab air samples are taken and analyzed in the counting lab. The DAC hours are isotopically calculated by the computer program or are done manually. Appropriate actions can then be taken based on accurate data.

Potentially airborne accessible areas are air sampled at regular intervals. The survey/sampling frequency is designed for that area by the routine survey program.

Low and high volume samples with filter paper or charcoal cartridges are used. These are described in more detail in Section 12.5.3.1.3.

Airborne monitoring provides the information necessary to determine stay times in given areas and applicable respiratory equipment. The information is also of value in identifying process system leakage. Such monitoring is conducted in accordance with the guidelines of Regulatory Guide 1.21.

Control and calibration of radioactivity monitoring (fixed and portable) are provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program described in Section 17.2.

Qualification and training of health physics and chemistry personnel will follow ANSI/ANS 3.1 (1978) guidance.

12.3.5 REFERENCES

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Table 12.3-1

PLANT RADIATION ZONE DESCRIPTION

DESIGNATION	MAXIMUM DESIGN DOSE RATE (mRem/hr)	DESCRIPTION
I	≤ 0.5	No radiation sources, no radiological control required
II	≤ 2.5	Low radiation sources, radiological control required
III	≤ 15	Low to moderate radiation sources, radiological control required
IV	< 100	Moderate radiation sources, radiological control required
V	≥ 100	High radiation sources, radiological control required

Table 12.3-2

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

GRACE 1	Multigroup, multiregion, gamma ray attenuation code used to compute gamma heating and gamma dose rates in slab geometry (Reference 12.3-6).
GRACE 2	Multigroup, multiregion, gamma ray attenuation code used to compute the dose rate or heat generation rate for a spherical or a cylindrical source with slab or concentric shields (Reference 12.3-7).
ANISN	Multigroup, multiregion code solving the Boltzmann transport equation for neutrons and gamma rays in one-dimensional slab, cylindrical, or spherical geometry (Reference 12.3-8).
SDC	Multigroup, multiregion, gamma ray attenuation code that calculates dose rates for 13 geometry options (Reference 12.3-9).
QAD	Multigroup, multiregion, three-dimensional, point kernel code that calculates fast neutron and gamma ray dose and heat generation rates (Reference 12.3-10).
NAP	Determines neutron activation and gamma emission source strengths as a function of neutron exposure and decay time (Reference 12.3-11).
MORSE-CG	Three-dimensional Monte Carlo neutron and gamma ray general transport code (Reference 12.3-12).
DOT 3	Two-dimensional neutron, gamma ray, discrete ordinate, transport code (Reference 12.3-13).
ORIGEN	Isotopic generation and depletion code that solves equations of radioactive growth and decay for isotopes of arbitrary coupling (Reference 12.3-14).
G-33	A general purpose gamma ray scattering code (Reference 12.3-15).

These programs are as used in original plant design. Updated versions or alternative programs applying the same basic industry standard analysis methodology may be used in evaluation of plant design changes.

Table 12.3-3

TURBINE ENCLOSURE VENTILATION SYSTEM DESIGN FEATURES

AREA	RADIOLOGICAL SAFETY FEATURES	SUPPLY/EXHAUST AIR FLOW RATE (cfm)
General Personnel Access Areas	Three 50% supply fans, three 50% exhaust fans	250,000/200,000
Equipment Areas	Two 100% exhaust fans, continuously filtered	0/63,000

Table 12.3-4

CONTROL STRUCTURE VENTILATION SYSTEM DESIGN FEATURES

OPERATION MODE	RADIOLOGICAL SAFETY FEATURES	AIR FLOW RATE (cfm)	EXPOSURE TO AIRBORNE CONCENTRATIONS
Normal	Two 100% supply and return fans	26,200	Background
Accident	Two 100% supply fans. Automatic/ manual switch to emergency intake and filtering and recirculation system on high activity signal; recirculation and filtering on high chlorine signal	3,000 maximum	Less than allowable limits set in 10CFR20

Table 12.3-5

REACTOR ENCLOSURE AND REFUELING AREA VENTILATION SYSTEMS DESIGN FEATURES

AREA	RADIOLOGICAL SAFETY FEATURES	SUPPLY/EXHAUST AIR FLOW RATE (cfm)
Reactor Enclosure General Personnel Access Areas	Three 50% supply fans, three 50% exhaust fans. No or low activity exhaust is not filtered; enclosure isolation on high activity signal. Automatic switch to recirculation and SGTS.	180,000/140,000 Emergency: 60,000 recirculation 3,000 maximum SGTS
Reactor Enclosure Equipment Areas	Two 100% exhaust fans, two 100% filters. Automatic switch to recirculation and SGTS.	0/40,000[(1)]
Refueling Area General Personnel Access Areas	Three 50% supply fans, three 50% exhaust fans. No or low activity exhaust is not filtered; area isolation on high activity signal. Automatic switch to SGTS.	54,000/54,000 3,000 maximum SGTS

(1) Transfer air flow from personnel access areas

Table 12.3-6

RADWASTE CHEMISTRY LABORATORY AND OFFGAS ENCLOSURES VENTILATION SYSTEM DESIGN FEATURES

AREA	RADIOLOGICAL SAFETY FEATURES	SUPPLY/EXHAUST AIR FLOW RATE (cfm)
Equipment Areas	Two 100% supply fan cabinets, two 100% supply fans, and two 100% exhaust fans; all exhaust are passed through HEPA filters.	0/53,000
Service and Control Areas	Same as equipment areas.	0/2,850
Fume Hoods	*Two 100% supply fan cabinets and two 100% exhaust fans; exhausts are passed through HEPA filters.	5510/11200
General Personnel Access Areas	Two 100% supply fan cabinets, two 100% supply fans, and two 100% exhaust fans.	73,500/16,490
Common Tank Vent	One 100% exhaust fan.	0/385

* Located in the Penthouse of the Chemistry Lab Building

[redacted]

Table 12.3-7 (Cont'd)

CHANNEL NUMBER[4]	SENSOR NUMBER	DESCRIPTION	ENCLOSURE[1]	LOCATION[2]	ELEVATION (feet)	RANGE (mRem/hr)
49	RE49-M1-0N001	Radwaste cask loading area	RWE	C-11	217	$.01\text{-}10^4$
50	RE50-M1-0N001	Railroad car airlock	RA	D-23	217	$01\text{-}10^4$
51	RE51-M1-0N001	Radwaste enclosure hallway	RWE	G-11	217	$.01\text{-}10^4$
52	RE52-M1-0N001	Hot maintenance shop	AB	J-34	217	$.01\text{-}10^4$
53	RE53-M1-0N001	Entrance, turbine enclosure railroad	TE	R-23	217	$.01\text{-}10^4$
54	RE54-M1-0N001	Radwaste building, el 239'	RWE	F-10	239	$.01\text{-}10^4$
55	RE55-M1-0N001	Radwaste exhaust fan area	RWE	G-10	257	$.01\text{-}10^4$
56	RE56-M1-0N001	Control room	CS	J-23	269	$.01\text{-}10^4$
57	RE57-M1-0N001	Turbine area operating floor/operational support center	TE	N-23	269	$.01\text{-}10^4$
58	RE58-M1-0N001	Standby gas treatment filter room	CS	M-25	332	$.01\text{-}10^4$
60	RE60-M1-0N001	North stack instrument room	RE	H-21	411	$.01\text{-}10^4$
-	RIAH-TA-025[3]	(Local) Source storage & calibration room	RWE	D-10	191	$.1\text{-}10^4$
-	RIAH-TA-001[3]	(Local) A turbine enclosure crane	TE	variable	310	$.01\text{-}10^4$
-	RIAH-TA-002[3]	(Local) B turbine enclosure crane	TE	variable	310	$.01\text{-}10^4$
-	RIAH-TA-101[3]	(Local) refueling platform	RA	variable	352	$.01\text{-}10^4$
-	RIAH-TA-201[3]	(Local) refueling platform	RA	variable	352	$.01\text{-}10^4$
-	RIAH-TA-102[3]	(Local) turbine deep bed vsl area	TE	M-7	217	$.01\text{-}10^2$
-	RIAH-TA-103[3]	(Local) turbine deep bed vsl area	TE	M-11	217	$.01\text{-}10^2$
-	RIAH-TA-202[3]	(Local) turbine deep bed vsl area	TE	M-38	217	$.01\text{-}10^2$
-	RIAH-TA-203[3]	(Local) turbine deep bed vsl area	TE	M-35	217	$.01\text{-}10^2$

(1) RE Reactor enclosure
RA Refueling area
TE Turbine enclosure
RWE Radwaste enclosure
CS Control structure
AB Administration building

(2) Locations indicate east-west and north-south column lines (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).

(3) Local monitor only - requires local power supply.

(4) When the same channel number is given twice, the sensor number ending in 1N001 refers to a Unit 1 sensor, and 2N001 refers to a Unit 2 sensor.

Table 12.3-8

AREA RADIATION MONITORING ENVIRONMENTAL PARAMETERS

ENVIRONMENTAL PARAMETER	DESIGN CENTER	MONITOR RANGE	DETECTOR, PREAMPLIFIER RANGE
Temperature	77°F	41°F-122°F	32°F-140°F
Relative Humidity	50%	20%-90%	20%-100%
Pressure	Atmospheric	Atmospheric	Atmospheric

Table 12.3-9

REMOTE EQUIPMENT PANEL LOCATIONS

EQUIPMENT	PANEL NOS.	
	Auxiliary Equipment Room	Control Room
Common equipment:		
Indicator and trip units	00C643	-
Recorders	-	00C624
Annunciators	-	00C824
Unitized equipment:		
Indicator and trip units	10C605, 20C605	
Recorders	-	10C600, 20C600
Annunciators	-	10C800, 20C800

12.4 DOSE ASSESSMENT

This section discusses the estimated radiation exposures both in-plant and at locations outside the plant structures. Sections 12.4.1 and 12.4.2 discuss direct radiation and airborne radiation exposures within the plant; Section 12.4.3 is concerned with exposures outside the plant structures; and Section 12.4.4 estimates the exposure to Unit 2 construction workers during the operation of Unit 1. The ISFSI radiation exposures are not considered here, as they are addressed in a separate 10CFR72.212 document.

Not all the methods for occupational radiation dose assessment discussed in Regulatory Guide 8.19 are employed for LGS. Alternative methods are described in Section 12.4.1.

12.4.1 Direct Radiation Dose Estimates For Exposures Within The Plant

To estimate the total annual man-rem dose from direct radiation to personnel within the plant, seven broad categories or job functions were defined, and the annual man-rem dose for each category was evaluated. Where the functions and expected radiation levels are predictable or clearly defined, analytical methods were employed for the man-rem estimates. In other cases, the estimate basis is historical exposure data from operating BWR power plants. Section 12.4.1.1 provides the definitions and components of each of the seven broad categories, and Section 12.4.1.2 describes briefly the estimation techniques used.

The resultant dose estimates are contained in Section 12.4.1.3, along with further discussion of the factors involved and the methodology used for each category and its related components.

12.4.1.1 Definition of Categories Used in Exposure Estimates

Seven broad categories were used in estimating the total annual man-rem dose. These categories are:

Routine Operations: This category is composed of the following three components or subcategories:

a. Routine patrols and surveillances of the reactor enclosure, turbine enclosure, control structure, and radwaste and offgas enclosures.

b. Periodic tests and checks in the reactor enclosure, turbine enclosure, control structure, and radwaste and offgas enclosures.

c. Control room operations; specifically, the dose received by operators in the control room and radwaste control room.

Routine Maintenance: This includes all scheduled maintenance. This does not imply that a particular date has been established, but rather that the maintenance is planned and occurs at least annually. This category also includes the preventative maintenance performed in the radiation areas of the reactor, turbine, radwaste, and offgas enclosures.

Inservice Inspections: These are inspections normally performed by NDT personnel and outside contractors. Such inspections normally occur during outages on piping and systems that cannot be checked while at power.

Special Maintenance: All maintenance that is not scheduled. This maintenance is not planned in advance and normally cannot be predicted.

Radwaste Processing: This includes any work with solid or liquid radwaste: movement of casks and liners; radwaste, condensate system, or fuel pool filter changes; resin moving; baling of low level trash, etc. Maintenance of radwaste equipment is covered by the maintenance categories and is not included in this job function.

Refueling: This is all work with fuel or reactor components performed in the reactor and pool area.

Health Physics: This covers all health physics activities and includes chemistry operations and sample collection.

12.4.1.2 Exposure Estimate Methodology

Analytical Method: The analytical method used for man-rem estimation is based upon the product of estimated exposure time and estimated ambient dose rate. Estimates of the occupancy time requirements for operations associated with equipment in plant radiation areas (e.g., maintenance time or surveillance time) are first developed. An applicable frequency of occurrence is then factored in to determine the annual exposure time for each operation. Areas with no significant radiation sources are not included in the exposure estimate. Where radiation sources are present, the design maximum dose rate of 2.5 mRem/hr is assumed for some Zone II areas (radiation zones are defined in Table 12.3-1). Similarly, 15.0 mRem/hr is assumed for some Zone III areas. All other estimated dose rates are based on either calculations or actual radiation levels encountered at operating plants. Radiation levels encountered or estimated are adjusted in proportion to power. The analytical method was used in determining the exposure estimates for the routine maintenance and routine operations categories.

Historical Method: In the historical method, the annual man-rem is estimated from the exposures received at operating BWR power plants. This method was used for all the remaining categories (special maintenance, inservice inspection, radwaste processing, refueling, and health physics). The data sources are the annual and semiannual BWR operating reports and plant correspondence with regulatory agencies. Included are a total of 61 reactor years of operation for 16 nuclear units, which are listed in Table 12.4-1. The average licensed power level of these units is 747 MWe with the smallest rated at 514 Mwe. The data were collected and assembled using the following guidelines:

a. No data before the first calendar year which contained at least nine months of commercial operation are used.

b. In multiple unit plants, each unit is assumed to contribute equally to the annual exposures.

c. If exposure contributions from two or more job functions cannot be separated, a conservative approach is taken by assigning all the exposure to one function and by making no entry in the data base for the other.

Table 12.4-2 contains the results of the historical data compilation and includes both the number of reactor years contributing to and the standard deviations associated with each job function.

The large standard deviations, which range from about 60% to 160% of the mean values, are indicative of the wide spread of data that have been reported within each exposure category.

12.4.1.3 Results of Annual Direct Radiation Dose Estimates

The annual man-rem estimates for each category and subcategory are detailed below in Sections 12.4.1.3.1 through 12.4.1.3.7. The methods used in their determination are as described previously, with any additional assumptions or required information included below.

In each of the following sections, the annual exposure estimates are reported for two plant configurations: single-unit operational, and two units operational. In general, the "two-unit dose" is twice the "single-unit dose"; however, the exposures associated with certain job functions are assumed to be independent of the number of units in operation since the functions are performed regardless of whether one or two units are operational. These specific job functions are:

a. Control room operations

b. Radwaste control room operations

c. Radwaste and offgas enclosure routine surveillances

d. Radwaste and offgas enclosure periodic testing

e. Radwaste and offgas enclosure routine maintenance

For these estimates, the single-unit dose is assumed to be the same as the two-unit dose.

A summary of the direct radiation dose estimates is given in Section 12.4.1.3.8.

