LaSalle UNITS 1 AND 2

UFSAR, REVISION 23

AND

FIRE PROTECTION REPORT (FPR), REVISION 8

THE SECURITY SENSITIVE INFORMATION HAS BEEN REDACTED FROM THE ORIGINAL DOCUMENT.

THIS DOCUMENT PROVIDES THE REDACTED VERSION.

THE REDACTED INFORMATION WITHIN THIS DOCUMENT IS INDICATED BY SOLID BLACKEDOUT REGIONS.

CHAPTER 12.0 - RADIATION PROTECTION

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

CHAPTER 12.0 - RADIATION PROTECTION

LIST OF TABLES

LIST OF TABLES (Cont'd)

CHAPTER 12.0 - RADIATION PROTECTION

LIST OF FIGURES AND DRAWINGS

FIGURES

DRAWINGS CITED IN THIS CHAPTER*

* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

CHAPTER 12.0 - RADIATION PROTECTION

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

12.1.1 Policy Considerations

It is the policy of EGC to maintain occupational radiation exposure as low as is reasonably achievable (ALARA), consistent with plant construction, maintenance, and operational requirements, and within the applicable regulations. This ALARA policy applies to total dose equivalent accumulated by all personnel, as well as to individual dose equivalents. EGC management provides the environment for this policy to function in a proper manner. Management's commitment to this policy is reflected in the design of the plant, the careful preparation of plant operating and maintenance procedures, the provisions for review of these procedures and for review of equipment design to incorporate the results of operating experience, and most importantly, the establishment of an ongoing training program. Training is provided for all personnel (Subsection 13.2.1), so that each individual is capable of carrying out his/her responsibility for maintaining his/her own exposure ALARA consistent with discharging his/her duties and also for observing rules adopted for his/her own radiological safety and that of others. The development of the proper attitudes and awareness of the potential problems in the area of health physics is accomplished by proper training of all plant personnel. The organizational structure related to assuring that occupational radiation exposure be maintained ALARA is described in Subsection 12.1.1.1.

12.1.1.1 Organization Structure

The operating organization structure of LaSalle County Station (LSCS) is described in Subsection 13.1.2. The structure of the radiation protection department and chemistry department is shown in Topical Report NO-AA-10 and approved organization charts as described in Section 5.2.1.a of Technical Specifications.

12.1.1.2 Personnel Activities and Responsibilities

The station Radiation Protection Supervisor is responsible for the health physics program and for handling and monitoring radioactive materials, including source and by-product materials. New and spent fuel is handled under the direction of a fuel handling foreman, who holds at least a limited senior reactor operator's license. In the case of operations which alter the configuration of the reactor core, licensed supervisory personnel are directly responsible for movement of the fuel.

12.1.1.3 Administration Concerns

The LaSalle County Station (LSCS) administrative personnel initially had a considerable amount of experience which was accumulated at operating stations. The health physics program is based on regulations and experience which includes and/or considers the following:

a. Detailed procedures are prepared and approved for radiation protection prior to reactor plant operation. Those procedures are a part of the station health physics program.

b. All outgoing shipments which may contain material are surveyed to assure compliance with 10 CFR 71, 10 CFR 73, and 49 CFR 1-190.

c. Any radiological incidents are thoroughly investigated and documented in order to minimize the potential for recurrence. Reports are made to the NRC in accordance with the Reportability Manual, 10 CFR 20.2202, and 10 CFR 20.2203.

d. Periodic radiation, contamination, and airborne activity surveys are performed and recorded to document radiological conditions. Records of the surveys are maintained in accordance with 10 CFR 20.2103.

e. Records of occupational radiation exposure are maintained and reports are made to the NRC as required by 10 CFR 20.2204 and 10 CFR 20.2206, and to individuals as required by 10 CFR 19.13.

f. Radiation, high radiation, and very high radiation areas are segregated and identified in accordance 10 CFR 20.1502, 10 CFR 20.1601 and 10 CFR 20.1602. Airborne activity is determined and posted in accordance with 10 CFR 20.1902. Positive control is exercised for each individual entry into high radiation areas and very high radiation areas.

g. Personnel are provided with personnel radiation monitoring equipment in accordance with 10 CFR 20.1501 and 10 CFR 20.1502 to measure their radiation exposure.

h. Process radiation, area radiation, portable radiation, and airborne activity monitoring instrumentation is periodically calibrated as required.

i. Access control points are established to separate potentially contaminated areas from uncontaminated areas of the station.

j. Protective clothing is used as required to help prevent personnel contamination and the spread of contamination from one area to another.

k. Items and materials should not be removed from radiologically posted areas until they have been surveyed or evaluated for radioactive contamination by a qualified individual. (A qualified individual is defined as a person meeting the radiation protection technician qualifications of RG 1.8, Rev. 1). The only exceptions are hand-carried personal effects (e.g., notebooks and flashlights) that are subject to the same survey requirements (e.g. via use of an automated monitor with preset alarm levels) as the individual processing them. Tools and equipment removed from a contaminated area are packaged to prevent the spread of contamination to other areas.

l. Radiation work permits (RWP) are issued for certain jobs in accordance with the station radiation protection procedures. Jobs involving significant radiation exposure to personnel are preplanned. (Where conditions dictate, a mock-up is used for practice to reduce exposure time on the actual job. The use of special tools and temporary shielding to reduce personnel exposure is evaluated on a job-by-job basis.)

m. A bioassay program is included as part of the health physics program. This program includes whole body counting and/or a urinalysis/fecal analysis sampling program to measure the uptake of radioactive material. A passive monitoring system also exists to detect the uptake of radioactive material above action level.

n. An environmental monitoring program is in operation to measure any effect of the station on the surrounding environment.

o. All significant radioactive effluent pathways from the station are monitored and records maintained.

12.1.2 Design Considerations

LSCS radiation protection design considerations establish a precise direction for maintaining radiation exposure to as low as reasonably achievable (ALARA). The direction is established by a set of radiation protection design goals. The maintaining of ALARA is accomplished by identifying problems and concerns associated with the plant and equipment. The third step applies the appropriate design criteria. The last step makes improvements wherever practicable.

12.1.2.1 Radiation Protection Design Goals

LSCS radiation protection design goals are directed to ensure compliance with the standards for radiation protection specified in 10 CFR 20. The following sequences of design goals are used:

a. Establish design dose rates for general access areas based upon EGC experience and 10 CFR requirements.

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

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

d. Determine shielding required to maintain design dose equivalent rates.

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

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

g. Make changes in design wherever practicable to achieve ALARA exposures.

12.1.2.2 Facility Design Considerations

LSCS's design goals are expanded to design objectives. These objectives are categorized into several radiation protection concerns. Station layout considers direct radiation (for this section, direct radiation is defined as scattered and unscattered gamma and/or neutron rays from a [several] nonairborne radiation source[s]), and ventilation considers airborne radiation. Health physics and access control are concerned with both direct and airborne radiation. Control of radiation fluids and effluents is concerned with the processing and detection of radioactive materials.

The design objectives are coupled with operating experience to obtain an improved station design.

12.1.2.2.1 Station Layout (Shielding)

The shielding is arranged and designed to the following objectives:

a. A sufficient quantity of access paths (general access areas) are to be furnished to allow personnel attendance to equipment.

b. The radiation levels in general access areas are to be kept ALARA.

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

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

e. Segregation of radiation zones is to be employed wherever practicable.

f. Shielding must accommodate equipment removal and maintenance.

g. Radiation "hot spots" are to be expected along the face of some shielding walls due to penetrations and imbedded voids (i.e., nonradioactive pipes and diagonal bracing). A radiation "hot spot" is a small area that has a (slightly) higher dose rate than the surrounding areas.

h. The radiation protection design is to be based upon the design criteria given in Section 12.3.

12.1.2.2.2 Ventilation

The station ventilation systems aid in heat removal and control of airborne radioactive material. Ventilation systems are designed to direct potentially airborne radioactive material away from occupied areas towards the station vent stack. The remaining HVAC systems have special functions, e.g., laboratory hood exhaust, drywell purge, and the standby gas treatment system (SGTS). The ventilation systems are described in greater detail in Section 9.4. The radiation protection aspects of the systems are discussed in Subsection 12.3.3.

12.1.2.2.3 <u>Access Control</u>

Access to radioactive equipment is designed so that properly trained radiation protection personnel can maintain ALARA radiation exposure to station workers during all modes of station operation. Access to radiation areas is to be strictly controlled.

12.1.2.2.4 <u>Health Physics</u>

The radiation protection design objectives for health physics are:

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

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

c. Radioactive effluent release paths to the environment are to be monitored.

d. Facilities for analysis of radioactive samples are to be furnished.

e. Cleaning and decontamination facilities are to be provided for equipment and personnel.

f. Periodic radiation surveys are to be performed when required, i.e., maintenance in radiation areas, receiving or shipping radioactive material, and decontamination and maintenance of equipment, parts, and tools.

12.1.2.2.5 <u>Control of Radioactive Fluids and Effluents</u>

Radioactive fluids (liquids and gases) are to be contained and controlled to keep the release of radioactive materials to general access areas and the environment ALARA. This objective applies to drain liquids, airborne radioactivity, and process liquids and gases (i.e., reactor water, fuel pool water, radwaste water, drywell purge, off-gas, and turbine seal). The number of release paths are minimized in order to simplify control.

The process liquids and gases are stored and/or processed within defined boundaries. Systems which operate at positive or negative gauge pressures have closed boundaries. During normal operation, fluids from such systems escape their boundary only through pressure control equipment and by leaking. Some systems which operate at atmospheric pressure may have openings in their boundary (vents).

The resulting airborne contaminants are directed away from high plant occupancy areas and through particulate filters to the elevated release point. The ventilation systems are discussed in Subsection 12.3.3. The resulting liquid contaminants are directed from rooms or areas where the leaking occurs to liquid radwaste storage tanks.

The equipment drain system is connected to the liquid collection points which are attached to most equipment. The collected liquid is directed to sump pumps which pump it to liquid radwaste.

The floor drain system is designed to handle large volumes of liquid (up to 300 gpm). Curbs are provided in component areas to prevent radioactive liquid that reaches the floor from contaminating low radiation areas, i.e., operating areas, general access areas, and corridors. The floor drain system in the reactor building segregates the radiation area floor drains from the remainder of the floor drains. Likewise, the liquid radwaste floor drain system is separated from the turbine building floor drain system, and the liquid radwaste tank curbing can hold in several instances one or two tank volumes in any two adjacent tank cubicles.

The ventilation and drain systems are designed to minimize the spread of contaminants to ALARA levels.

12.1.2.2.6 Safety Objectives

a. 10 CFR 20 limits are to be maintained for operating personnel and the general public.

b. 10 CFR 50 limits for the control room are to be met for a DBA and lesser accidents.

c. Radiation protection design objectives related to 10 CFR 100 are given in Chapter 15.0.

12.1.2.2.7 Improvements in Facility Design Due to Past Experience and Operation

EGC currently operates multiple licensed BWR's and PWR's. The operating experience obtained from seven of these reactors has been incorporated into the design of LSCS. In addition, published information on radiation problems and radiation protection (in nuclear power stations) is used to anticipate and minimize occupational radiation exposure. Experienced operating personnel have continually reviewed the station design as the design progressed and have provided recommendations based on their experience.

Routine survey data from EGC's operating stations has been used to correct or improve the design of LSCS. Design improvements directly attributed to experiences and operations are as follows:

a. An adequate number of equipment decontamination areas have been included to reduce congestion and reduce maintenance time.

b. A sample preparation laboratory was added.

c. A separate area is provided for handling, washing, inspecting, and storing of respiratory protective equipment.

d. Valves are shielded from radioactive components wherever practicable.

e. The valves are centrally located in valve aisles whenever practicable.

f. Valve operating stations are located in general access areas.

g. Remote operators (such as motors, reach rods, chains, etc.) are used on valves wherever practicable.

h. Shielding separates pumps from their associated tanks or other vessels.

i. Space and adequate floor strength for temporary shielding are supplied where practicable.

12.1.2.3 Equipment Design Considerations

Radiation protection design consideration of equipment involve shielding, equipment access, equipment selection, and equipment maintenance. Equipment design objectives deal with access to and segregation of radioactive equipment. Proper selection of equipment can reduce radiation exposure, and proper equipment design and selection of replaceable parts can reduce exposure due to maintenance.

12.1.2.3.1 Equipment Design Objectives

The following design objectives were incorporated into the design wherever it was practicable to do so:

a. locate equipment in accessible parts of cubicles;

b. keep equipment that operates infrequently in accessible areas, i.e., radwaste pumps;

c. provide galleries, gratings, and hatches to enhance accessibility to equipment located high above a floor;

d. provide access for easy removal of equipment requiring frequent changing;

e. provide localized shielding or space and adequate structure for localized shielding as part of the shielding design;

f. locate equipment which processes low radioactivity material in separate cubicles from equipment which processes high radioactivity material;

g. separate high from low radioactivity lines that connect to a single component;

h. use of unmortared removable block walls to minimize the radiation exposure in gaining access to highly radioactive components when removal is required; and

i. provide cranes or lifting lugs to aid in equipment servicing, maintenance, and removal.

In the selection of equipment that processes radioactive materials, consideration is given to minimizing leakage, spillage, and maintenance requirements. The radiological considerations for equipment selection and maintenance are given in the following subsections.

12.1.2.3.2 Equipment Selection

The selection of equipment to handle and process radioactive materials is based upon system requirements and radiation protection requirements. Material and coating selection are chosen for decontamination properties as well as durability. Some components which may become contaminated are designed with provisions for flushing or cleaning. Reduced occupational radiation exposure is attained by utilizing operating experience and where practical, providing prudent equipment selections such as:

a. plug valves which require less maintenance in place of diaphragm valves;

b. diaphragm seal valves which require no packing;

c. longer lived graphite-filled packing, instead of standard packing;

d. fluid connections for the capacity to back flush;

e. remote systems (or connections) for remote chemical cleaning where practicable;

f. air connections to tanks containing spargers to allow for air injection to uncake contaminants;

g. crossties between redundant equipment and/or related equipment capable of redundant operation to allow removal of contaminated equipment from service;

h. remote handling equipment to handle radioactive materials and components; and

i. special access to reduce personnel radiation exposure during maintenance and equipment removal, i.e., wall plugs and removable wall sections.

j. locate and route ducts and piping to minimize the buildup of contaminants.

12.1.2.3.3 <u>Equipment Maintenance</u>

In the design of the station and its equipment, provisions were incorporated to assure that the occupational radiation exposure is kept ALARA. Previously mentioned facility improvements and equipment selection also aid in reducing personnel radiation exposure from equipment maintenance. Additional facility improvements, not previously mentioned were added following experience that was accumulated in earlier BWR plants to aid operating personnel in reducing maintenance time when servicing and removing radioactive equipment. This reduced maintenance time supports the ALARA objective of reduced occupational radiation exposure. This category of improvements includes:

a. easily removable wall plugs used to enable particular maintenance operations, such as, tube plugging of low pressure heaters; and

b. small cylindrical passageways (ports) through labyrinth walls to enhance access to cubicles.

12.1.2.3.4 Servicing of Equipment and Instruments

The servicing of equipment that handles or is associated with radioactive fluids has been considered in the LSCS plant design. The types of servicing considered include maintenance, sampling, inservice inspection, equipment decontamination, instrument calibration, radiological surveys, and manual operation of equipment. Surveys, sampling, and inspections are performed frequently (i.e., weekly to daily); manual operation and calibration are performed occasionally (i.e., monthly to weekly); and decontamination and maintenance are performed infrequently (i.e., once or twice a year).

Whenever practicable, certain types of equipment are located in special aisles and rooms (pumps or valves) or in the lowest practicable radiation areas available (sampling stations or instruments). The design of special aisles and rooms consider equipment removal as well as servicing. Removal paths and methods are designed so that occupational exposures are ALARA.

The distance between equipment is made as large as practicable when segregation is not possible. In many cases the separation is sufficient to permit temporary shielding.

Equipment design requirements occasionally require instrumentation to be in the immediate vicinity of the equipment being monitored. Such equipment is located to minimize servicing time and to produce ALARA occupational exposure during in situ instrument calibration. Remote readouts are utilized when they do not increase servicing requirements.

Sampling stations are discussed in Subsection 12.3.1.8.8.

12.1.2.4 Compliance with Regulatory Guide 8.8 (Section C.3)

Special attention has been given to maintaining occupational radiation dose ALARA while establishing the final design of LSCS. The designs of facilities, equipment, structures, and access areas consider exposure obtained during routine operations (sampling, surveys, inspections, etc.), transient operations (changing power levels, startup and shutdown), operational, occurrences (identification, removal from service, etc.), maintenance, moving and storing radioactive materials, and accidents. These designs take into account equipment removal, decontamination, ventilation, orientation of equipment, in situ calibration and maintenance, sampling, monitoring, shielding, controlling contaminated fluids, minimizing leakage and spillage, and radiation exposure.