12.4.1.3.1 Routine Operations Dose Estimate

During normal operations, routine patrols and surveillances are performed by plant operators. The majority of items checked are rotating equipment (pumps, fans, etc), and each is viewed to verify the absence of leaks, excessive vibrations, or other abnormal conditions. For the surveillance man-rem exposure estimation, the following assumptions were made:

a. Dose rates are estimated as outlined in Section 12.4.1.2. Additionally, because of the high potential dose rates associated with certain equipment, routine surveillances of such equipment are performed from a remote location (such as the equipment cell doorway) and credit is taken for the lower ambient radiation level at that point.

b. Each patrol consists of only one person.

The results of the routine surveillance exposure estimate are contained in Tables 12.4-3 through 12.4-5.

Similarly, the details and results of the exposure estimate for the periodic testing subcategory are also contained in Tables 12.4-3 through 12.4-5. Since periodic testing is assumed to occur during

equipment shutdown, the estimated shutdown dose rates are therefore used for determining the associated exposures.

The remaining subcategory is control room operations exposures. Exposures have been estimated from the anticipated control room radiation levels and the staffing requirements for the main and radwaste control rooms. It is assumed that the staffing levels of both control rooms are identical for either one or two units operational. Table 12.4-6 contains the details of the control room operations exposure estimate.

The total annual exposure estimate for the routine operations category is the sum of the three subcategory annual exposures as shown in Table 12.4-7 and summarized below:

Annual Exposure Estimate: Routine Operations

42.6 man-rem (single-unit operational)

68.9 man-rem (two units operational)

12.4.1.3.2 Routine Maintenance Dose Estimate

A detailed review of plant radiation areas was performed to produce a listing of the types and quantities of selected equipment present in each area. Next, total annual maintenance man-hours were estimated for each equipment type identified based on a combination of operating experience and engineering judgement. Total estimated man-hours are shown in Table 12.4-8 and are intended to include all expected routine activities for each equipment type such as valve repacking, valve relapping, pump seal replacement, fan overhaul, etc.

In any area, the total annual man-hours for routine maintenance is then the summation of the quantity-man-hour products for all equipment types found in the area. Multiplying the area's annual maintenance man-hours by the anticipated area dose rate produces the estimated man-rem by area. As with periodic testing, the estimated shutdown dose rate was used for estimating maintenance exposures.

A "total annual" maintenance approach was used for each component since currently available data generally does not contain sufficient information to provide a basis for man-hour breakdowns by maintenance activity. In addition, the area-by-area methodology employed makes estimate compilations by system unnecessary, since locations where high man-rem expenditures are expected are clearly indicated. Tables 12.4-3 through 12.4-5 contain the details of the routine maintenance exposure estimate for each enclosure. The sum from the three enclosures is presented below:

Annual Exposure Estimate: Routine Maintenance

232.5 man-rem (single-unit operational)

427.7 man-rem (two units operational)

12.4.1.3.3 Inservice Inspection Dose Estimate

The annual exposure estimate for inservice inspection is based upon the data from operating BWRs given in Table 12.4-2, and is:

Annual Exposure Estimate: Inservice Inspection

27.5 man-rem (single-unit operational)

55.0 man-rem (two units operational)

12.4.1.3.4 Special Maintenance Dose Estimate

The annual exposure estimate for special maintenance is based upon the data from operating BWRs given in Table 12.4-2, and is:

Annual Exposure Estimate: Special Maintenance

273.1 man-rem (single-unit operational)

546.2 man-rem (two units operational)

12.4.1.3.5 Radwaste Processing Dose Estimate

Most of the operations in the plant associated with the radwaste processing category are performed remotely and are therefore not suitable for evaluation by the analytical estimation technique. Consequently, the annual man-rem estimate for radwaste processing is more properly taken from the historical BWR operating data of Table 12.4-2, since this provides a conservative estimate of the anticipated exposure.

Annual Exposure Estimate: Radwaste Processing

37.0 man-rem (single-unit operational)

74.0 man-rem (two units operational)

12.4.1.3.6 Refueling Dose Estimate

The annual exposure estimate for refueling is based upon the data from operating BWRs given in Table 12.4-2, and is:

Annual Exposure Estimate: Refueling

19.2 man-rem (single-unit operational)

38.4 man-rem (two units operational)

12.4.1.3.7 Health Physics Dose Estimate

The annual exposure estimate for health physics monitoring is based upon the data from operating BWRs given in Table 12.4-2, and is:

Annual Exposure Estimate: Health Physics

29.3 man-rem (single-unit operational)

58.6 man-rem (two units operational)

12.4.1.3.8 Summary of Direct Radiation Dose Estimates

The annual dose estimates in the preceding seven sections are summarized and totaled in Table 12.4-9. As shown in this table, the estimate of total annual in-plant exposure from direct radiation is:

Annual Exposure Estimate: Total

661.2 man-rem (single-unit operational)

1,268.7 man-rem (two units operational)

Dose estimates for inservice inspection, special maintenance, radwaste processing, refueling, and health physics are based on historical data from operating facilities. Any further breakdown of the dose estimate (such as was made for routine operations and routine maintenance) for these types of activities would still rely primarily on the historical information available. The resultant dose estimate would therefore not be any more precise than an estimate based solely on reported radiation exposures.

In all five areas where historical data is used in the dose estimate, the LGS design includes design features which will reduce actual exposures received by plant personnel. However, due to the lack of sufficiently detailed information to allow the precise quantification of the dose reduction, the calculation of the reduction was not attempted.

As an example of design features which will result in dose reduction, the following design features have been incorporated to facilitate inservice inspection:

a. Quick removal insulation around the RPV nozzles

b. Access panels in the shield wall to the RPV bottom head welds

c. The use of a remote, trackless vehicle for RPV weld inspection

d. The use of remote automatic weld inspection of the RPV nozzle welds

12.4.2 AIRBORNE RADIOACTIVITY DOSE ESTIMATES FOR EXPOSURES WITHIN THE PLANT

The estimated exposures to plant personnel from airborne radioactivity are based upon the source distributions and radionuclide concentrations presented in Section 12.2.2 and Tables 12.2-93 through 12.2-101. Because of the limited geometry afforded by the finite compartment sizes within the plant, personnel exposures due to noble gas immersion are expected to be insignificant when compared to inhalation exposures and therefore have not been estimated.

In order to determine whether exposure contributions from airborne radioactive particulates are significant, an evaluation was made in each area of the ratio of total particulate maximum permissible concentration fractions to total radioiodine MPC fractions (which is equivalent to the

ratio of particulate MPC-hours to iodine MPC-hours). For the turbine and reactor enclosure areas, the particulate-to-iodine ratios are approximately 0.02 and 0.05, respectively, indicating that the particulate inhalation exposures are not significant in those areas. In the radwaste enclosure areas, however, the particulate-to-iodine ratio is approximately 1.11. Since over 75% of the total particulate MPC fraction is attributable to Co-60, both the thyroid inhalation dose due to radioiodines and the lung inhalation dose due to Co-60 were estimated for the radwaste enclosure (the thyroid and the lung are the critical organs for iodines and Co-60, respectively).

In addition to the inhalation exposures due to radioiodines (all enclosures) and to Co-60 (radwaste enclosure), whole body exposures due to airborne tritium were also estimated for those areas where tritium is assumed to be present (turbine and reactor enclosures).

Tables 12.4-10 through 12.4-12 contain the compilations of the estimated annual occupancy times and the estimated annual exposures for each of the areas identified in Section 12.2.2 as potential sources of airborne radioactivity. The occupancy times are based upon detailed reviews of each area and the determination of the operations which might occur in those areas. The exposures are based upon the estimated airborne concentrations in Tables 12.2-98 through 12.2-100, dose factors from table C-1 of Regulatory Guide 1.109, and an assumed breathing rate of 3.47×10^{-4} m^3/sec.

12.4.3 Exposures At Locations Outside Plant Structures

The radiation exposures at locations outside the plant structures were estimated for two areas: the site boundary and the visitor's center. Section 12.4.3.1 discusses direct radiation exposure at these locations, and Section 12.4.3.2 deals with airborne exposures.

12.4.3.1 Direct Radiation Dose Estimates Outside Plant Enclosures

At locations outside plant enclosures, the direct radiation exposure has two principal components:

a. Sources of activity stored outside the enclosures, specifically, the refueling water storage tanks and the CST.

b. Turbine shine due to the N-16 present in the reactor steam.

Based on the calculated surface dose rates for the refueling water storage tanks and CST given in Section 12.2.1.7, the dose contribution at locations outside the plant enclosures due to these tanks is considered negligible.

The N-16 present in the reactor steam in the primary steam lines, turbines, and moisture separators provides a dose contribution to locations outside the plant enclosure as a result of the high energy gamma rays that it emits as it decays. To reduce the turbine shine doses, radiation shielding is provided around each turbine train.

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. Thus, areas where steam piping exists will experience a greater level of radiation due to the increased N-16. The increased exposure is within the estimates provided in Chapter 11 and 12.

Exposure at locations outside plant structures will also increase due to direct radiation and turbine shine from the N-16, which is a high energy gamma emitter. In order to minimize these effects, additional radiation shielding has been installed around the high pressure turbine and combined intermediate valves.

The resultant annual exposure due to turbine shine was calculated with the SKYSHINE computer program (Reference 12.4-1). Point sources are used to represent the components on the turbine deck and the source strengths are given in Table 12.4-13.

With an assumed 100% occupancy factor and an 80% capacity factor, the maximum calculated dose rate occurs at the northeast site boundary (Figure 12.4-1 and Table 12.4-14) and is 5.5 mRem/year.

The dose rate in the visitor's center is calculated by the SKYSHINE program to be 1.53×10^{-3} mRem/hr with two units operational. Assuming a visitor stays at the visitor's center one day a year for eight hours, the estimated dose for the visitor is 1.2×10^{-2} mRem/year, as shown in Table 12.4-14.

12.4.3.2 Airborne Radioactivity Dose Estimates Outside Plant Enclosures

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

12.4.4 EXPOSURES TO CONSTRUCTION WORKERS

12.4.4.1 Direct Radiation Dose Estimates

The estimated dose rates from direct radiation and turbine shine received by construction workers on Unit 2 due to the operation of Unit 1 are well within the limits of 10CFR20 for exposure to individuals in unrestricted areas. Access to Unit 1 for Unit 2 workers is restricted by permanent concrete walls, security fences or chains, etc.

The estimated dose rates are the sums of the direct radiation from the Unit 1 reactor enclosure, turbine enclosure, radwaste enclosure, and offgas enclosure; and the turbine shine doses resulting from the decay of N-16 in the steam lines and turbine equipment of Unit 1. As discussed in Section 12.4.3.1, dose contributions from outside storage tanks are considered negligible and are not included in the exposure estimate.

The annual dose from Unit 1 operation has been estimated for various points in the Unit 2 construction area. The results of this estimate are listed in Table 12.4-15, and their corresponding points are shown on Figure 12.4-1.

The doses from turbine shine were calculated with the SKYSHINE computer program in the manner described in Section 12.4.3.1. The resultant dose includes the direct as well as air-scattered contribution. Credit is taken for the shielding which is afforded by the Unit 2 walls and floor slabs. The radioactive wastes are processed and stored in the radwaste enclosure, where shielding is provided to ensure that the dose outside the enclosure is less than 0.5 mRem/hr. With an allowance for distance between the radwaste enclosure and the Unit 2 construction area, the estimated direct shine dose is less than 0.01 mrem/hr under normal operating conditions.

The exposure for Unit 2 construction workers has been estimated based on the following assumptions:

a. The current construction schedule is met.

b. Personnel assigned to the control structure or Unit 2 turbine enclosure are assumed to work in the turbine enclosure only. Personnel assigned to the Unit 2 reactor enclosure or drywell are assumed to work in the reactor enclosure only.

c. The average dose rate in the yard areas is the average of the dose rates at points 1 through 5 of Figure 12.4-1, 0.0335 mRem/hr. The average dose rate in the field office is 0.0215 mRem/hr. Each of these dose rates includes 0.01 mRem/hr for the direct shine contribution and is for 100% plant capacity.

d. The capacity factor for Unit 1 in determining total estimated exposure is 80%.

e. 40 hours per week per person at the work site, 50 weeks per year.

Exposure to personnel in various categories and locations is summarized in Table 12.4-15, which gives the total estimated exposure to Unit 2 construction workers as 41.16 man-rem.

10CFR20.202 specifies that personnel monitoring equipment is required if the maximum expected dose per calendar quarter for workers in an area exceeds one-fourth of the 1250 mRem/quarter limit. It is determined that, even in areas with the highest radiation levels (the turbine deck), no construction worker would receive a dose greater than this, so personnel monitoring equipment is not necessary. However, periodic radiation surveys are to be made by the radiation protection staff.

12.4.4.2 <u>Exposures Due to Airborne Radioactivity</u>

Based on expected annual releases of gaseous effluents (Table 11.3-1), a worst location annual average atmospheric dispersion factor of 1.48×10^{-5} sec/m^3, and an annual occupancy of 2000 hr (40 hr/week for 50 weeks/year) the total body gamma, beta skin, and thyroid gamma dose rates from airborne radionuclides to a construction worker are estimated to be 2.47 mRem/year, 4.77 mRem/year and 0.57 mRem/year, respectively, during the construction stage. Dose rates at specific receptor points are presented in Table 12.4-16. A comparison of these values with those in Table 12.4-14 shows that doses to construction workers resulting from direct shine are dominant.

12.4.5 REFERENCES

12.4-1 M.G. Wells, D.G. Collins, R.B. Small and J.J. Newell, "SKYSHINE, A Computer Procedure For Evaluation Effect of the Structure Design on N-16 Gamma Ray Dose Rates", RRA-T7209, (November 1, 1972).

Table 12.4-1

SUMMARY OF HISTORICAL DATA USED IN COMPILATION OF
EXPOSURES RECEIVED AT OPERATING BOILING WATER REACTORS

					NUMBER OF OPERATING YEARS ACTUALLY CONTRIBUTING TO DATA BASE							
PLANT	UNIT	NET MWe[1]	DATE OF COMMERCIAL OPERATION[1]	TOTAL OPERATING YEARS[2]	ROUTINE OPERATIONS	ROUTINE MAINT	INSERVICE INSPECTION	SPECIAL MAINT	WASTE PROCESSING	REFUELING OPERATIONS	HEALTH PHYSICS	ANNUAL TOTAL
Brunswick	2	821	11/75	1	1	1	1	1	1	1	1	1
Cooper	-	778	7/74	2	2	2	2	2	2	2	2	2
Dresden	2	809	8/70	6	4	4	0	4	4	4	3	6
Dresden	3	809	10/71	5	4	4	0	4	4	4	3	5
Duane Arnold	1	538	1/75	2	1	1	1	1	1	1	1	2
Edwin I. Hatch	1	786	5/75	1	1	1	1	1	1	1	1	1
Millstone	1	690	3/71	6	3	3	1	3	1	1	2	6
Monticello	-	545	6/71	5	4	4	2	4	3	4	2	5
Nine Mile Point	1	610	12/69	7	5	5	1	5	1	5	5	6
Oyster Creek	1	650	12/69	7	7	7	3	4	3	2	6	7
PBAPS	2	1065	7/74	2	2	1	1	1	1	2	2	2
PBAPS	3	1065	12/74	2	2	1	1	1	1	2	2	2
Pilgrim	1	655	12/72	4	3	3	2	3	2	3	2	4
Quad-Cities	1	809	8/72	4	4	4	0	4	4	4	3	4
Quad-Cities	2	809	10/72	4	4	4	0	4	4	4	3	4
Vermont Yankee	-	514	11/72	4	2	2	2	2	2	2	2	4
			Totals:	62	49	47	18	44	35	42	40	61

(1) Source - "U.S. Central Station Nuclear Electric Generating Units: Significant Milestones", ERDA 77-30/1, (January 1, 1977).
(2) Total number of operating years available to the data base through the end of 1976 beginning with the first calendar year which includes at least nine months of commercial operation

Table 12.4-2

OCCUPATIONAL EXPOSURES BY JOB FUNCTION FOR
OPERATING BOILING WATER REACTORS

	NUMBER OF REACTOR YEARS AVERAGED	AVERAGE MAN-REM PER REACTOR YEAR	STANDARD DEVIATION (MAN-REM PER REACTOR YEAR)
Job Functions:			
Routine Operations	49	60.1	43.8
Routine Maintenance	47	110.8	103.4
Inservice Inspection	18	27.5	29.1
Special Maintenance	44	273.1	285.7
Waste Processing	35	37.0	38.0
Refueling	42	19.2	30.1
Health Physics	40	29.3	17.0
Total Reported[1] Annual Exposure:	61	511.4	454.8

(1) Total exposure by job function differs from the annual reported total exposure due to conservatisms employed in compilation of job function exposures.