LSCS's initial staff includes experienced health physicists who have worked at other BWR stations. Their experience in radiation protection has been incorporated into the final design of LSCS during review and comment stages. In addition, design reviews have been conducted by other competent health physicists.

The design philosophy established for LSCS strives to maintain occupational radiation exposure ALARA and is in compliance with applicable regulations.

12.1.3 Operational Considerations

Operational radiation protection objectives deal with access to radiation areas, exposure to personnel, and decontamination. Working on or near highly radioactive components requires planning, special methods, and criteria to keep occupational radiation exposure ALARA. Special training prior to jobs and briefing following jobs contribute toward reduced exposures. Decontamination also helps to reduce exposure.

Implementation of procedures and techniques is based upon operational criteria and experience.

12.1.3.1 Operational Objectives

The operational radiation protection objectives include those given in Subsection 12.1.2.2.4 and the following, where applicable:

a. accurate knowledge of station designs;

b. sufficient experienced personnel to direct and train other personnel;

c. detailed job planning for high exposure work;

d. job simulations to improve productivity on the job, thereby keeping exposure ALARA;

e. briefings after jobs to identify time consuming work and to identify problems; and

f. improving procedures and techniques (defined in the following paragraph) for future jobs.

12.1.3.2 Implementation of Procedures and Techniques

The criteria and/or conditions under which various operating procedures and techniques for ensuring that occupational radiation exposures are ALARA for systems associated with radioactive liquids, gases and solids, along with the means for planning and developing procedures for radiation-exposure related operations, are given in the following:

a. Section 12.1 Assuring That Occupational Radiation Exposures Are ALARA,

b. Section 12.3 Radiation Protection Design Features, and

c. Section 12.5 Health Physics Program.

12.2 RADIATION SOURCES

The radiation protection and shielding designs for LaSalle County Station (LSCS) are based on radiation sources contained in the reactor core, reactor water, steam and process systems. The sources are based on both measurements and conservative assumptions. These sources provide a rational basis for design. The source data assumes that the station has operated at design power for a full cycle. In addition, the time dependency of shutdown sources is included.

12.2.1 Contained Sources

Contained sources are tabulated for the reactor vessel and for components which contain steam, condensate water, off-gas, reactor water, radwaste, and drain water. The radwaste and off-gas sources are given in Chapter 11.0. The remainder are given in the following subsections.

12.2.1.1 Reactor Vessel Sources

12.2.1.1.1 Core Sources

The physical data for the core and core sources were furnished by the NSSS vendor and are contained in Tables 12.2-1, 12.2-2, and 12.2-3.

12.2.1.1.2 Radioactive Sources in the Reactor Water

The radioactive sources in the reactor water are given in Table 12.2-4. The sources include fission products and activation products produced during normal full power operation. The postoperation reactor water sources are discussed in Subsection 12.2.1.3.

12.2.1.1.3 Radioactive Sources in the Steam

The radioactive isotopic concentrations in the steam leaving the reactor vessel are given in Table 12.2-5. The sources include activation products, radiogases, and radiohalogens which accompany the steam. The postoperation sources in the steam are discussed in Subsection 12.2.1.3.

12.2.1.2 Fluid Stream Sources Outside the Reactor Pressure Vessel

All of the radiation sources outside the reactor pressure vessel (RPV) are based on the sources originating in the RPV and carried beyond the vessel by the fluid stream. The source activities in a given component are dependent upon the time it takes the fluid stream to reach the component and the amount of process filtration and demineralization experienced by the fluid stream. Processing equipment accumulate the radioactive material extracted from the fluid stream. Local flow stream disturbances create minor "hot spots" in certain areas of fluid stream flow.

12.2.1.2.1 Radioactive Reactor Water Sources

The reactor water contains radioactive materials and leads to sources in associated cooling and process subsystems, e.g., recirculation, reactor water cleanup (RWCU), and residual heat removal subsystems. Reactor water is present in the residual heat removal system after shutdown. Table 12.2-6 gives the radioactive isotopic sources for the RWCU and equipment drain components in these subsystems. The postoperation sources decrease roughly on the same $e\text{-}\lambda^t$ bases, where λ is a decay constant and t is time.

12.2.1.2.2 Radioactive Steam

The steam contains activation products and fission products (primarily noble gases). The dominating isotope in the steam with respect to shielding considerations is nitrogen-16, formed by (n,p) reactions with oxygen-16. The nitrogen-16 concentration becomes insignificant compared to fission products 90 seconds after the steam leaves the reactor vessel. Therefore, the circuit time for the steam through the equipment is very important in determining the sources.

Tables 12.2-7 and 12.2-8 give the calculated nitrogen-16 accumulation for certain systems and components.

12.2.1.2.3 Radioactive Sources in Condensates and Feedwater

Steam condensate is collected in low pressure heaters, feedwater heaters, drain tanks, and the main condenser. The nitrogen-16 concentrations in the condensate and feedwater vary greatly (10^{-3}) to 10^{-20} μCi/ml) due to different flow patterns and transit times. Due to the heater drain tank and the main condenser holdup capacity, the condensate and feedwater returning to the reactor do not contain significant amounts of nitrogen-16. When the nitrogen-16 has decayed, radiohalogens become the dominating sources in the condensate and the feedwater.

12.2.1.2.4 Radioactive Sources in the Off-Gas System

The off-gas system uses core steam to pull condensable and noncondensable gases from the main condenser. The majority of the condensables (approximately 99%) are removed in the SJAE condenser, and the majority of the short half-life radionuclides (i.e., N-16, 0-19, etc.) decay in the cooler condenser to levels of 10^{-4} μCi/cc in the recombiner. Krypton and xenon are the principal radioactive sources for the residue vapors in the off-gas system. These sources are identified and given in Table 11.3-3.

12.2.1.2.5 Activity in the Radioactive Waste Processing System

The liquid radwaste storage facility collects and stores potentially radioactive fluids generated throughout the station due to leaks, cleaning, etc. These liquids can be processed through a radwaste solidification system which removes the majority (DF $\approx 10^4$) of the radioactive material, if necessary. This solid radioactive material is packaged for storage; the remaining liquid is sampled to determine whether it will be recycled, stored, or discharged. Radioactivity is present in varying amounts in tanks, filters, demineralizers, pipelines, and solidification components of the radwaste system.

The radioactive sources in the liquid and solid radioactive waste systems are identified in Sections 11.2 and 11.4, respectively.

12.2.1.2.6 Turbine Shine Sources

Nitrogen-16 (N-16) dose at the exclusion area boundary is due to air scattered radiation, referred to as "skyshine." Because of shielding, the direct radiation from the steam handling equipment on the turbine operating floor is small. The boundary dose rate due to "skyshine" results from the N-16 gamma rays that do not intercept the turbine shield wall which is 24 feet high on the north and south sides and 33 feet high on the west side of the turbine. In the east direction the auxiliary and reactor buildings act as a shadow shield.

The major source components are the high-pressure turbine, the low-pressure turbines, the low-pressure turbine control valves, and the connecting piping. The first two items in Table 12.2-8 give the nitrogen-16 sources used in the calculation of "skyshine" at the site boundary.

12.2.1.2.7 Fuel Pool Sources

The radioactive isotopes in the fuel pool come from three sources:

a. leaks in stored spent fuel,

b. the mixing of fuel pool water with reactor water at the start of refueling, and

c. cleaning and decontaminating NSSS components and parts.

The activity in the pool is decreased by the fuel pool cooling and cleanup system (FPCCS) and by decay. The cleanup rate (in volumes/sec) is greater in the FPCCS than in the reactor water cleanup (RWCU) system resulting in lower equilibrium concentrations.

The fuel pool sources are at a maximum during fuel unloading of the core, because the sources listed previously occur simultaneously.

The design-basis source concentrations for the FPCCS are: 20% of the short lived isotopes (<8 days) and 50% of the remaining isotopes in Table 12.2-4, decayed for 1 day. The sources in the pool during normal operation contain no significant quantities of short lived isotopes and one-half the design basis concentrations for the remainder of the isotopes.

12.2.1.3 Postoperation Sources

Postoperation begins with the end of reactor shutdown (control rod completely inserted into the core) and terminates when the reactor is reactivated. During postoperation periods, a large portion of the station's maintenance is performed. The dominating sources during postoperation are:

a. reactor water in the RHR system,

b. activity in the fuel pool water,

c. radwaste,

d. crud buildup, and

e. plateout.

12.2.1.3.1 RHR Sources

The RHR system is usually activated 3 or more hours after reactor shutdown. Table 12.2-9 gives the major radioactive isotopes in the RHR heat exchanger. This table also gives the concentrations for several days of operation (each isotope decays as $e^{-\lambda t}$).

In addition to the sources in the reactor water, the RHR system accumulates crud in the piping and components. Crud buildup is discussed in Subsection 12.2.1.3.4.

12.2.1.3.2 Fuel Pool Sources

The design-basis sources for the FPCCS also apply during postoperation. These sources were discussed in Subsection 12.2.1.2.7.

Crud buildup is also an important source during postoperation, especially for piping and heat exchangers. Crud buildup is tabulated in Subsection 12.2.1.3.4.

12.2.1.3.3 Radwaste Sources

The radwaste postoperation sources are the same as the sources given in Table 11.2-5. Crud buildup sources become important after radwaste equipment is drained.

12.2.1.3.4 Crud Buildup Sources

Crud buildup occurs in all systems that handle radioactive fluid streams. Low activity streams seem to be dominated by long half-life isotopes, so that, after a number of years, the crud buildup sources become the dominant sources. For high activity streams, crud buildup sources are significant during postoperation periods.

Operating BWR stations have collected person-rem data (Table 12.4-2) which shows an increase in station exposure from one year to the next. Whether or not crud buildup increases forever or converges to some peak value has not been determined.

Shielding for crud buildup is based on measurements taken at Dresden NPS and the Quad-Cities Station. Some of the values are:

Component	mrem/hr
RHR heat exchanger	120-2,000
RHR piping	100-5,000
Fuel pool heat exchanger	120-2,000
Fuel pool piping	60-150
Radwaste sludge piping	500-10,000
Reactor building equipment drains	100-400
Strainers	to 10,000

The high values occur at crud traps (pipe elbows, pipe reducers, etc.), and the data is for 2 or 3 years of operation. Some of the values may also include standing water which is radioactive.

Crud buildup can be removed chemically or by abrasives. Decontamination of entire systems may be required if crud buildup sources become too large.

12.2.1.3.5 Plateout Sources

Plateout sources are permanently bounded to the piping or components. Some of these sources can be removed chemically, but not all.

The plateout sources are estimated to be 10% of the crud buildup sources, and they are only significant after system decontamination.

Refer to section 10.4.7.2 for a discussion of the zinc injection system used for reduction of plate-out sources in the recirculation piping and components.

12.2.1.3.6 Drywell Sources

The drywell sources are given in Table 12.2-2 in MeV/(sec-watt); an additional postoperation drywell source includes the reactor water in the recirculation piping, which decays approximately as $e^{-\lambda t}$. (See Table 12.2-4 for reactor water sources and decay constants.)

12.2.1.4 Postaccident Sources

The fission product sources and the assumptions used during design-basis accidents are given in Chapter 15.0.

12.2.2 Airborne Radioactive Material Sources

With the exception of noble gases, airborne radioactive sources are generated from radioactive liquid sources by the mechanisms discussed in the next subsection. The generation of airborne radioactivity in radiation areas can affect the areas normally accessible to operating personnel, mainly pump and valve areas. The airborne radioactivity during normal operation for accessible areas is discussed in Subsection 12.2.2.3. The calculational model is given in Subsection 12.2.2.4.

In addition to affecting the station, the airborne radioactive material which exits the filtration systems enters the environment via the station vent stack.

12.2.2.1 Production of Radioactive Airborne Material

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

COMPONENT	AIRBORNE METHOD
fuel pool	evaporation
radwaste	evaporation (venting)
high energy line leak	vapor, evaporation
spray from high energy equipment	vapor, droplets, evaporation
low energy line leak	evaporation
spill	droplets, evaporation

The major contributors to airborne radioactivity during normal operation are: (1) leaks in the RWCU system, (2) evaporation from fuel pool, (3) leaks in radwaste systems, (4) venting of radwaste tanks, and (5) leaks in the off-gas system. Minor contributors are: (1) cleaning and decontaminating tools and equipment, (2) contaminated wearing apparel, (3) turbine seal system, (4) sample preparation and analysis, and (5) leaks in the condensate and feedwater systems.

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

12.2.2.2 Sources in Areas Normally Accessible to Operating Personnel

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

The ventilation flow path is from low potential airborne areas to high potential airborne areas. The ventilation system has been designed to control the radioactive airborne activity in the laboratories, maintenance areas, and the refueling floor of the reactor building. The level of airborne radioactivity is periodically determined by the radiation protection staff to ensure that radiation exposure is ALARA. Except for the off-gas building and the heater bays, the most significant radioactive isotopes are the halogens (primarily iodine). The iodines have the highest concentration in relation to the concentration limits for liquids.

Maintenance accounts for a sizeable portion of the internal exposure because station personnel have to perform many of these functions in areas with relatively high airborne radioactivity. The airborne radioactivity is caused by leaks, spills, venting, etc. The airborne radioactivity concentrations are calculated for the occurrences that are the most common, namely, leaks and venting. These concentrations are given in the next subsection. Infrequent anticipated operational occurrences and abnormal occurrences are handled in the manner established in the personnel internal exposure program (Subsection 12.5.3.3).

12.2.2.3 Calculated Concentrations During Operation

The calculated concentrations of airborne radioactive material in normally accessible cubicles are based upon the model given in Subsection 12.2.2.4. These concentrations are given in Tables 12.2-10, 12.2-11, and 12.2-12. The general access areas have very little if any airborne contaminants (i.e., $<10^{-12}$ μCi/cc) during normal operation except for those mentioned in Subsection 12.2.2.2. Concentrations in normally accessible areas are maintained ALARA by periodic air sampling as specified in the health physics program.

12.2.2.4 Models and Parameters Used in Calculations of Concentration

The equation used to calculate the equilibrium airborne concentrations during normal operation is as follows:

$$C_A = \frac{(\dot{L})(C\ell)(P)}{7.48(\lambda V + F)} \qquad \text{(Equation 12.2-1)}$$

where:

C_A	=	airborne concentrations in each cubicle (μCi/cc)
$\dot{L}$	=	leak rate (gpm)
$C\ell$	=	liquid concentration (μCi/cc)
P	=	Fraction of activity released to air
7.48 is conversion factor (7.48 gal/ft^3)		
λ	=	decay constant (min^{-1})
V	=	enclosed volume (ft^3)
F	=	Air exhaust flow rate (ft ^{3}/min).

12.2.2.5 Stack Effluents

All ventilation system exhausts (except for office space and service building HVAC) are routed to the station vent stack. Each ventilation system is designed to exhaust into the station vent stack simultaneously with all other ventilation systems. Ventilation systems containing potentially high airborne radioactivity levels are provided with filters specifically designed to hold-up or remove radioactive material (Section 9.4).

The dominating radioisotopes released through the station vent stack are the noble gases from the off-gas system and the noble gas leakage expected in the turbine building. The expected yearly releases during normal operation are discussed in Section 11.3.

12.2.2.6 Standy Gas Treatment System Sources

The standby gas treatment system (SGTS) is normally used only during accidents. The greatest quantity of sources becomes available during a design-basis accident (DBA). The DBA and its radioactive releases to the environment are discussed in Chapter 15.0.

12.2.2.7 Airborne Sources on Refueling Floor

The airborne radioactivity during refueling at LSCS is assumed to be similar to the activities experienced at Dresden and Quad-Cities Stations. Experience at these stations has shown that some airborne radioactivity can result from the following:

a. When the reactor water in the reactor cavity goes above 100°F, noticeable increases in the I-131 airborne activity result; increasing with temperature.

b. When the reactor water level in the vessel is low, previously covered metallic surfaces dry out. If cobalt dioxide (CoO_2) is plated out on these surfaces, air moving across these surfaces can dislodge fine particles of CoO_2. The drier and separator are also susceptible to this phenomena.

These two airborne activity problems have been solved by maintaining water temperatures below 100°F and by preventing probable CoO_2 contaminated surfaces from drying out.

Recent data from a Dresden CAM located on the refueling floor near the fuel pool is given in Table 12.2-13. The data covers a 40-day outage which occurred from September 20, 1976 to October 29, 1976.

Quad-Cities has experienced slightly higher levels of beta activity (up to 1×10^{-10} μCi/cc). Their iodine levels are similar to Dresden.

The RPV head is provided with a normally open vent that vents non-condensables to a main steam header during normal plant operation. Before removing the RPV head during shutdown, a separate head vent is connected directly to the containment vent outlet through temporary ducting. Therefore, any radioactive gases that may have accumulated in the RPV head are vented through the containment atmosphere cleanup system, and do not pose any airborne exposure problem to plant personnel.