Table 12.4-6

ESTIMATED EXPOSURE FOR OPERATORS
IN CONTROL ROOMS

	OPERATORS[1] PER SHIFT	SHIFTS MANNED[1] PER DAY	ANNUAL OPERATOR MAN-HOURS[2]	ESTIMATED DOSE RATE (mRem/hr)	ANNUAL OPERATOR MAN-REM[1]
Control Room	5	3	43,800	0.25	11.0
Radwaste Control Room	1	3	8,760	0.25	2.2
TOTAL	6		52,560		13.2

(1) Assumed to be independent of the number of units in operation
(2) Based on each operator spending 8 hours in the control room and the control rooms being manned 365 days per year

Table 12.4-7

SUMMARY OF ROUTINE OPERATIONS EXPOSURE ESTIMATE [(3)]

	ANNUAL ESTIMATED MAN-REM SINGLE-UNIT	TWO UNITS
Routine Surveillances:		
Turbine enclosure[(1)]	10.4	20.7
Reactor enclosure	2.5	5.0
Radwaste enclosure[(2)]	1.6	1.6
	14.5	27.3
Periodic Tests:		
Turbine enclosure[(1)]	3.9	7.8
Reactor enclosure	9.6	19.2
Radwaste enclosure[(2)]	1.4	1.4
	14.9	28.4
Control Room Operations:		
Control room	11.0	11.0
Radwaste control room	2.2	2.2
	13.2	13.2
TOTAL	42.6	68.9

(1) Includes control structure exposures
(2) Includes offgas enclosure exposures
(3) Estimated and measured dose rates will increase approximately in proportion to power.

Table 12.4-8

ESTIMATE OF EXPECTED ROUTINE MAINTENANCE REQUIREMENTS

EQUIPMENT TYPE OR ACTIVITY	ESTIMATED MAN-HOURS PER YEAR
1. Valves	
a. Under 3 inches	3
b. 3 to 6 inches	9
c. 8 to 10 inches	12
d. 12 to 16 inches	17
e. Over 16 inches	23
2. Valve Operators	
a. Reach rod	1
b. Air operator	2
c. Motor operator	2
3. Pumps	
a. Reactor recirculation	150
b. RHR or condensate	100
c. Any other pump	50
4. Motors	
a. 200 hp or greater	20
b. Less than 200 hp	13
5. Motor-generator	35
6. Main generator	1000
7. Turbines	
a. Main turbine	2400
b. Any other turbine	240
8. Heat exchanger	30
9. Deleted	50
10. Chiller	5
11. Unit heater	5
12. Unit cooler	5
13. Compressor	24

Table 12.4-8 (Cont'd)

EQUIPMENT TYPE OR ACTIVITY	ESTIMATED MAN-HOURS PER YEAR
14. Fan or blower25	
15. Main condenser (per shell)	150
16. Hoist or crane	60
17. Instrument panel	150
18. Instrument rack	150
19. Motor control center	5
20. Switchgear	120
21. Centrifuge	37
22. Agitator	5
23. Transfer cart or conveyor	15
24. TIP drive	20
25. CRD hydraulic units (total per reactor)	600
26. HVAC filter unit (prefilter, HEPA filters, and charcoal filter)	20
27. Offgas system (holdup pipe to stack - total for station)	780

Table 12.4-9

SUMMARY OF IN-PLANT DIRECT RADIATION EXPOSURE ESTIMATES [1]

CATEGORY	ANNUAL ESTIMATED MAN-REM SINGLE-UNIT	TWO UNITS
Routine Operations	42.6	68.9
Routine Maintenance	232.5	427.7
Inservice Inspection	27.5	55.0
Special Maintenance	273.1	546.2
Radwaste Processing	37.0	74.0
Refueling	19.2	38.4
Health Physics	29.3	58.6
Total	661.2	1,268.8

(1) Estimated and measured dose rates will increase approximately in proportion to power.

Table 12.4-10

ESTIMATED TURBINE ENCLOSURE AND CONTROL STRUCTURE INHALATION EXPOSURES DUE TO AIRBORNE RADIOACTIVITY(1)

ROOM(4)	ESTIMATED ANNUAL MAN-HOURS SURVEILLANCE	TESTING	MAINTENANCE	TOTAL	ESTIMATED THYROID DOSE(2) (MAN-REM/YR)	ESTIMATED TRITIUM DOSE(3) (MAN-REM/YR)
Condenser Areas						
253	0	15	745	760	2.3	$2.1x10^{-2}$
254	0	0	450	450	1.4	$1.3x10^{-2}$
255	30	35	532	597	1.8	$1.5x10^{-2}$
256	91	92	1481	1664	5.0	$4.7x10^{-2}$
332	20	158	804	982	2.9	$2.7x10^{-2}$
342	0	3	47	50	$1.5x10^{-1}$	$1.4x10^{-3}$
499	7	56	1696	1759	5.3	$4.9x10^{-2}$
SJAE Areas						
154	0	24	203	227	$5.7x10^{-2}$	$5.2x10^{-4}$
333,334	46	0	1286	133	$3.3x10^{-1}$	$3.1x10^{-3}$
Mechanical Vacuum Pump Areas						
337	13	15	422	450	3.0	$2.9x10^{-2}$
Turbine Hall Areas						
544	9	0	3568	3577	$6.4x10^{-1}$	$6.1x10^{-3}$

Table 12.4-10 (Cont'd)

ROOM[4]	ESTIMATED ANNUAL MAN-HOURS SURVEILLANCE	TESTING	MAINTENANCE	TOTAL	ESTIMATED THYROID DOSE[2] (MAN-REM/YR)	ESTIMATED TRITIUM DOSE[3] (MAN-REM/YR)
Other Equipment Areas						
161,162	30	0	282	312	$5.9x10^{-2}$	$5.6x10^{-4}$
165	0	23	288	311	$5.9x10^{-2}$	$5.6x10^{-4}$
249	0	3	38	41	$7.8x10^{-3}$	$7.4x10^{-5}$
264	30	96	1909	2035	$3.9x10^{-1}$	$3.7x10^{-3}$
299	0	11	0	11	$2.1x10^{-3}$	$2.0x10^{-5}$
340	20	21	927	968	$1.8x10^{-1}$	$1.7x10^{-3}$
407	0	0	30	30	$5.7x10^{-3}$	$5.4x10^{-5}$
438	4	78	67	149	$2.8x10^{-2}$	$2.7x10^{-4}$
439,440,441	13	36	1041	1090	$2.1x10^{-1}$	$2.0x10^{-3}$
518	0	0	88	88	$1.7x10^{-2}$	$1.6x10^{-4}$
545,546,547	13	9	426	448	$8.5x10^{-2}$	$8.1x10^{-4}$
551	4	8	376	388	$7.4x10^{-2}$	$7.0x10^{-4}$
621	13	0	76	89	$1.7x10^{-2}$	$1.6x10^{-4}$
624	0	46	106	152	$2.9x10^{-2}$	$2.7x10^{-4}$
TOTALS	343	729	16,888	17,960	$2.4x10^{+1}$	$2.2x10^{-1}$

(1) All values in the table are on a per unit basis.
(2) Thyroid dose attributable only to inhalation of radioiodines.
(3) Uniform dose to the total body from uptake of tritium.
(4) Room numbers are shown on 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.

Table 12.4-11

ESTIMATED REACTOR ENCLOSURE INHALATION EXPOSURES DUE TO AIRBORNE RADIOACTIVITY[1]

ROOM[6]	ESTIMATED ANNUAL MAN-HOURS SURVEILLANCE	TESTING	MAINTENANCE	TOTAL	ESTIMATED THYROID DOSE[4] (MAN-REM/YR)	ESTIMATED TRITIUM DOSE[5] (MAN-REM/YR)
RWCU Pump Areas						
503,504,505	13	29	537	579	9.8×10^{-1}	-
507,508,509	20	33	360	413	7.0×10^{-1}	-
510	0	0	135	135	2.3×10^{-1}	-
522	0	24	125	149	2.5×10^{-1}	-
RWCU Filter/ Demineralizer Areas						
600	13	6	519	538	1.7	-
Refueling Areas						
700	46	119	4,233[2]	4,398	6.6×10^{-1}	3.8×10^{-2}
ECCS Areas						
102	46	66	752	864	5.8×10^{-1}	-
103	46	66	752	864	5.8×10^{-1}	-
108	46	72	650	768	5.1×10^{-1}	-
109	46	83	917	1,046	7.0×10^{-1}	-
203	0	11	422	433	2.9×10^{-1}	-
204	0	11	238	249	1.7×10^{-1}	-
297	0	0	84	84	5.6×10^{-2}	-
298	0	0	26	26	1.7×10^{-2}	-
309	0	5	683	688	4.6×10^{-1}	-
400	0	0	191	191	5.5	-
400A	0	0	442	442	$1.3 \times 10^{+}$	-
400B	0	0	532	532	$1.5 \times 10^{+1}$	-
400C	0	15	184	199	5.8	-
400D	0	0	303	303	8.8	-
400E	0	0	11	11	3.2×10^{-1}	-

Table 12.4-11 (Cont'd)

ROOM[6]	ESTIMATED ANNUAL MAN-HOURS				ESTIMATED THYROID DOSE[4]	ESTIMATED TRITIUM DOSE[5]
	SURVEILLANCE	TESTING	MAINTENANCE	TOTAL	(MAN-REM/YR)	(MAN-REM/YR)
Other Equipment Areas						
115	46	23	288	357	$1.9x10^{-1}$	-
209	0	0	98	98	$5.1x10^{-2}$	-
402A	137	0	600	737	$3.8x10^{-1}$	-
403	0	222	0	222	$1.2x10^{-1}$	-
406	0	0	126	126	$6.6x10^{-2}$	-
407A	0	0	114	114	$5.9x10^{-2}$	-
501	0	39	44	83	$4.3x10^{-2}$	-
511	30	66	570	666	$3.5x10^{-1}$	-
518A	0	0	74	74	$3.8x10^{-2}$	-
523	0	0	95	95	$4.9x10^{-2}$	-
599	0	0	156	156	$8.1x10^{-2}$	-
616,617	13	0	40	53	$2.8x10^{-2}$	-
618	13	3	40	56	$2.9x10^{-2}$	
TOTALS	515	893	14,341	15,749	$5.8x10^{+1}$[3]	$3.8x10^{-2}$

(1) All values in the table are on a per unit basis.

(2) Maintenance man-hours include 2,165 man-hours for refueling operations.

(3) The total thyroid dose of 58 man-rem takes no credit for respiratory protection. If a protection factor as low as 5 is assumed in the drywell (which is estimated to produce over 80% of the reactor enclosure thyroid exposures), the total thyroid dose is reduced to below 20 man-rem per year.

(4) Thyroid dose attributable only to inhalation of radioiodines.

(5) Uniform dose to the total body from uptake of tritium.

(6) Room numbers are shown on 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.

Table 12.4-12

ESTIMATED RADWASTE AND OFFGAS ENCLOSURES INHALATION EXPOSURES DUE TO AIRBORNE RADIOACTIVITY[(1)]

ROOM[(3)]	ESTIMATED ANNUAL MAN-HOURS				ESTIMATED THYROID DOSE	ESTIMATED LUNG DOSE
	SURVEILLANCE	TESTING	MAINTENANCE	TOTAL	(MAN-REM/YR)	(MAN-REM/YR)
Solid Radwaste Handling Areas						
124	6	6	212	224	$3.6x10^{-2}$	$1.7x10^{-2}$
126,130	12	10	449	471	$7.5x10^{-2}$	$3.5x10^{-2}$
127,129	12	12	449	473	$7.6x10^{-2}$	$3.5x10^{-2}$
138	6	3	146	155	$2.5x10^{-2}$	$1.2x10^{-2}$
139	6	3	146	155	$2.5x10^{-2}$	$1.2x10^{-2}$
142	6	17	109	132	$2.1x10^{-2}$	$9.9x10^{-3}$
144	6	11	233	250	$4.0x10^{-2}$	$1.9x10^{-2}$
233	0	3	3	6	$9.6x10^{-4}$	$4.5x10^{-4}$
234	6	2	122	130	$2.1x10^{-2}$	$9.8x10^{-3}$
422	0	0	60	60	$9.6x10^{-3}$	$4.5x10^{-3}$
423	0	0	40	40	$6.4x10^{-3}$	$3.0x10^{-3}$
470,471	61	0	34	95	$1.5x10^{-2}$	$7.1x10^{-3}$
515,516	30	0	100	130	$2.1x10^{-2}$	$9.8x10^{-3}$
[(2)]	0	0	4000	4000	$6.4x10^{-1}$	$3.0x10^{-1}$
Liquid Radwaste Handling Areas						
132	6	5	165	176	$1.6x10^{-1}$	$7.6x10^{-2}$
133	6	5	571	582	$5.3x10^{-1}$	$2.5x10^{-1}$
137	6	2	168	176	$1.6x10^{-1}$	$7.6x10^{-2}$
147	6	3	287	296	$2.7x10^{-1}$	$1.3x10^{-1}$
148	6	6	0	12	$1.1x10^{-2}$	$5.2x10^{-3}$
231	6	6	203	215	$2.0x10^{-1}$	$9.2x10^{-2}$
232	6	5	117	128	$1.2x10^{-1}$	$5.5x10^{-2}$

Table 12.4-12 (Cont'd)

ROOM[3]	ESTIMATED ANNUAL MAN-HOURS				ESTIMATED THYROID DOSE	ESTIMATED LUNG DOSE
	SURVEILLANCE	TESTING	MAINTENANCE	TOTAL	(MAN-REM/YR)	(MAN-REM/YR)
Liquid Radwaste Handling Areas (continued)						
237	6	26	253	285	2.6×10^{-1}	1.2×10^{-1}
238	6	17	300	323	2.9×10^{-1}	1.4×10^{-1}
242,243	12	16	100	128	1.2×10^{-1}	5.5×10^{-2}
245	0	0	78	78	7.1×10^{-2}	3.4×10^{-2}
247	6	5	142	153	1.4×10^{-1}	6.6×10^{-2}
248	6	5	142	153	1.4×10^{-1}	6.6×10^{-2}
Other Equipment Areas						
146	6	21	288	315	1.2×10^{-1}	5.7×10^{-2}
228	6	59	404	469	1.8×10^{-1}	8.4×10^{-2}
236	6	72	549	627	2.4×10^{-1}	1.1×10^{-1}
485,486	0	0	40	40	1.5×10^{-2}	7.2×10^{-3}
OFFGAS	0	0	780	780	3.0×10^{-1}	1.4×10^{-1}
TOTALS	247	320	10,690	11,257	4.3	2.0

(1) All values are on a per unit basis.
(2) Solid radwaste processing, container handling, and radwaste shipping.
(3) Room numbers are shown on 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.