Safety-relief venting occurs during normal plant operations when the primary containment is not normally accessible. All SRV discharge lines are piped to the suppression pool and discharge under water. Any accumulation of non-condensables will be removed from the suppression chamber by the primary containment vent and purge system prior to entering the containment.

TABLE 12.2-1

BASIC REACTOR DATA*

Reactor thermal power:			3293 MW
Overall average core power density:			50.7 W/cm^3
Core power peaking factors:			
At core center:	$Pmax/Pavg_Z$ (axial)	=	1.5
	$Pmax/Pavg_R$ (radial)	=	1.4
At core boundary:	$Pmax/Pavg_Z$ (axial)	=	0.5
	$Pmax/Pavg_R$ (radial)	=	0.7

Core volume fractions:

Material	Density (g/cm^3)	Volume Fraction
UO_2	10.4	0.254
Zr	6.4	0.140
H_20	1.0	0.274
Void	0	0.332

Average water density between core and vessel and below the core: 0.74 g/cm^3

Average water-steam density above core:

In the plenum region:	0.23 g/cm^3
Above the plenum (homogenized):	0.6 g/cm^3

Average steam density:	0.036 g/cm^3

* This table represents the initial (3293 MWT) physical data required to form the reactor vessel model which includes volume fractions, reactor power, and power distribution.

TABLE 12.2-2
(SHEET 1 OF 2)

GAMMA RAY SOURCE ENERGY SPECTRA

GAMMA RAY SOURCES IN THE CORE DURING OPERATION*

AND

POSTOPERATION GAMMA SOURCES IN CORE** (MeV/sec-watt)

ENERGY BOUNDS	GAMMA RAY SOURCE (MeV/sec-watt)
16.5 MeV	3.0×10^{8}
8.0 MeV	4.4×10^{9}
6.0 MeV	3.9×10^{10}
4.0 MeV	4.9×10^{10}
3.0 MeV	4.6×10^{10}
2.6 MeV	6.1×10^{10}
2.2 MeV	6.8×10^{10}
1.8 MeV	8.0×10^{10}
1.4 MeV	9.0×10^{10}
1.0 MeV	1.4×10^{11}
0.5 MeV	7.6×10^{10}
0.0 MeV	

* This table gives the average full power gamma energy spectra in the original (3293 MWt) operating core and the gamma ray energy spectra for postoperation times. The operating core spectra represents the prompt fission plus the equilibrium fission product gammas and the gammas due to neutron interactions with structural materials. The postoperating spectra gives the fission product decay gammas per watt of core power for several times after shutdown at the core midplane.

** Operating history of 3.2 years.

TABLE 12.2-2
(SHEET 2 OF 2)

(POSTOPERATION GAMMA SOURCES IN CORE)**

ENERGY BOUNDS	O SEC	TIME AFTER SHUTDOWN 1 DAY	1 WEEK	1 MONTH
6.0 MeV				
	8.2×10^{10}	$<1.0 \times 10^{6}$	$<1.0 \times 10^{6}$	$<1.0 \times 10^{6}$
4.0 MeV				
	1.8×10^{10}	7.0×10^{6}	4.6×10^{6}	$<1.0 \times 10^{6}$
3.0 MeV				
	1.1×10^{10}	5.7×10^{6}	6.3×10^{6}	$<1.0 \times 10^{6}$
2.6 MeV				
	1.7×10^{10}	2.9×10^{8}	1.7×10^{6}	$<1.0 \times 10^{6}$
2.2 MeV				
	2.1×10^{10}	4.5×10^{8}	4.0×10^{7}	5.2×10^{6}
1.8 MeV				
	3.3×10^{10}	3.1×10^{9}	2.1×10^{9}	6.4×10^{8}
1.35 MeV				
	3.7×10^{10}	2.3×10^{9}	1.6×10^{9}	1.1×10^{9}
0.9 MeV				
	5.1×10^{10}	7.5×10^{9}	3.8×10^{9}	2.1×10^{8}
0.4 MeV				
	1.2×10^{10}	1.8×10^{9}	8.7×10^{8}	3.6×10^{8}
0.1MeV				

** Operating history of 3.2 years.

TABLE 12.2-3
(SHEET 1 OF 2)

GAMMA RAY AND NEUTRON FLUXES

OUTSIDE THE VESSEL WALL*

OUTSIDE VESSEL AT MIDPLANE

GAMMA ENERGY (MeV)	GAMMA FLUX (MeV/cm^2-sec)
1.0	3.0×10^7
1.5	1.7×10^8
2.3	1.1×10^9
3.0	7.4×10^9
5.0	2.2×10^9
7.0	4.4×10^9

Fast neutron flux $\geq$ 1.0 MeV = 1.0×10^7 neutrons/cm^2-sec

OUTSIDE VESSEL AT TOP HEAD

GAMMA ENERGY (MeV)	GAMMA FLUX (MeV/cm^2-sec)
1.0	9.4×10^2
1.5	6.9×10^3
2.3	9.4×10^4
3.0	9.3×10^5
5.0	1.2×10^6
7.0	4.7×10^6

Fast neutron flux $\geq$ 1.0 MeV = 4.3×10^{-1} neutrons/cm^2-sec

* This table presents original (3293 MWt) reactor core neutron fluxes and gamma ray energy fluxes at the outside of the reactor vessel. The fast neutron flux was calculated using an Albert-Welton attenuation kernel. The gamma fluxes are direct core fluxes only. Fluxes resulting from scattering beyond the reactor vessel are not included.

TABLE 12.2-3
(SHEET 2 OF 2)

OUTSIDE VESSEL AT BOTTOM HEAD

GAMMA ENERGY (MeV)	GAMMA FLUX (MeV/cm^2-sec)
1.0	7.4×10^{-4}
1.5	3.2×10^{-1}
2.3	1.2×10^{2}
3.0	4.5×10^{3}
5.0	4.0×10^{4}
7.0	3.1×10^{5}

Fast neutron flux ≥ 1.0 MeV = 4.7×10^{-7} neutrons/cm^2-sec

TABLE 12.2-4
(SHEET 1 OF 2)

DESIGN BASES REACTOR WATER SOURCES

COOLANT ACTIVATION PRODUCTS
(Equilibrium Values - Entering Recirculation Lines)

ISOTOPE	HALF-LIFE	DECAY CONSTANT (second^{-1})	CONCENTRATION (μCi/g)
N-16	7.13 sec	9.7216-2	$4.8 \times 10^{+1}$

NONCOOLANT ACTIVATION PRODUCTS

ISOTOPE	HALF-LIFE	DECAY CONSTANT (second^{-1})	CONCENTRATION (μCi/g)
Na-24	15 hr	1.2826-5	2×10^{-3}
P-32	14.31 day	5.6063-7	2×10^{-5}
Cr-51	27.8 day	2.8858-7	5×10^{-4}
Mn-54	313 day	2.563-8	4×10^{-5}
Mn-56	2.582 hr	4.9714-4	5×10^{-2}
Co-58	71.4 day	1.1236-7	5×10^{-3}
Co-60	5.258 yr	4.18-9	5×10^{-4}
Fe-59	45 day	1.7828-7	8×10^{-5}
Ni-65	2.55 hr	7.5506-5	3×10^{-4}
Zn-65	243.7 day	3.292-8	2×10^{-6}
Zn-69m	13.7 hr	1.4054-5	3×10^{-5}
Ag-11Om	253 day	3.171-8	6×10^{-5}
W-187	23.9 hr	8.0561-6	3×10^{-3}
			6.2×10^{-2}

FISSION PRODUCTS HALOGENS

ISOTOPE	HALF-LIFE	DECAY CONSTANT (second^{-1})	CONCENTRATION (μCi/g)
Br-83	2.40 hr	8.0225-5	1.5×10^{-2}
Br-84	31.8 min	3.6328-4	2.7×10^{-2}
Br-85	3.0 min	3.8508-3	1.7×10^{-2}
I-131	8.065 day	9.9474-7	1.3×10^{-2}
I-132	2.284 hr	8.4300-5	1.2×10^{-1}
I-133	20.8 hr	9.2568-6	8.9×10^{-2}
I-134	52.3 min	2.2089-4	2.4×10^{-1}
I-135	6.7 hr	2.8738-5	1.3×10^{-1}
			6.5×10^{-1}

TABLE 12.2-4
(SHEET 2 OF 2)

FISSION PRODUCTS - OTHER ISOTOPES

ISOTOPE	HALF-LIFE	DECAY CONSTANT (second^{-1})	CONCENTRATION (μCi/g)
Sr-89	50.8 day	1.5792-7	3.1×10^{-3}
Sr-90	28.9 yr	7.605-10	2.3×10^{-4}
Sr-91	9.67 hr	1.9911-5	6.9×10^{-2}
Sr-92	2.69 hr	7.1577-5	1.1×10^{-1}
Zr-95	65.5 day	1.2248-7	4.0×10^{-5}
Zr-97	16.8 hr	1.1461-5	3.2×10^{-5}
Nb-95	35.1 day	2.2856-7	4.2×10^{-5}
Mo-99	66.6 hr	2.8910-6	2.2×10^{-2}
Tc-99m	6.007 hr	3.2053-5	2.8×10^{-1}
Tc-101	14.2 min	8.1355-4	1.4×10^{-1}
Ru-103	39.8 day	2.0157-7	1.9×10^{-5}
Ru-106	368 day	2.1800-8	2.6×10^{-6}
Te-129m	34.1 day	2.3527-7	4.0×10^{-5}
Te-132	78 hr	2.4685-6	4.9×10^{-2}
Cs-134	2.06 yr	1.0670-8	1.6×10^{-4}
Cs-136	13 day	6.1712-7	1.1×10^{-4}
Cs-137	30.2 yr	7.278-10	2.4×10^{-4}
Cs-138	32.2 min	3.5877-4	1.9×10^{-1}
Ba-139	83.2 min	1.3885-4	1.6×10^{-1}
Ba-140	12.8 day	6.2676-7	9.0×10^{-3}
Ba-141	18.3 min	6.3128-4	1.7×10^{-1}
Ba-142	10.7 min	1.0797-3	1.7×10^{-1}
Ce-141	32.53 day	2.4662-7	3.9×10^{-5}
Ce-143	33.0 hr	5.8346-6	3.5×10^{-5}
Ce-144	284.4 day	2.8209-8	3.5×10^{-5}
Pr-143	13.58 day	5.9076-7	3.8×10^{-5}
Nd-147	11.06 day	7.2537-7	1.4×10^{-5}
Np-239	2.35 day	3.4138-6	2.4×10^{-1}
			1.61×10^{0}

TABLE 12.2-5

SOURCES IN THE REACTOR STEAM*

COOLANT ACTIVATION PRODUCTS

ISOTOPE	HALF-LIFE	CONCENTRATION (μCi/g)	RELEASE RATE (μCi/sec)
N-16	7.13 sec	2.5×10^2	$5 \times 10^{+8}$

FISSION PRODUCT HALOGENS

2% of the values given in Table 12.2-4

OTHER FISSION PRODUCTS AND ACTIVATION PRODUCTS

<0.1% of the values given in Table 12.2-4

EMISSION RATES OF NOBLE GASES

These rates are based on sufficient fuel cladding defects to result in a total off-gas release rate of 100,000 μCi/sec after 30-minute decay.

Isotope	Half-Life	Release Rate @ t = 0 (μCi/sec)
Kr-83m	1.86 hr	3.4×10^3
Kr-85m	4.4 hr	6.1×10^3
Kr-85	10.74 yr	10 to 20 *
Kr-87	76 min	2.0×10^4
Kr-88	2.79 hr	2.0×10^4
Kr-89	3.18 min	1.3×10^5
Kr-90	32.3 sec	2.8×10^5
Kr-91	8.6 sec	3.3×10^5
Kr-92	1.84 sec	3.3×10^5
Kr-93	1.29 sec	9.9×10^4
Kr-94	1.0 sec	2.3×10^4
Kr-95	.5 sec	2.1×10^3
Kr-97	1 sec	1.6×10^1
Xe-131m	11.96 day	1.5×10^1
Xe-133m	2.26 day	2.9×10^2
Xe-133	5.27 day	8.2×10^3
Xe-135m	15.7 min	2.6×10^4
Xe-135	9.16 hr	2.2×10^4
Xe-137	3.82 min	1.5×10^5
Xe-138	14.2 min	8.9×10^4
Xe-139	40 sec	2.8×10^5
Xe-140	13.6 sec	3.0×10^5
Xe-141	1.72 sec	2.4×10^5
Xe-142	1.22 sec	7.3×10^4
Xe-143	.96 sec	1.2×10^4
Xe-144	9 sec	5.6×10^2
	TOTALS	$\sim 2.5 \times 10^6$

* Estimated from experimental observations.
The steam flow rate is $\sim 1.79 \times 10^6$ g/sec.

TABLE 12.2-6
(Historical)

RADIOACTIVE ISOTOPIC INVENTORIES IN
REACTOR BUILDING COMPONENTS

ISOTOPE	RWCU REGENERATIVE HEAT EXCHANGER	RWCU NONREGENERATIVE HEAT EXCHANGER	REACTOR BUILDING EQUIPMENT DRAIN TANK
N-13	0.02	0.01	0.28
N-16	0.07	< 0.01	< 0.01
O-19	0.04	< 0.01	< 0.01
F-18	< 0.01	< 0.01	0.01
Na-24	< 0.01	< 0.01	0.04
Mn-56	0.05	0.03	0.03
Co-58	0.005	0.003	0.09
Co-60	< 0.01	< 0.01	< 0.01
W-187	< 0.01	< 0.01	0.06
Br-83	0.02	0.01	0.06
Br-84	0.03	0.02	0.02
Br-85	0.02	0.01	< 0.01
Sr-89	< 0.01	< 0.01	0.06
Sr-91	0.07	0.04	1.31
Sr-92	0.12	0.06	2.10
Mo-99	0.03	0.02	0.42
Tc-99m	0.28	0.14	5.32
Tc-101	0.14	0.07	0.05
I-131	0.02	0.01	0.25
Te-132	0.05	0.03	0.93
I-132	0.12	0.06	0.45
I-133	0.09	0.05	1.70
I-134	0.24	0.12	0.34
I-135	0.13	0.07	1.31
Cs-137	< 0.01	< 0.01	< 0.01
Cs-138	0.19	0.10	0.17
Ba-139	0.16	0.08	0.36
Ba-140	< 0.01	< 0.01	0.17
Ba-141	0.17	0.09	0.08
Ba-142	0.17	0.09	0.05
Np-239	0.24	0.12	4.56

RWCU PUMPS
N-16 = 0.05 Ci
Other Isotopes = 0.05 Ci

RWCU VALVES
N-16 = 0.03 Ci
Other Isotopes = 0.09 Ci

TABLE 12.2-7

(Historical)

NITROGEN-16 INVENTORIES IN TURBINE SUBSYSTEMS*
(Values are based upon ~94 Ci/sec leaving the RPV)**

Main Steam (Steam only)		449
Moisture Separator Drain		63
Moisture Separtor First Reheat		28
Moisture Separator Second Reheat		28
Heaters		
First Stage		17
Second Stage		19
Third Stage		20
Fourth Stage		9
Fifth Stage		28
Sixth Stage		41
Condenser and hotwell		251
Feedwater Turbines		6.3
Off-gas		1.3
Radwaste Reboiler		2.3
Seal Steam		1.1
	TOTAL	964

* Assume slug flow; values are given in curies.

** The N-16 source rate was rounded to 94 Ci/sec for the power uprate. This results in a source factor of 1.055.

TABLE 12.2-8

(Historical)

DESIGN BASIS NITROGEN-16 INVENTORIES IN TURBINE BUILDING COMPONENTS

Component	N-16 Curies*
High pressure Turbine (1)	11
Low Pressure Turbine (2)	17
Condensor and Hotwell (3)	269
Moisture Separator/Reheater (4)	61
Heaters (3)	
Third Stage	4.3
Fourth Stage	2.7
Fifth Stage	8.0
Sixth Stage	17.4
Feedwater Turbine (5)	0.1
Steam Seal Evaporator	0.7
Radwaste Reboiler	2.1
Off-Gas	
SJAE Condenser	0.44
Off-Gas Preheater	0.02
Recombiner	0.02
Cooler Condenser	0.19

(1) Turbine + 2 Supply line + 2 exhaust lines
(2) 3 turbine + 6 valves + supply line
(3) First and second stage heaters are part of condensor
(4) Includes first and second stage reheat
(5) Includes 10 feet of supply line

The following flow conditions were assumed:

a. Slug flow in all pipes,

b. Complete mixing in all components upstream of the chosen components, and

c. Slug flow through the chosen component.

Complete mixing minimizes the source accumulation in the up-stream components, and slug flow maximizes the source accumulation in the chosen component when a conservative source term is obtained for the component of interest.