Table 12.4-13

SKYSHINE SOURCE TERMS FOR ONE-UNIT OPERATING[(1)]

SOURCE DESCRIPTION	γ (6.2 MeV)/sec
HP Turbine	4.209×10^{9}
LP Turbine A	3.525×10^{9}
LP Turbine B	3.405×10^{9}
LP Turbine C	4.038×10^{9}
28"ϕ HP Turbine Inlet Pipe (Two)	3.010×10^{10} (Each)
42"ϕ HP Turbine Exhaust Pipe (Two)	2.836×10^{10} (Each)
42"ϕ Cross-Around Pipe (Six)	1.866×10^{10} (Each)
Steam Seal Evaporator	1.239×10^{10}

(1) These are all the exposed sources above the turbine operating deck, el 269'-0".

Table 12.4-14

DIRECT RADIATION DOSE RATES(1)

YARD AREA(2)

RECEPTOR POINT	DOSE RATE(3) (mRem/hr)
1. Field Office	$2.30x10^{-2}$
2. Lay-down Area	$1.47x10^{-2}$
3. Circulating Water Pump Structure	$5.67x10^{-2}$
4. Sewage Treatment Plant	$2.52x10^{-2}$
5. Access Road	$5.97x10^{-2}$

SITE BOUNDARY AND VISITORS' CENTER(4)

RECEPTOR POINT	DISTANCE FROM ORIGIN (ft)	DOSE RATE (mRem/yr)
6. Visitor's Center	1724	$1.18x10^{-2}$(5)
7. N	2835	4.3
8. NNE	2550	5.4
9. NE	2560	5.5
10. ENE	2579	3.9
11. E	2581	3.6
12. ESE	2579	1.9
13. SE	2570	1.5
14. SSE	3247	0.6
15. S	2504	3.7
16. SSW	2549	2.9
17. SW	2942	0.7
18. WSW	2787	1.3
19. W	2825	1.8
20. WNW	2636	2.8
21. NW	2571	3.4
22. NNW	2818	4.3

(1) Locations of receptor points and origin are shown on Figure 12.4-1. For receptor points 7 through 22, the distances given are the closest location to the origin within each of the 16 sectors.
(2) For Unit 1 only, operation at a capacity of 100%
(3) Dose rates include 0.01 mRem/hr for direct radiation from the radwaste enclosure.
(4) For both units operating at full power at a capacity of 80%
(5) Based on continuous occupancy of 8 hours per year at 100% plant capacity.

Table 12.4-15

ESTIMATED EXPOSURE TO UNIT 2 CONSTRUCTION WORKERS

PERSONNEL	ASSUMED LOCATION	ESTIMATED OCCUPANCY[1] (MAN-HOURS)	AVERAGE DOSE RATE (mRem/hr)[4]	ESTIMATED EXPOSURE[2] (MAN-REM)
Manual	Yard Area[3]	89,000	0.0335[5]	2.39
Manual	Unit 2 Reactor Enclosure Below Refueling Floor and Containment	91,000	0.01	0.73
Manual	On and Above Refueling Floor	25,000	0.311	6.22
Manual	Unit 2 Turbine Enclosure Below Turbine Deck	150,000	0.01	1.20
Manual	On and Above Turbine Deck	62,000	0.546	27.08
Nonmanual	Field Office	62,000	0.0215	1.07
Nonmanual	Unit 2 Reactor Enclosure Below Refueling Floor and Containment	19,000	0.01	0.15
Nonmanual	On and Above Refueling Floor	5,300	0.311	1.32
Nonmanual	Unit 2 Turbine Enclosure Below Turbine Deck	32,000	0.01	0.26
Nonmanual	On and Above Turbine Deck	13,000	0.0711	0.74
	TOTAL	548,300		41.16

(1) For remainder of Unit 2 construction, after Unit 1 operation commences.
(2) Based on an assumed availability of 80% for Unit 1.
(3) Average of points 1 through 5 shown on Figure 12.4-1.
(4) Based on operation at 100% power.
(5) From Table 12.4-14.

Table 12.4-16

AIRBORNE RADIOACTIVITY DOSES TO CONSTRUCTION WORKERS IN YARD AREA

RECEPTOR POINT[1]	DOSE RATES (mRem/yr) TOTAL BODY (GAMMA)	SKIN (BETA)	THYROID (GAMMA)
1. Field Office	2.47	4.77	0.57
2. Lay-down Area	1.14	2.21	0.27
3. Circulating Water Pump Structure	1.14	2.21	0.27
4. Sewage Treatment Plant	1.14	2.21	0.27
5. Access Road	2.47	4.77	0.57

[1] Locations of receptor points are shown on Figure 12.4-1.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

The organizational structure, including the responsibilities of those charged with performance of health physics duties, is described in Section 13.1. Health physics operations are made effective and ALARA goals are achieved through the cooperation and active participation of the various group represented in this structure.

The objectives of health physics operations are to:

a. Detect, identify and define radiation hazards.

b. Provide protection for personnel against radiation hazards.

c. Monitor and measure radioactive effluents from the plant.

d. Control plant-related radiation exposures (occupational and general public) to levels ALARA.

e. Control plant radioactive effluents to levels ALARA.

f. Conduct plant activities in conformance with authorized procedures and applicable regulations.

These objectives are accomplished or enhanced by the various training programs, by the use of prepared procedures, by performance of area surveys and monitoring of personnel, by record keeping, by applying good health physics practices, by periodic review and revision of procedures, by evaluation of activities for ALARA purposes, and by the control of plant operations to minimize occupational exposures and releases to the environment. Health physics operations include the posting, notification, and reporting provisions of 10CFR19 and General Employee Training (Section 13.2) provides the requisite instruction to workers.

Health physics operations conform with the guidelines of Regulatory Guide 8.2 (Rev 0) and of ANSI N13.2 (1969), Guide for Administrative Practices in Radiation Monitoring, with the clarification that controlled areas which are locked or otherwise prevent personnel access are not required to be surveyed at a specific periodicity. In addition to this section, Sections 13.1, 13.2, 13.5, and 12.1 describe the management commitment, organization, responsibilities, authority, training, procedures, and review techniques which implement Regulatory Guide 8.8 (Rev 3), as it applies to the operating phase and Regulatory Guide 1.8 (Rev 1-R). As described in Section 12.1, a formal ALARA review program, which is consistent with the guidelines given in Regulatory Guide 8.8, was implemented during the design and construction phase. The qualification requirements for the Radiation Protection Manager are described in Section 13.1.3. Chapter 17 addresses implementation of other applicable regulatory guides. Implementation of Regulatory Guide 1.16 is addressed in the Technical Specifications.

12.5.2 FACILITIES, EQUIPMENT, AND INSTRUMENTS

This section describes the basic facilities, equipment, and instruments directly related to the health physics operations.

12.5.2.1 Health Physics and Chemistry Facilities

12.5.2.1.1 Supervisory Offices and Field Offices

Health Physics and Chemistry supervisory offices are located outside the RCA. Field offices and work areas for technicians are located adjacent to the RCA.

12.5.2.1.2 Chemical Laboratories

Chemical laboratories are located in the Chemistry Laboratory Expansion including the Hot Lab, the Instrumentation Lab (within the bounds of the RCA) and the Cold Lab (beyond the bounds of the RCA). A conventional chemistry section and radiochemistry section are provided. These sections are segregated so that noncontaminated samples (such as river water) can be prepared and analyzed without having to work under radiologically controlled conditions. The laboratories are accessible to the counting room for convenience in transporting prepared samples for counting. Sample stations are located near their respective systems for sampling condensate, feedwater, reactor water, and radwaste. The laboratories are designed to support routine chemical analysis, as well as, developmental work for emergent issues and appropriate emergency response functions as described in the Emergency Plan.

12.5.2.1.3 Counting Room

The counting room is located in the Chemistry Laboratory Expansion in a low background area. The room is designed with thick shield walls to further reduce background radiation, thus improving instrumentation sensitivity.

12.5.2.1.4 Control Points

Control points and change areas are located throughout the plant near the various job sites. These facilities are for the most part temporary. They are set up during outages and special jobs. They are staffed and stocked to provide health physics monitoring and protective equipment on an as-needed basis.

12.5.2.1.5 Testing and Laboratory Facilities

Calibration and instrument repair facilities are provided. Licensed sources are stored in this facility under approval of the Manager Radiation Protection. Sources are used to calibrate portable radiation survey meters, direct reading dosimeters, certain process and effluent radiation instruments, and area monitors.

12.5.2.1.6 Medical Support

Arrangements with local physicians and hospitals provide offsite support for continued medical treatment of traumas involving radioactive contamination.

A contract is maintained with a primary health physics-medical consultant. The consultant retains a qualified staff that provides expertise in the long-term treatment of patients exposed to excessive external radiation or internal contamination.

12.5.2.1.7 Personnel Decontamination Facility

A personnel decontamination facility located in the radwaste enclosure at el 217' is provided with a sink, shower, change area, and contamination monitoring devices. Personnel decontamination is performed in accordance with Health Physics procedures. Control of personnel decontamination is provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program.

No male/female discrimination has been made in the design for the decontamination facilities. Adequate showers and sinks are available for personnel decontamination. Procedures will be implemented to provide the appropriate privacy. Additional male and female facilities are provided in the administration building but are not intended for contamination removal.

12.5.2.1.8 Laundry and Respirator Facility

A laundry and respirator facility is located adjacent to the Unit 1 turbine enclosure at el 217'. The facility includes space for sorting anticontamination clothing, and for receiving clothing from an offsite laundry. The facility includes space for inspecting, surveying, bagging, and repairing respirators. Procedures are established for inspection, cleaning and maintenance of such protective equipment.

12.5.2.2 Instruments and Equipment

Instrumentation design and capabilities improve as the state of the art changes. Normally, the instrumentation listed in Table 12.5-1 or instruments of similar capabilities will be provided. Any instrumentation or equipment described in these sections may be replaced by items providing similar or improved capabilities.

Instrumentation for detecting and measuring radiation consists of counting room equipment, portable instrumentation, and air samplers. Capabilities for detecting alpha, beta, gamma, and neutron radiation are provided. Sufficient inventory is provided to accommodate use, repairs and calibration. The equipment is described later in this section and is listed in Table 12.5-1. Sufficient chemical equipment and analytical instruments are provided to perform the required sample preparations and analyses.

LGS will equip the counting room and provide the portable instruments, personnel monitoring instruments, and protective equipment with the capabilities as described in Regulatory Guide 8.8, section 4. Support facilities at LGS will also follow the guidance of Regulatory Guide 8.8.

LGS is in compliance with Regulatory Guide 1.97 (Rev 2) in regard to Health Physics and Chemistry Laboratory and survey equipment.

Control of instrument storage, calibration and maintenance is provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program.

Operability of personnel monitors, radiation survey instruments, and laboratory equipment is verified periodically in accordance with plant procedures and work processes.

12.5.2.2.1 Chemical Laboratories

The chemical laboratories are segregated so that one section handles only low level or background samples and the remaining section handles contaminated samples.

Grab samples are transported to the appropriate section of the laboratories. The laboratories, are equipped with constant air flow fume hoods. The fume hoods permit safe preparation and processing of samples under controlled conditions. Effluent of fume hoods within the RCA are filtered through HEPA filters. The laboratories are stocked with chemical reagents and equipment to provide for sample preparation and analysis.

The laboratories contain conventional analytical instruments, equipment and chemicals. Samples intended for activity analysis and isotopic identification are prepared for transport to the counting room. A frisking station is located in the Chemistry Laboratory Expansion.

12.5.2.2.2 Counting Room

Samples processed in the chemical laboratories for activity analysis and isotopic identification, and samples direct from the plant (such as air samples and smears) are transported to the counting room located in the Chemistry Lab Expansion. Equipment is available for gross alpha, gross beta, and gross gamma activity measurements and for determination of the activity levels of specific isotopes. Proportional counters and gamma spectrometers are provided. The room itself has thick shield walls to reduce instrument background radiation. The room is temperature controlled and the voltage supply is regulated for instrument stability.

Major instrumentation includes a Gamma Spectrometer for specific isotopic identification; alpha and beta counters; a scintillation detector; and a liquid scintillation analyzer for low energy emitters such as tritium.

Background and efficiency checks are performed routinely. Alpha and beta plateaus are established to determine operating voltages for proportional counters. Calibrations are based on NIST related standards for the isotopes of interest. Conventional reference standards such as Sr, C, and Ra in equilibrium with daughter activity, Cs, and Co are used for calibration when gross counting is the objective.

12.5.2.2.3 Portable Survey Instruments and Equipment

Portable survey instruments and equipment are used primarily for conducting area surveys and for monitoring personnel throughout the plant. Some portable instruments and equipment are designated for emergency use and will be stored at the Technical Support Center, readily accessible to personnel responding to an emergency.

Portable instruments and equipment for routine plant use are provided to permit alpha, beta, gamma, and neutron radiation measurements, and for obtaining samples of surface and airborne contamination. The inventory includes ion chambers, GM probes, neutron rem counters or equivalent, alpha probes, alpha and beta counters, scintillation detectors, high and low volume air samplers, filters, charcoal cartridges, and smears. Portable instruments and equipment available for emergency use include air sampling equipment with particulate filters and silver zeolite cartridges, portable ion chambers, alpha scintillation probes, energy compensated beta/gamma GM probes (for low energy photons), and portable beta/gamma geiger counters.

At least some of the ion chambers and GM probes have movable beta shields to enable distinguishing between beta and gamma radiation. The ion chambers are the prime devices for dose rate determinations and have beta factors specified as appropriate for the type of meter and its use.

GM probes may be used for dose rate determinations. Their application for this purpose is of value when the ability of an ion chamber to respond reliably is impaired by high humidity, high or low temperature, or very low dose rates. Certain instrument designs with GM probes have extendable arms. This feature allows the user to remain in a relatively lower radiation area while measuring high dose rates at the point of interest. This application is a good example of an effective ALARA policy.

The ranges and numbers of instruments are adequate for their intended use whether routine or emergency.

12.5.2.2.4 Personnel Dosimeters

Personnel monitoring is provided by the use of such devices as optically stimulated luminescent, direct reading dosimeters (electronic dosimeters or pocket ion chambers), or calculations from area survey data and exposure times. Personnel monitoring is provided per 10CFR20. The form of personnel monitoring depends on the type of radiation and the expected radiation level. Types of dosimetry devices change as the state of the art improves.