* A source factor of 1.05 has been applied to conservatively address a 5% power uprate to 3489 MWt.

TABLE 12.2-9

RHR HEAT EXCHANGER SHUTDOWN SOURCES*

(values given in curies)

ISOTOPE	3 HOURS	1 DAY	2 DAYS	4 DAYS	8 DAYS
Na-24	1.7-2	<1.0-2	<1.0-2	**	**
Mn-56	2.2-1	<1.0-2	**	**	**
Co-58	5.0-2	5.0-2	4.9-2	4.8-2	4.6-2
Co-60	**	**	**	**	**
W-187	2.8-2	1.4-2	<1.0-2	<1.0-2	**
Br-83	6.3-2	<1.0-2	**	**	**
I-131	1.3-1	1.2-1	1.1-1	9.2-2	6.5-2
I-132	4.8-1	<1.0-2	**	**	**
I-133	8.1-1	4.0-1	1.8-1	3.6-2	<1.0-2
I-134	2.2-1	<1.0-2	**	**	**
I-135	9.5-1	1.1-1	<1.0-2	**	**
Sr-89	3.1-2	3.1-2	3.0-2	2.9-2	2.8-2
Sr-91	5.6-1	1.2-2	< 1.0-2	**	**
Sr-92	5.1-1	<1.0-2	**	**	**
Mo-99	2.1-1	1.7-1	1.3-1	8.1-2	3.0-2
Tc-99m	2.0	1.8-1	1.1-2	<1.0-2	**
Te-132	4.8-1	4.0-1	3.2-1	2.1-1	8.9-2
Cs-137	**	**	**	**	**
Ba-140	9.0-2	8.5-2	8.1-2	7.2-2	5.8-2
Np-239	2.3	1.8	1.3	7.4-1	2.3-1
TOTAL	~9.0	~3.5	~2.3	~1.4	~0.6

* The above values are based upon Table 12. 2-4. The mass of reactor water in the RHR heat exchanger was taken as 1.0 x 10^7 grams.

** These values are less than 1.0 x 10^{-3} curies.

TABLE 12.2-10
(SHEET 1 OF 2)
(Historical)

CALCULATED AIRBORNE IODINE ACTIVITY FOR DESIGN-BASIS LEAK RATE

IN RADWASTE BUILDING (a)

AREA	LIQUID IODINE CONCENTRATION (µCi/cc)	TOTAL NUMBER OF LEAK PATHS (b)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC FOR IODINES (c)
[redacted]	3.10 x 10^{-1}	5	9100	9.73 x 10^{-4}
	6.55 x 10^{-2}	7	9050	2.89 x 10^{-4}
	2.11 x 10^{-4}	7	8925	1.30 x 10^{-6}
	7.26 x 10^{-1}	10	14115	6.83 x 10^{-3}
	2.76 x 10^{-3}	2	1180	8.99 x 10^{-5}
	5.59 X 10^{-4}	5	2000	1.03 x 10^{-5}
	3.21 x 10^{-1}	2	1080	9.46 x 10^{-3}
	3.22 x 10^{-1}	3	800	1.93 x 10^{-2}
	4.57 x 10^{-4}	3	860	1.18 x 10^{-5}

TABLE 12.2-10
(SHEET 2 OF 2)
(Historical)

AREA	LIQUID IODINE CONCENTRATION (μCi/cc)	TOTAL NUMBER OF LEAK PATHS (b)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC FOR IODINES (c)
[redacted]	6.46×10^{0}	1	1000	1.03×10^{-1}
[redacted]	3.09×10^{0}	3	5080	1.52×10^{-2}
[redacted]	8.16×10^{0}	4	6250	8.34×10^{-2}
[redacted]	2.65×10^{0}	12	2790	1.66×10^{-1}
[redacted]	2.15×10^{-1}	6	800	1.83×10^{-2}

(a) The fraction of MPC values given in the table are based on the design base leakage of 0.02 gpm per leak path. Such large amounts of leakage are expected to be very rare, therefore, the actual fraction of MPC is expected to be a small fraction of the values given. The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 01/01/94.

(b) The number of leak paths is equal to the number of valves and motion components (i.e., pumps and turbines) which experience an internal pressure greater than atmospheric.

(c) The partition factor for a hot liquid (>120°F) is 1×10^{-1}.
The partition factor for a cold liquid (<120°F) is 1×10^{-4}.

TABLE 12.2-11

CALCULATED AIRBORNE IODINE ACTIVITY FOR DESIGN-BASIS LEAK RATE IN TURBINE BUILDING (a)

AREA	LIQUID IODINE CONCENTRATION (μCi/cc)	TOTAL NUMBER OF LEAK PATHS (b)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC FOR IODINES (c)
[redacted]	1.18×10^{-2}	1	17750	2.04×10^{-6}
	1.18×10^{-2}	2	2300	3.16×10^{-5}
	1.18×10^{-2}	4	300	4.83×10^{-4}
	1.18×10^{-2}	7	3000	8.49×10^{-2}
	1.18×10^{-2}	2	22700	3.20×10^{-3}
	1.18×10^{-2}	6	62500	3.48×10^{-3}
	1.18×10^{-2}	5	3600	5.04×10^{-2}

(a) The fraction of MPC values given in the table are based on the design base leakage of 0.02 gpm per leak path. Such large amounts of leakage are expected to be very rare; therefore, the actual fraction of MPC is expected to be a small fraction of the values given. The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 01/01/94.

(b) The number of leak paths is equal to the number of valves and motion components (i.e., pumps and turbines) which experience an internal pressure greater than atmospheric.

(c) The partition factor for a hot liquid (>120°F) is 1×10^{-1}. The partition factor for a cold liquid (<120°F) is 1×10^{-4}.

(d) Hot Liquid

TABLE 12.2-12
(SHEET 1 OF 3)

CALCULATED AIRBORNE IODINE ACTIVITY FOR DESIGN-BASIS LEAK RATE

IN REACTOR BUILDING (a)

AREA	LIQUID IODINE CONCENTRATION (μCi/cc)	TOTAL NUMBER OF LEAK PATHS (b)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC FOR IODINES (c)
[redacted]	4.65×10^{-4}	2	40,000	1.72×10^{-7}
[redacted]	5.92×10^{-1}	7	1100	1.16×10^{-2}
[redacted]	5.92×10^{-1}	3	300	1.81×10^{1}
[redacted]	Airborne during shutdown only. No measurable quantities of iodine during normal operation.			
[redacted]	5.92×10^{-1}	10	400	4.53×10^{1}
[redacted]	5.92×10^{-1}	6	1000	1.09×10^{1}
[redacted]	5.44×10^{0}	8	1200	2.7×10^{-1}

TABLE 12.2-12
(SHEET 2 OF 3)

AREA	LIQUID IODINE CONCENTRATION (μCi/cc)	TOTAL NUMBER OF LEAK PATHS (b)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC FOR IODINES (c)
[redacted]	8.35×10^{0}	4	3000	8.74×10^{-2}
	1.48×10^{-3}	5	16910	1.27×10^{-5}
	8.35×10^{0}	7	1890	2.43×10^{-1}
	7.45×10^{-4}	6	400	3.22×10^{-4}
	3.2×10^{-2}	1	1300	6.99×10^{-5}
	Outside air during normal operation. No measurable quantities of iodine			
	1.18×10^{-2}	2	3000	2.24×10^{-2}
	5.92×10^{-1}	2	500	7.25×10^{0}

TABLE 12.2-12
(SHEET 3 OF 3)

AREA	LIQUID IODINE CONCENTRATION (µCi/cc)	TOTAL NUMBER OF LEAK PATHS (b)	EXHAUST AIR FLOW RATE (cfm)	FRACTION OF MPC FOR IODINES (c)
[redacted]	1.48×10^{-3}	1	25000	2.03×10^{0}

(a) The fraction of MPC values given in the table are based on the design base leakage of 0.02 gpm per leak path. Such large amounts of leakage are expected to be very rare; therefore, the actual fraction of MPC is expected to be a small fraction of the values given. The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 01/01/94.

(b) The number of leak paths is equal to the number of valves and motion components (i.e.. pumps and turbines) which experience an internal pressure greater than atmospheric.

(c) The partition factor for a hot liquid (>120°F) is 1×10^{-1}. The partition factor for a cold liquid (<120°F) is 1×10^{-4}.

(d) Hot Liquid

(e) The fraction of MPC values are based on the total iodine release rate of 2.23×10^{-1} µCi/sec from the pool.