Devices are available and used appropriately for determining whole body or equivalent exposure, extremities exposure and skin exposure. Health Physics practices include the use of multiple badges in addition to the whole body badge as required by procedures.

A direct reading dosimeter will be used by all personnel entering the RCA who are required to be monitored for exposure per 10CFR20. The DRD is used to provide real time indications of dose received by an individual and provides estimated exposure data. As required, these dosimeters are calibrated and/or drift checked on a routine basis per procedures.

Dosimetry devices used for determining the official exposure doses are subject to extensive quality control programs. This is true whether the processing is by a contractor or onsite. Currently, DLRs are used for penetrating (gamma, neutron) and nonpenetrating (beta) radiation. Whenever neutron exposure is of concern, the current technique is to use neutron survey instruments to determine a rem dose rate, then to multiply by the stay time or a monitoring device/DLR which has been accredited for neutron monitoring through NAVLAP N13.11 criteria.

The personnel dosimetry program is conducted by qualified personnel under the direction of the Radiation Protection Manager.

Control of personnel monitoring (e.g., whole body counter) and external (e.g., DLR system) is provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program.

12.5.2.2.5 Miscellaneous Instruments and Equipment

Other monitoring devices are available and may be assigned to personnel or located in work areas for the purpose of alerting personnel to changes in radiation condition.

Radiation monitors that "chirp" at a rate proportional to gamma dose rate or contain an audible alarming feature are assigned to personnel as deemed necessary. These devices are of particular value when worn by individuals whose duties routinely involve entering various areas of the plant. Changing operating conditions or the progression of the individual into the area could cause sudden changes in radiation levels. These devices alarm at preset values to alert personnel of the changed conditions as the individual walks into the area.

Alarming rate meters may be positioned in a given work area. They serve the purpose of alerting individuals working in that area of increases in radiation level above a preset value.

Friskers, personnel contamination monitors, or portal monitors are positioned at or near various change areas and plant exit points. The purpose of these devices is to control the spread of radioactive contamination. Friskers are generally used within the plant for personnel to monitor themselves at any time, especially when leaving a controlled area. Personnel contamination monitors or friskers are generally used at all primary exits from the radiologically controlled area and at the access points to approved eating, drinking, and smoking areas within the radiological controlled area. The placement of these devices may vary in support of the station's contamination control program, and other devices with the required sensitivity may be considered for use if the situation warrants. Instrumentation, such as small article monitors or friskers, is used to monitor tools, equipment and personal items. The instruments are used to ensure that items exiting the RCA are within procedurally defined limits.

Continuous air monitors are used to monitor airborne concentration at specific work locations. These CAMs provide the means to observe trends or sudden changes in the airborne concentrations. They are not intended for quantitative analysis. The fixed filter-type can be used as a low volume grab air sample in that the filter medium can be removed and analyzed in more detail in the counting room.

Fixed area radiation monitors are mounted in selected locations. Each contains a gamma-sensitive detector, local indicator and local alarm, as well as indicators, alarms, and recorders in the control room. These monitors will alert personnel to unexpected or abnormally high radiation levels in these areas.

12.5.2.2.6 Bioassay

Internally deposited radioactive material is evaluated by use of whole body counting or in-vitro analysis. The whole body counter sensitivity for gamma emitting isotopes of interest will be equivalent to small fractions of isotopic organ burdens.

The bioassay program will incorporate features of Regulatory Guide 8.9 and Regulatory Guide 8.26. Conservative investigation levels are established. When investigation levels are exceeded, an investigation will be performed and, as necessary, will include consultation with independent experts for further evaluation of internal exposure consistent with Regulatory Guide 8.9 and International Commission on Radiological Protection criteria.

12.5.2.2.7 Personnel Protective Equipment

Special protective equipment such as coveralls, plastic suits, shoe covers, gloves, head covers, and respirators are available, and are stored in various plant locations and clothing change areas. This equipment is used to prevent deposition of radioactive material internally or on body surfaces. Most of the plant will be kept free of contamination so that no special protective equipment will be needed. Contaminated areas are identified with posted signs. Radiological postings, work orders or RWPs (or equivalent) are the primary mechanisms for defining the equipment required to enter these contaminated areas.

A variety of combinations of protective equipment may be prescribed depending on the nature and level of the contamination. The use of engineering controls will be considered to minimize the need for protective equipment and clothing as practical. Cotton clothes may be adequate normally; but in wet areas plastic rain suits or bubble suits may be prescribed. Respirators may be required if airborne hazards exist or if surface contamination could cause an airborne hazard as defined in the implementing procedures. Sufficient quantities of NIOSH approved respiratory equipment will be provided to adequately protect workers in general and emergency conditions. This will include SCBAs, respirators (general use or welding), and all the necessary supplementary equipment required (hoses, regulators, etc.).

The guidance of Regulatory Guide 8.15 will be followed to ensure the proper selection, care, fitting, and use of respiratory protective equipment. Regular sampling and survey programs will be provided to determine the need for the appropriate respiratory protective devices. Bioassays, record keeping, and medical evaluations will be incorporated into the LGS respiratory protection program. Written procedures will be developed with the help of the operating experience at PBAPS to provide an effective and acceptable respiratory protection program. Section 12.5.3.5.3 also addresses compliance with Regulatory Guide 8.15.

Control of respiratory protection, including testing, is provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program.

12.5.3 PROCEDURES AND PRACTICES

Health physics procedures are classified according to specific concerns. Thus, there are operating procedures, analytical procedures, emergency procedures, and surveillance test procedures. Health physics is a consideration in other plant procedures as well. This section describes the fundamental procedures applicable to health physics operations. The procedures and methods of operation which, in combination with licensee policy and training, ensure that radiation exposures will be ALARA are those which implement the controls and prescribe the use of equipment described in this section and in Section 12.5.1.

12.5.3.1 Radiation Surveys (Area Surveys)

Area survey procedures describe the purpose and techniques of detecting the presence of and measuring the level of radiation and contamination. Contamination may be on surfaces or airborne. Area surveys are conducted throughout the plant. Such surveys may be routine or may be related to specific jobs upon request. An area survey may be performed before, during, and after various work activities. Area surveys are performed by health physics technicians or other trained and qualified personnel (i.e. Advanced Radiation Workers). Control of radioactivity

sampling (air, surface, liquids) is provided by procedures that are responsive to the appropriate portions in the Quality Assurance Program.

12.5.3.1.1 Radiation Detection

The primary instrument for beta/gamma dose rate measurements is an ion chamber. Circumstances may require the use of other instruments to determine dose rates. GM probes may be used for low radiation levels or where environmental conditions (temperatures, humidity) cause erratic responses from ion chambers.

Surveys for neutrons are performed by instruments designed for that purpose. A rem counter or equivalent that has the ability to measure neutron dose rate in rem per hour is the preferred neutron measurement instrument. However, other instruments or devices for determining neutron energies may also be used. For example, moderated and unmoderated boron triflouride (BF_3) probes may be used to detect the presence of low and high energy neutrons. Some instruments are designed with extension arms. These types of instruments may be used in high dose rate areas to provide distance from the source to the technician, thus reducing personnel dose.

12.5.3.1.2 Surface Contamination Detection

A variety of techniques are necessary to detect and measure radioactive contamination. Procedures describe the use of smears (e.g., small paper discs) and swipes (e.g., large area survey) to wipe a surface to pick up removable contamination. Smears and swipes are analyzed using portable survey meters and/or counting room equipment. Fixed contamination is determined by scanning a surface with portable survey meters. Equipment is available for alpha and beta/gamma activity measurements.

12.5.3.1.3 Airborne Contamination

Airborne contamination is determined by using air samplers to draw a known volume of air through a filter paper or charcoal cartridge. A charcoal cartridge is used with the filter paper where iodine is of concern. The filter paper is analyzed by gross beta/gamma count and/or gamma spectrometry. Gamma spectroscopy is also performed on the charcoal cartridges. The gamma spectrometry identifies the particulate and iodine isotopic activity. DAC-hours shall be tracked in accordance with Heath Physics procedures. Gross beta/gamma count data is used to judge the need for filter paper gamma spectrometry. High volume air samplers and low volume air samplers having nominal sample rates of 25 scfm and 1 scfm, respectively, are available. The high volume air sampler is used primarily to obtain grab samples rapidly before, during, and after work activities. The low volume air sampler is used primarily to obtain the average air concentration for the work period.

12.5.3.1.4 Survey Frequency and Techniques

The frequency and extent (scope) of area surveys is a function of dose rate in the area, accessibility to the area, and nature of the area in plant operations. For example, contamination checks in control rooms and eating areas are performed nominally on a daily basis; seldom entered high radiation areas are surveyed as infrequently as monthly, or only as needed for entries. As part of the ALARA program, the performance of area surveys are coordinated so that, wherever feasible, routine area survey data is used to perform RWP work. This practice avoids duplication and reduces exposure to technicians.

Area survey data are usually obtained or known before work starts in an area. Resurveys may be performed in the area during the job if the work activity is prolonged. Other conditions for resurveys would be if the work activity or other plant operation caused changed conditions. Surveys may also be performed at the completion of the work activity.

Depending on the survey results, the area surveyed is roped off and posted appropriately to alert personnel to the radiation conditions and requirements for entry.

Procedures for area surveys describe the use of instruments, effective survey techniques, and documentation of data. The specifications for clean, radiation and high radiation areas are defined. Various levels of surface and airborne contamination are established as guidelines for prescribing protective equipment such as clothing and respirators. Procedures include consideration for potential as well as actual radiation hazards.

12.5.3.2 Radiation Work Permits (or Equivalent)

Where radiation dose rates, airborne concentrations, or surface contamination levels exceed procedural limits, an RWP (or equivalent) is required for work. Health Physics will evaluate the radiological conditions associated with the work to be performed. Health Physics will specify the appropriate protective clothing, equipment, and monitoring required including dosimetry by use of radiological postings, work orders, or RWPs. Area survey frequency is established by Health Physics. All personnel performing work under a particular RWP (or equivalent) must be familiar with the permit conditions and must sign a RWP (or equivalent) compliance sheet. Information documented from RWP (or equivalent) entries include name, time in and out, and estimated exposure for entry. The RWP (or equivalent) computer may be used in lieu of a compliance sheet to document this information. Health Physics may terminate an RWP (or equivalent) if radiation conditions change.

Health Physics supervision selectively reviews RWPs and survey documentation which are then filed. RWPs (or equivalent) serve as a source of data for dose comparison on repeat jobs. They can be used to determine the effectiveness of ALARA efforts.

12.5.3.3 Handling and Storage of Radioactive Material

Health Physics personnel are notified of the receipt of radioactive material, and of intended shipment of radioactive material. This is done so that required surveys can be performed and to verify that correct labeling and placarding has been accomplished.

Calibration sources for radiation instrumentation and sources used to prepare secondary standards are stored in a source vault. This vault is capable of being locked. The lock is under the control of the Radiation Protection Manager.

Small quantities of sealed or unsealed sources may be stored for convenience in shielded cabinets, caves, or safes. Such sources are used locally in the chemical laboratories, counting room, or when response-checking instruments throughout the plant.

Spare instruments containing built-in sources and slightly contaminated equipment intended for reuse may be stored at times in the warehouse. Such items are clearly identified as containing radioactive material.

12.5.3.4 Controlling Access and Stay Time

The plant is surrounded by security fencing. Entrance must be via the guardhouse. Security procedures are applied by the guards at this point to identify each individual and to determine their purpose for entry. Security and dosimetry badges are assigned. Escorts are provided to satisfy procedural requirements. Entrance to the RCA is via the Health Physics Access Point. Health Physics procedures are applied at this point to set exposure limits and ensure Health Physics Training of each individual prior to entry.

12.5.3.4.1 Access to Radiation and High Radiation Areas

Radiation areas are identified by posted radiation signs. Signs are used to define requirements for entry. Where appropriate, yellow and magenta or black rope or tape is used as a barrier to prevent access or to divert personnel to a specific control point for access. RWPs are used to define the exposure limits, document entry and exit, and record estimated exposure for each individual. Procedures describe the purpose and application of the RWPs. Administrative guides for personnel exposure are established by procedure. These guides are set at values less than the exposure dose limits in 10CFR20. Variations from these guides must be requested. Procedures describe the steps for approval of dose extensions. Additionally, positive control is exercised over each high radiation area by use of barricades, conspicuously posting the area, and requiring an RWP for entrance. Areas greater than 1000 mR/hr are controlled by using lock doors or gates. Exceptions to this practice are permitted per Technical Specification 6.12. Issuance of the keys are controlled by Health Physics. Each key used and the individual using the key are recorded. Certain work activities having high dose rates and short stay time may be monitored directly by Health Physics Technicians to prevent personnel from inadvertently exceeding the prescribed dose for the work. Personnel are advised to observe the reading on their direct reading dosimeter frequently.

12.5.3.4.2 Contamination Control

Contaminated areas are conspicuously identified to prevent inadvertent access. Floor coverings, called step off pads, typically are used to access these areas. Appropriate procedures give guidance for the selection and application of protective clothing and respirators under various specific conditions. Personnel requiring access to contaminated areas must determine the radiological controls and requirements via Health Physics briefings, radiological postings, work orders or RWPs.

As personnel leave the work area, they remove the protective clothing and respirators before stepping across the step-off-pad. Personnel must monitor themselves to assure that no contamination has been transferred to their bodies or clothing.

Tools or equipment to be removed from a contaminated area are surveyed or contained prior to removal from the area. Items suspected of or having loose surface contamination must be placed in a container (e.g., a plastic bag), surveyed, and tagged at the step-off-pad. The tools may be placed in storage, may be taken to another contaminated area for reuse, or they may be designated for appropriate storage, disposal, or decontamination. Tools, equipment and personal items are surveyed prior to removal from the RCA. Health Physics procedures describe the survey techniques and limits.

The presence of radioactive contamination, whether surface or airborne, inhibits mobility of personnel around the plant. Protective equipment that must be worn creates inconveniences and

introduces other factors that affect performance. For these reasons, plus the obvious external and internal radiation hazards, decontamination is initiated judiciously to confine the contamination to as small an area as practicable and/or to reduce the contamination levels to minimize protective requirements. Special coatings that aid in decontamination are applied to walls and floors. The ventilation flow pattern is from clean areas to contaminated areas. Process equipment is isolated in various cavities or cells. These cells are vented in a controlled manner, usually through filters to effluent stacks that monitor flow rate and radioactivity. Highly contaminated equipment drains are piped to sumps to avoid the use of floor drains and attendant spillage of fluids on the floor.

Control of radioactive contamination measurement and analysis and radioactive contamination control are provided by procedures that are responsive to the appropriate portions of the Quality Assurance Program.

12.5.3.5 Training

See Section 13.2.

12.5.3.5.1 Respiratory Fit Test

A functioning Respiratory Protection Program, which meets the requirements of 10CFR20. and Regulatory Guide 8.15 (October 1976), permits the application of protection factors to select the appropriate type of respirators. This can only be done for individuals who have received training in the use of the respirators, have been tested for the validity of fit for the type of respirators intended for their use, and who have been medically certified. Procedures are established that describe the technique, fit test frequency, and define acceptance criteria.

12.5.3.5.2 Compliance with Regulatory Guides

Regulatory Guide 8.2 - Section 12.1 describes the general ALARA program and policies. As specific procedures and the LGS ALARA plan are developed, Regulatory Guide 8.2 will be used for guidance.

Regulatory Guide 8.7 - Exposure records will be kept in accordance with 10CFR20 and maintained until the NRC authorizes their disposition. As the procedures are developed, the guidance of ANSI N13.6 (1969) as endorsed by Regulatory Guide 8.7 will be followed.

Regulatory Guide 8.8 - Sections 13.1, 13.2, 13.5, 12.1, and 12.5 address the management commitment, organization responsibilities, authority, training, procedures, and review techniques which implement Regulatory Guide 8.8 for LGS operation.

Regulatory Guide 8.9 - As stated in Section 12.5.2.2.6, the bioassay program will incorporate features of Regulatory Guide 8.9. When conservative investigative levels are exceeded, a consultant will assist in a more detailed evaluation of internal exposure. These procedures and practices will follow the guidance of International Committee of Radiation Protection publications and Regulatory Guide 8.9.

Regulatory Guide 8.10 - The development and implementation of a formal ALARA program will follow the guidance of Regulatory Guide 8.8 which is a nuclear power plant specific reference of Regulatory Guide 8.10.

Regulatory Guide 8.13 - Instruction to workers concerning prenatal radiation exposure will be given as part of the General Employee Training Program. This program will provide all employees with the information that is identified in Regulatory Guide 8.13.

Regulatory Guide 8.15 - The development and implementation of a respiratory protective equipment program will follow the guidance of Regulatory Guide 8.15.

Regulatory Guide 8.26 - The bioassay program will incorporate features of Regulatory Guide 8.26.

Regulatory Guide 1.8 - LGS is in compliance with this guide to the extent discussed in Sections 13.1 and 13.2. The licensee has implemented ANSI/ANS 3.1 (1978), section 5. This standard is a revision to ANSI N18.1 (1971), which is endorsed by Regulatory Guide 1.8.

Regulatory Guide 1.16 - LGS Technical Specifications were based on NUREG-0123, Revision 4, "Standard Technical Specifications for General Electric BWRs," as described in Chapter 16. Reporting of operating information will be in accordance with the Technical Specifications,

Regulatory Guide 1.33 - LGS will follow the guidance of Regulatory Guide 1.33 which endorses/modifies ANSI N18.7 (1976) with certain alternate approaches which are addressed in detail in Section 17.2.

Regulatory Guide 1.39 - LGS is in general conformance to the guidance of Regulatory Guide 1.39, which endorses/modifies ANSI N45.2.3 (1973). LGS will comply with Regulatory Guide 1.39 with certain alternate approaches which are outlined in detail in Section 17.2.

Regulatory Guide 1.97 - LGS is in compliance with Regulatory Guide 1.97 in regard to Health Physics and Chemistry Laboratory and Survey Equipment.

Table 12.5-1

HEALTH PHYSICS INSTRUMENTATION

INSTRUMENT	SENSITIVITY/RANGE	USE	NORMALLY REQUIRED QUANTITIES NOW
Gamma Spectrometer	10 minute count (μCi): ⎡ Co-60, Cs-134, ⎤ ⎢ Cs-137, I-131, ⎥ - $1x10^{-4}$ ⎣ Zn-65 ⎦ Noble Gases - $1x10^{-3}$	Gamma scans	4
Alpha/Beta Counter	Beta Efficiency 10%-50% (E dependent)	Alpha/beta discrimination	2
Whole Body Counter System	<1/20 of the International Committee of Radiation Protection Publication 30 Annual Limit of Intake	Internal dosimetry	1

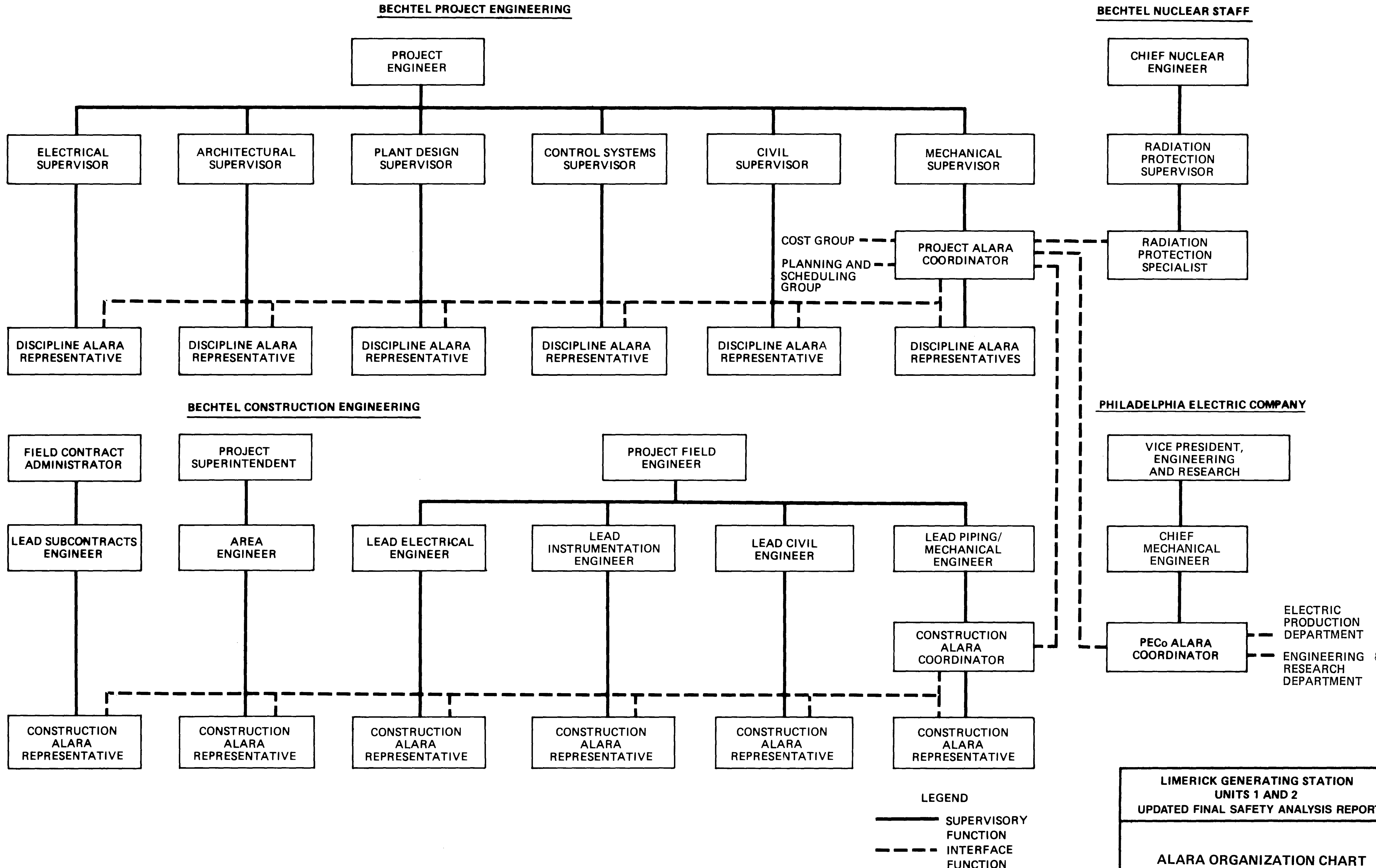
RADIOSURVEILLANCE INSTRUMENTS	SENSITIVITY/RANGE	USE	NORMALLY REQUIRED QUANTITIES NOW
RO-2	0-5000 mR/hr	Field survey	10
RO-2A	0-50 R/hr	Field survey	15
E-520	0.2-2000 mR/hr	Field survey	5

Table 12.5-1 (Cont'd)

RADIOSURVEILLANCE INSTRUMENTS	SENSITIVITY/RANGE	USE	NORMALLY REQUIRED QUANTITIES NOW
E-140N	0-50,000 cpm	Field survey	15
HP270 probe	0.1 mR/hr-10 R/hr	Field survey	15
HP210AL probe	-	Field survey	15
HP210T probe	-	Field survey	15
Teletector	0-1000 R/hr	Field survey	3
HP220A probe	-	Field Survey	1
HP260A probe	-	Field Survey	10
EC4-X (& cables)	0.01 mR/hr-10,000 R/hr	Portable area monitors	10
DA1-6 probe (for EC4-X)		Portable area monitors	10
PRM-6 with AC-3	0-500,000 cpm	Alpha surveys	0
PNR 4	0-5000 mR/hr	Neutron surveys	4
HP280 - 3 in. sphere		Neutron surveys	0
RO7 + BH midrange detector + BH high range detector, + RX5 - 5 ft extender, + 2 (15 ft) cables	0-200 R/hr; 0-20,000 R/hr	High range ion chamber	2
Underwater probe	0-5000 R/hr 5-5000 mR/hr	High range under water	2
Rm 20	0-500,000 cpm	Frisking stations	25
Alpha + Beta CAMs		Air monitoring	3
Low Volume air samplers	-	Air monitoring	25
SAC-4		Alpha Survey	3

Table 12.5-1 (Cont'd)

RADIOSURVEILLANCE INSTRUMENTS	SENSITIVITY/RANGE	USE	NORMAL REQUIRED QUANTITIES NOW
High Volume air samplers	-	Air monitoring	10
Portal Monitors	Approx. MDA 1 mR	Contamination control	5
Pocket Ion Chamber:			
Charging Units		Exposure control	as needed
0-200 mR	0-200 mR		
0-500 mR	0-500 mR		
0-1500 mR	0-1500 mR		
Electronic Dosimeter		Exposure Control	as needed
Portable MCA + HP Ge detector + 24 hr dewar			1
E-530N	0.02-20 R/hr	Field surveys	2
Shepherd 142-10 panoramic calibrator	0-1,600 mR/hr	Calibration	1
Shepherd 89 calibrator	0.01 mR/hr-1200 R/hr	Calibration	1
Condenser R meter	-	Calibration	1



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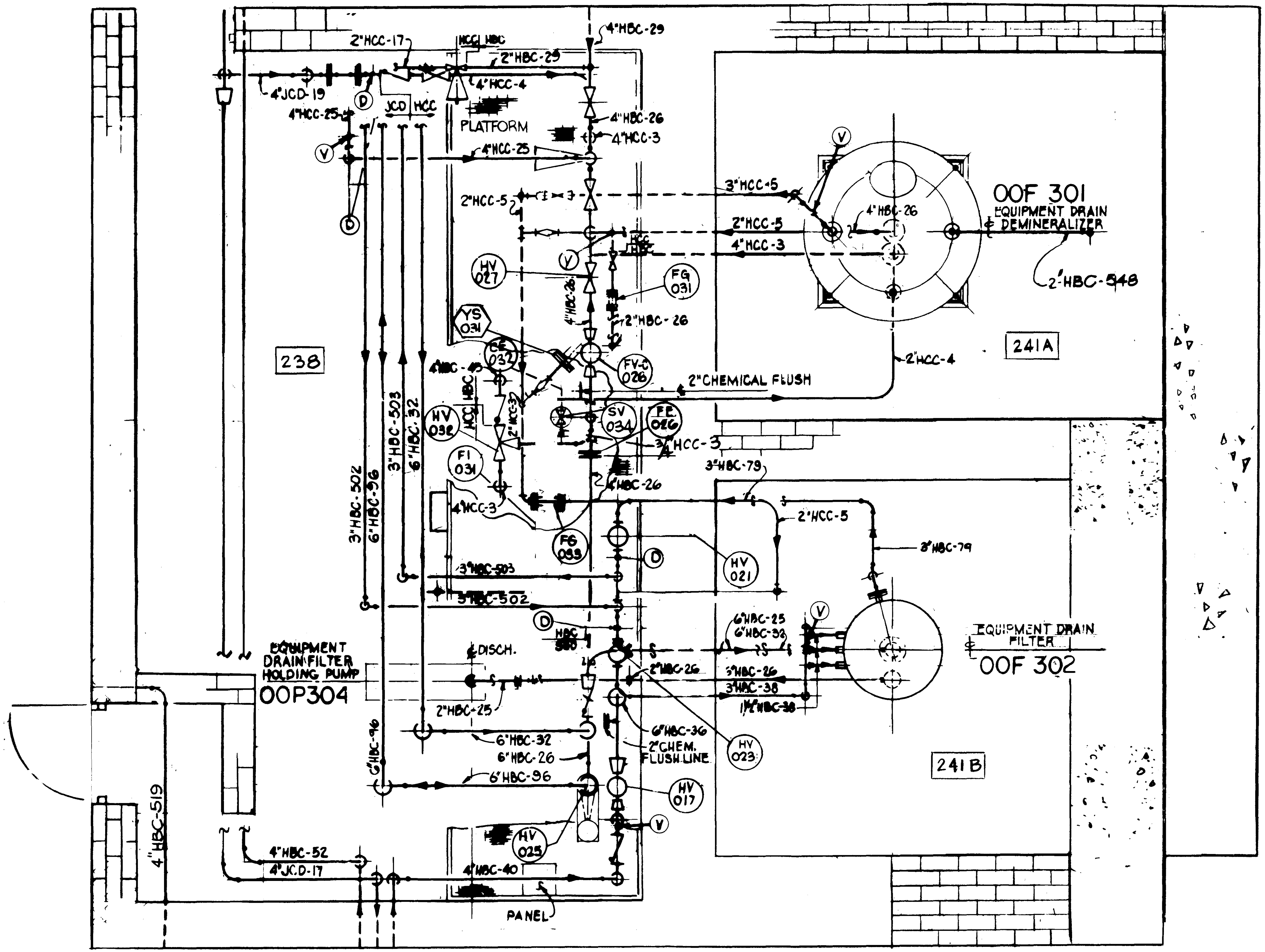
ALARA ORGANIZATION CHART

FIGURE 12.1-1



NOTES:

HV–REMOTE OPERATED CONTROL VALVE
FV–C-FLOW CONTROL VALVE
SV–SOLENOID OPERATED VALVE
LF–LINE FILTER
FE–FLOW ELEMENT
FG–FLOW GAUGE
FI–FLOW INDICATOR
YS–Y STRAINER

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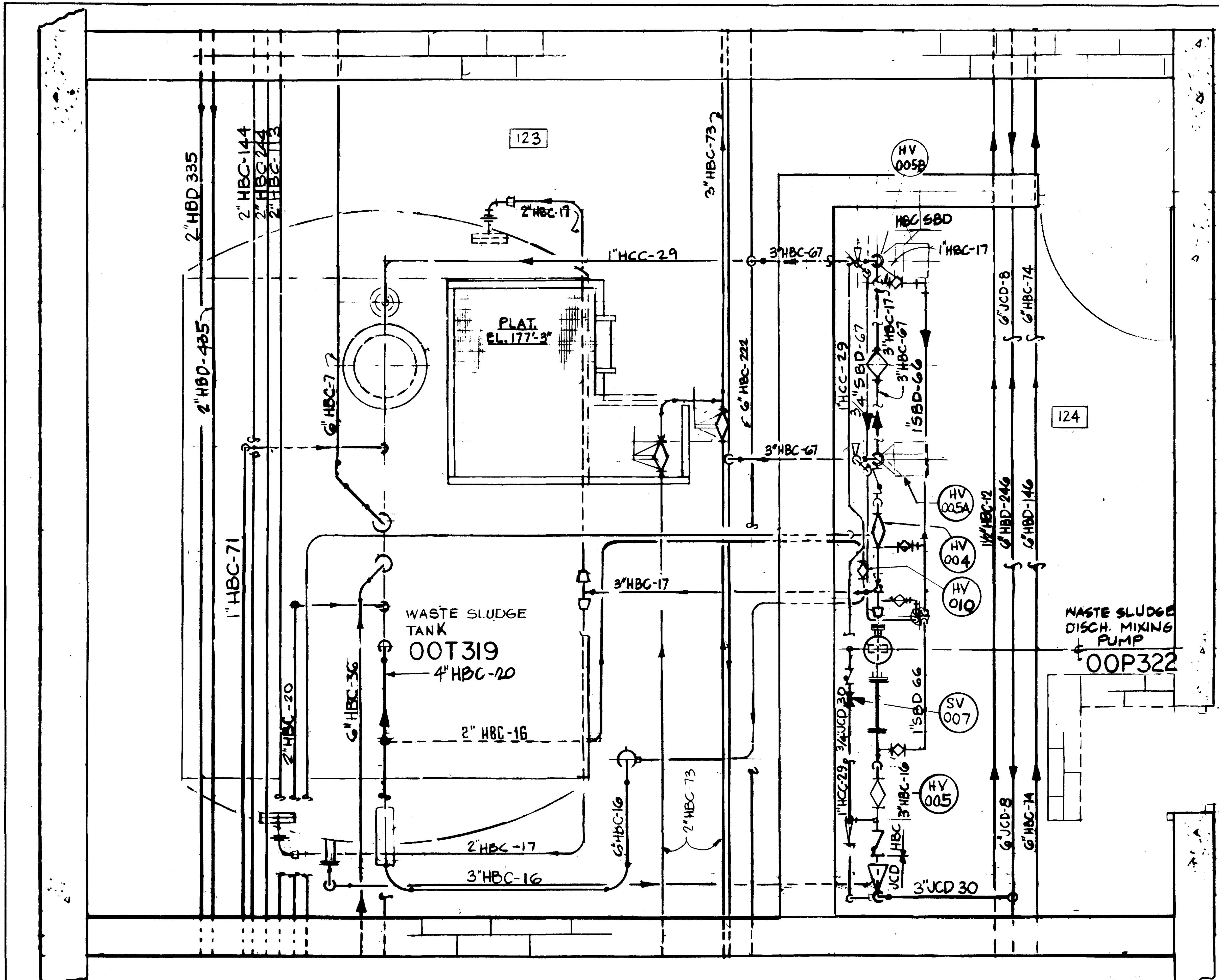
TYPICAL EQUIPMENT LAYOUT ARRANGEMENT DRAWING FOR FILTERS, DEMINERALIZERS AND VALVES

FIGURE 12.3-1



NOTES:

HV - REMOTE OPERATED CONTROL VALVE

SV - SOLENOID OPERATED VALVE

- PLUG VALVE

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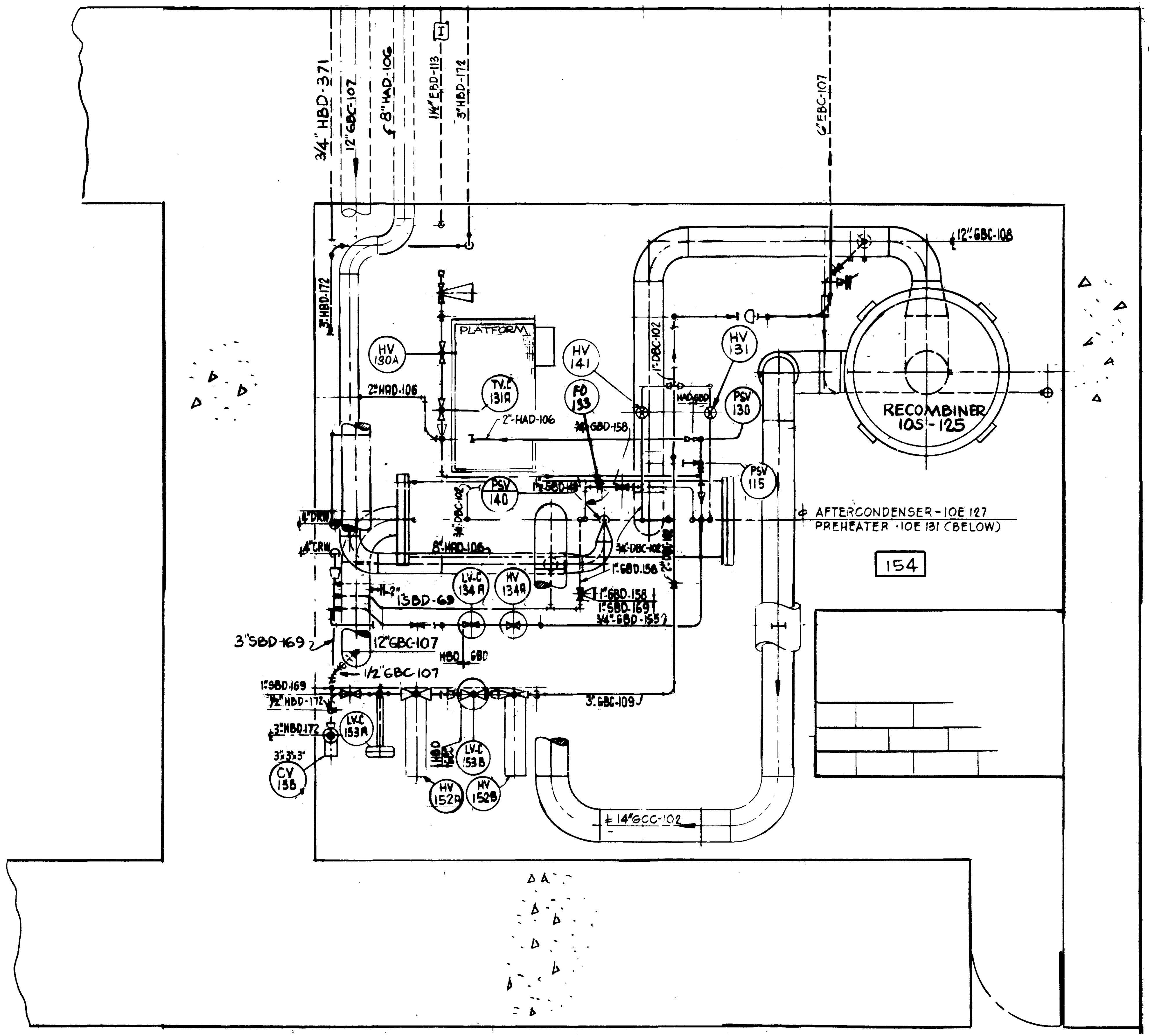
TYPICAL EQUIPMENT LAYOUT ARRANGEMENT DRAWING FOR RADWASTE TANKS AND PUMPS

FIGURE 12.3-2



NOTES:

HV–REMOTE OPERATED CONTROL VALVE
LV-C–LEVEL CONTROL VALVE
TV-C–TEMPERATURE CONTROL VALVE
CV–CONDUCTIVITY CONTROL VALVE
PSV–PRESSURE SAFETY VALVE

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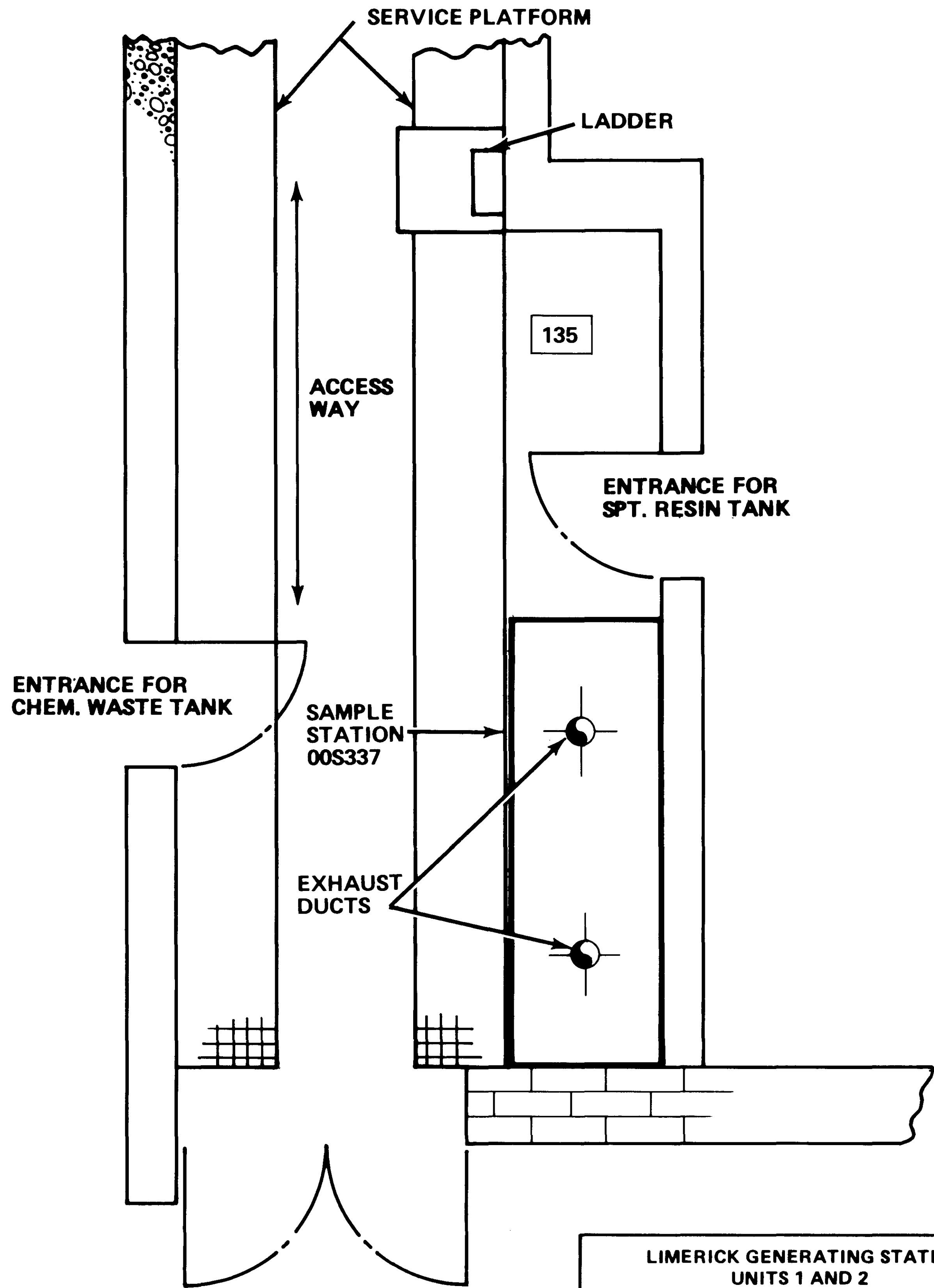
TYPICAL EQUIPMENT LAYOUT ARRANGEMENT DRAWING FOR OFFGAS RECOMBINERS

FIGURE 12.3-4



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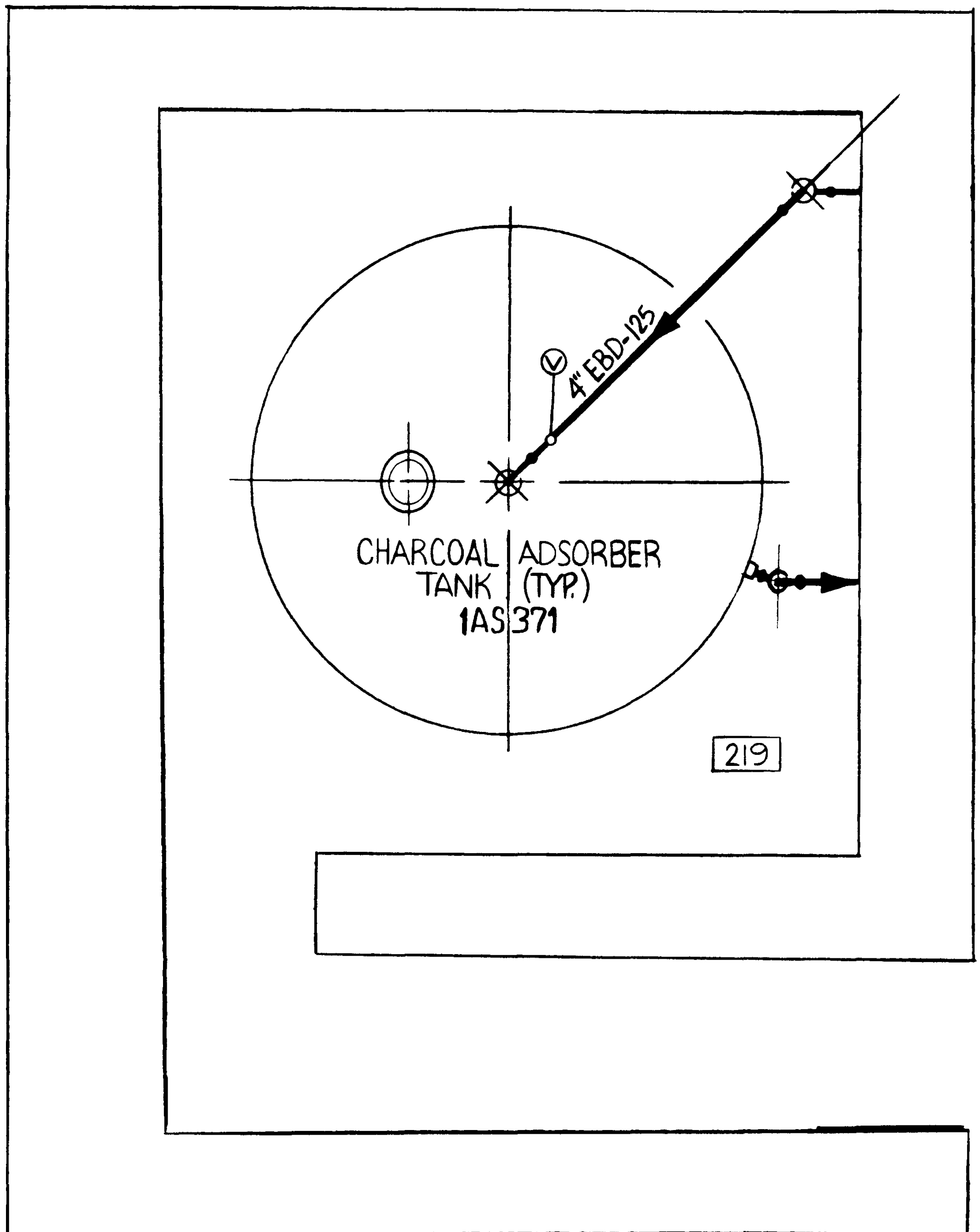
TYPICAL EQUIPMENT LAYOUT
ARRANGEMENT DRAWING
FOR SAMPLE STATIONS

FIGURE 12.3-5



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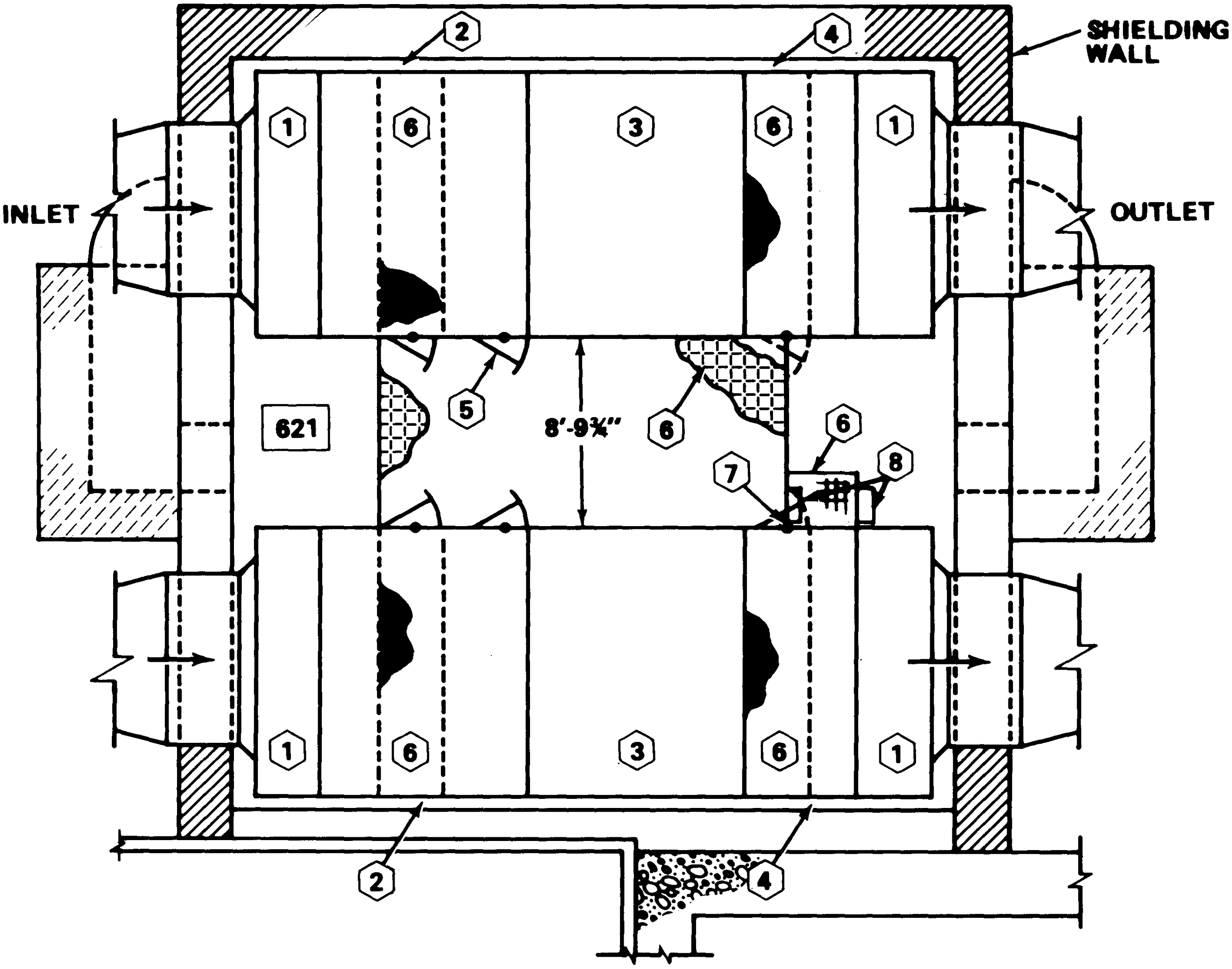
TYPICAL EQUIPMENT LAYOUT ARRANGEMENT DRAWING FOR CHARCOAL ADSORBER BEDS

FIGURE 12.3-6



(1) PLENUM

(2) PREFILTER AND HEPA FILTER BANK

(3) RECHARGABLE CARBON FILTERS

(4) HEPA FILTER BANK

(5) ACCESS DOORS

(6) PLATFORM

(7) TESTING CONNECTIONS

(8) LADDER

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TYPICAL SHIELDING ARRANGEMENT FOR CHARCOAL AND PARTICULATE FILTERS

FIGURE 12.3-7

Figures 12.3-8 thru 12.3-25

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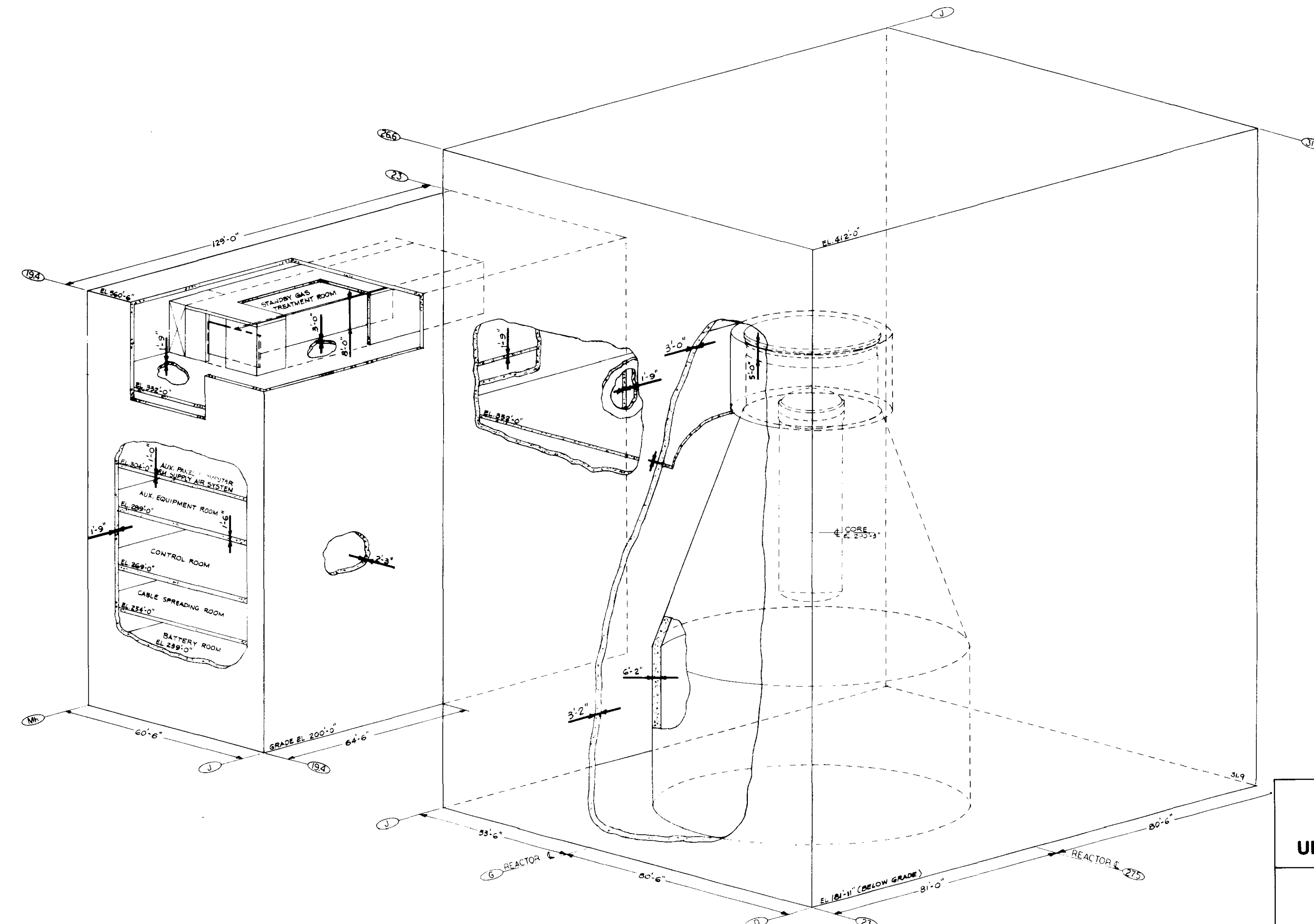

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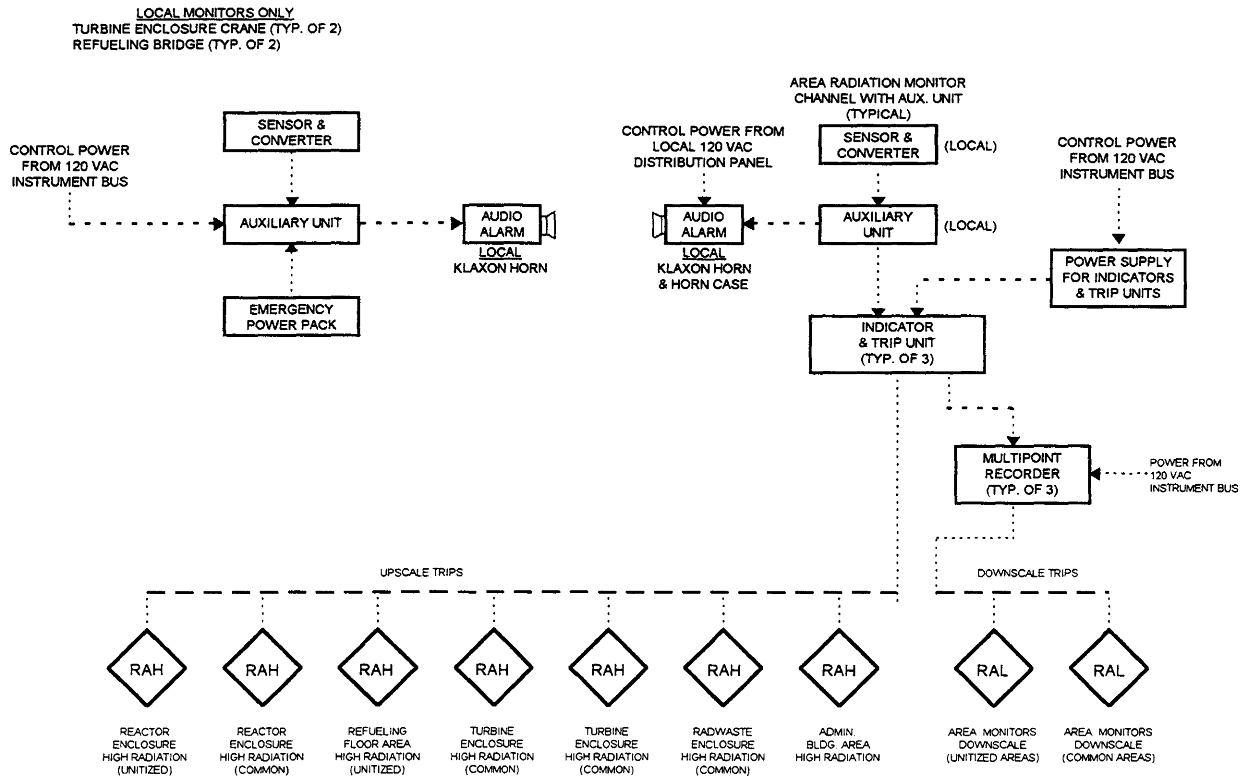
SCALED ISOMETRIC OF
CONTROL STRUCTURE WITH
RELATION TO CONTAINMENT

FIGURE 12.3-26



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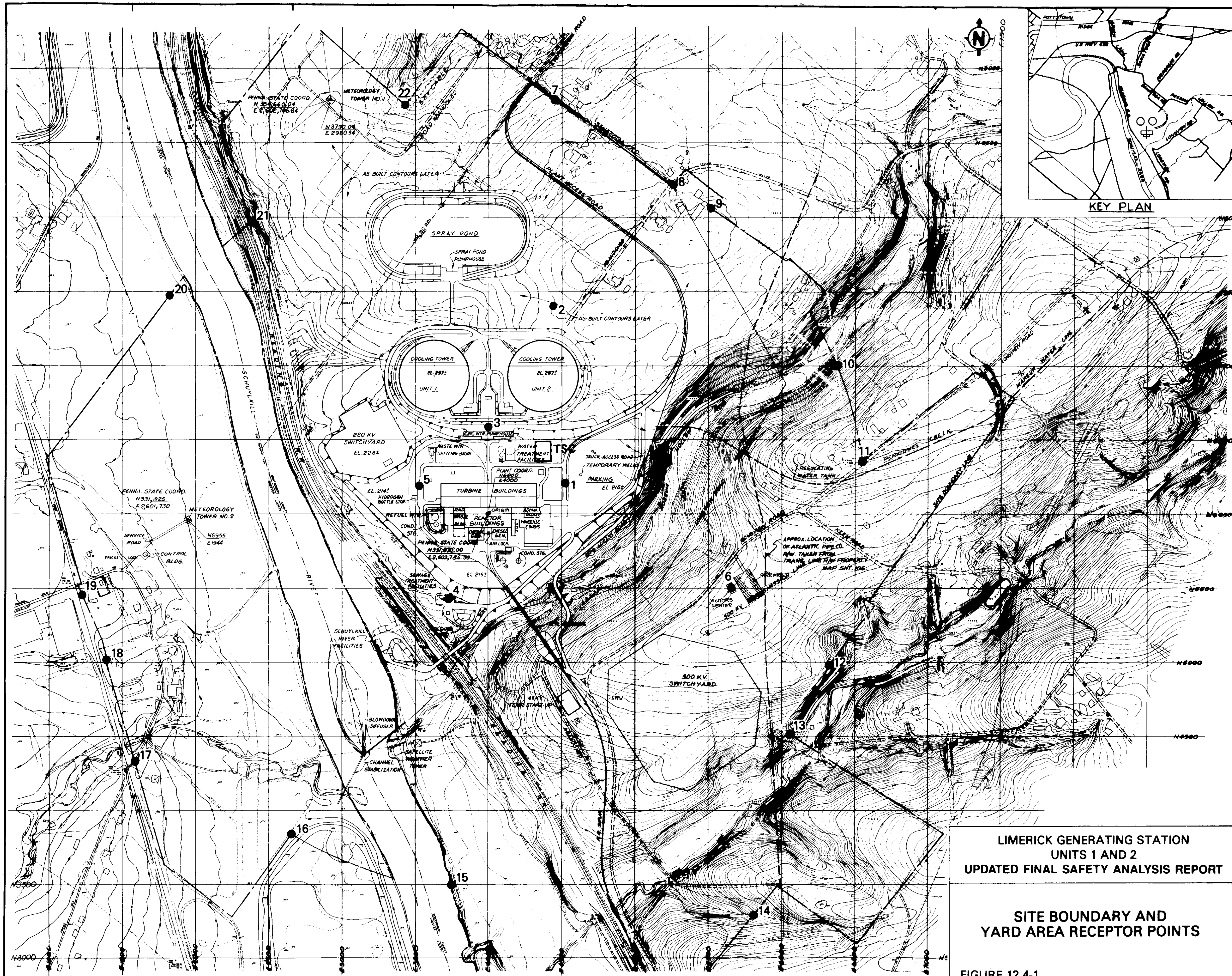
AREA RADIATION MONITORING SYSTEM SCHEMATIC DIAGRAM

FIGURE 12.3-27 Rev. 10 10/00



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SITE BOUNDARY AND
YARD AREA RECEPTOR POINTS

FIGURE 12.4-1