TABLE 12.2-13

MEASURED AIRBORNE ACTIVITY DURING DRESDEN REFUELING*

DAY OF OUTAGE	WORK FUNCTION	20-MINUTE COUNT** (μCi/cc)	6 HOUR COUNT** (μCi/cc)	IODINE (μCi/cc)
3	Remove Vessel Head	2.3×10^{-11}	7.0×10^{-12}	Not Available
8	Remove Fuel	2.8×10^{-11}	1.6×10^{-12}	2.5×10^{-11}
12	Sipping	3.1×10^{-11}	5.4×10^{-12}	1.5×10^{-10}
29	Reload Fuel	2.0×10^{-11}	5.4×10^{-12}	$1.1 \text{ X } 10^{-11}$
39	Replace Dryer and Separator	7.0×10^{-11}	5.4×10^{-12}	3×10^{-12}
Power Operation		$3\ 5 \times 10^{-11}$	$8\ 8 \times 10^{-12}$	5×10^{-12}

* The data is the beta activity due primarily to particulates and gamma activity due primarily to iodine on the refueling floor. Noble gas activity was not recorded because previous experience has shown that the noble gas activity was small. The dominant airborne activity was due to radioactive cobalt and Iodine - 131.

** Continuous air monitor (CAM) filter paper readings.

12.3 Radiation Protection Design Features

Radiation protection design features are provided to reduce direct radiation to personnel and equipment, calibrate radiation monitors, and maintain personnel radiation exposure as low as reasonably achievable (ALARA).

12.3.1 Facility Design Features

The location of offices, laboratories, and maintenance areas determine the traffic patterns. The traffic patterns and the location of sources determine the radiation design dose rates, which in turn determine the shielding and ventilation requirements to maintain personnel radiation exposure ALARA. The traffic patterns can also be controlled by establishing access control points.

Additional aids to radiation protection include, (1) classification of all areas of the station into radiation zones, (2) adequate laboratory facilities, and (3) decontamination capabilities.

12.3.1.1 Station Untilization

The servicing requirements of equipment is influenced by its location which affects the station's traffic and equipment removal patterns.

12.3.1.1.1 Traffic Patterns

The majority of the station traffic occurs between the service building and the auxiliary building. The remainder of the traffic occurs in operating areas (where panels and motor control centers are located), hallways, elevators, and stairwells. Radwaste operations personnel mainly travel between the radwaste building, the basement, and upper basement of the turbine building (west side), and the service building.

12.3.1.1.2 Duct and Pipe Routing

In order to obtain ALARA radiation levels in high occupancy areas, most piping that handles radioactive fluids are routed through radioactive areas or through shielded pipe tunnels. The only exceptions occur in the drain systems. Some drains having very low activities (a few μCi per liter) cannot be totally contained in radiation areas or in the floor slab. Such drain lines are designed to minimize crud buildup, and they are located at high elevations so that ALARA radiation levels can be maintained.

Piping from several systems are occasionally routed through the same radiation area. Whenever practicable, these lines are designed and located to minimize crud buildup and occupation exposure in the radiation area. The most common method for minimizing crud buildup in piping, is flushing after using and sloping the lines to facilitate draining them.

The ventilation ducting is designed to carry station air from the low radiation areas to the high radiation areas. The air from the high radiation areas is exhausted through shielded pipe tunnels and particulate filters to the station vent stack. Thus, the majority of the ducting which passes through low radiation areas do not handle significant quantities of contaminants. Duct and pipe routing does not interfere with the proposed station utilization.

12.3.1.1.3 Equipment Removal

LSCS's station design has established equipment removal paths to handle removal and replacement. Mortared and unmortared removable block wall sections and ceiling plugs are properly sized. Strategically placed removal aids (e.g., lifting lugs, crane rails, etc.) reduce removal time, resulting in reduced occupational exposure.

12.3.1.1.4 Pull Spaces

Equipment removal paths usually utilize pull spaces when they are available. The pull space can be used as work areas if the need arises. Pull spaces are provided for most equipment which uses one fluid stream to heat or cool another fluid stream e.g., evaporators, heaters, and condensers.

When maintenance is performed in pull spaces, potential contamination is handled in accordance with the contamination control program (Subsection 12.5.3.4).

12.3.1.2 Access Control

The objectives of radiological access control are outlined in Subsection 12.5.3. Provisions of design and equipment are intended to facilitate the goals stated therein.

The principal access control point is located in the Unit 1 turbine building ground floor at column line 4. An additional access control point can be established for Unit 2, located on the ground floor of the north service building. Associated with these control points are the radiation protection offices. The additional access control point for Unit 2, if it is required in the future, can be established using procedure controls. Prior to entering the in-plant areas through these points, a person can receive instructions from health physics personnel, special radiation work permits, and special dosimetry devices, as needed. All persons leaving the inplant areas normally pass through the turbine building checkpoint where they are monitored for contamination, and decontaminated as needed, prior to entering the general service building areas. Protective clothing and pesonnel protective respiratory equipment are located in the turbine building on the ground floor near the entrance to the auxillary building.

Access control to and from the reactor building is provided on the ground level where the reactor building and auxiliary building meet. This control is provided by two airlocks. The airlocks consist of two doors in series, interlocked so that only one door can be opened at a time.

A contamination control boundary is provided around the auxiliary building complex. A radiation detection monitor and a personnel decontamination room are provided in this area.

12.3.1.3 Radiation Zones

12.3.1.3.1 Dose Rates Within the Station Areas

The Radiologically Posted Area (RPA) is defined as an area within the restricted area to which access is limited for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials, and which is posted with a radiological sign. At LSCS, these areas are identified, set up, and controlled by the Radiation Protection Department.

Five main radiation zones have been defined prior to operation as a means of classifying the occupancy restrictions on various areas within the plant boundary. The design criteria for each zone are described in the subsections which follow and are tabulated in Table 12.3-1. The radiation zones assigned to the areas of the plant, and upon which the shielding has been designed, are shown in the radiation zone maps in Figures 12.3-1 and 12.3-2. Actual radiological conditions are currently used to control access and post areas within the plant.

12.3.1.3.2 Zone I

Zone I applies to areas that have a maximum design dose rate of 1 mrem/hr. Habitation of such an area on a 40-hour-per-week, 50-week-per-year basis results in a whole-body deep dose equivalent of less than 5.0 rems per calendar year, the limit specified in 10 CFR 20.1201. Occupancy of such an area is therefore generally unlimited. Corridors in the turbine building or auxiliary building and areas outside radioactive cubicles are examples of areas designated Zone I. Auxiliary equipment (such as valves, dampers, and controls) located in these areas can be maintained without draining process equipment.

12.3.1.3.3 Zone II

Zone II is used to indicate the lowest level radiation areas within the plant to which access may be controlled. Occupancy on a 40-hour-per-week, 50-week-per-year basis results in a whole-body deep dose equivalent which does not exceed the 5.0 rems per calendar year limit for individuals in a restricted area specified in 10 CFR 20.1201.

12.3.1.3.4 Zone III

Areas which are designed as Zone III may be "radiation areas," as defined in 10 CFR 20.1003. The design dose rate in Zone III areas does not exceed 15-mrem/hr. All such areas may be posted with signs bearing the radiation caution symbol and the words "CAUTION RADIATION AREA," as prescribed by 10 CFR 20.1902.

12.3.1.3.5 Zone IV

Zone IV is used to designate radiation areas where the maximum design dose rate does not exceed 100-mrem/hr. These areas may be "radiation areas," as defined in 10 CFR 20.1003, and may be posted with "CAUTION - RADIATION AREA" signs as required by 10 CFR 20.1902.

12.3.1.3.6 Zone V

Areas which are designated as Zone V may be "high radiation areas, "as defined in 10 CFR 20.1003. The design dose rate in Zone V areas may exceed 100-mrem/hr. All such areas may be posted with "DANGER - HIGH RADIATION AREA," as prescribed in 10 CFR 20.1902.

Access points to Zone V areas may be secured by a locked door, barricade, or other suitable means and maintained in accordance with 10 CFR 20 and the Technical Specifications. An area survey of radiation levels is normally conducted prior to occupancy of Zone V areas in order to determine the maximum habitation time.

12.3.1.4 Radiation Shielding

Permanent shielding is discussed in Subsection 12.3.2.

Temporary shielding, such as concrete blocks, lead bricks, lead sheets, and lead wool blankets, is used where necessary to reduce external personnel radiation exposures during operational and maintenance activities in radiation and high radiation areas. Each activity in these areas requires a specific evaluation of dose rates, job complexity (number of people and total time), available space, and the time required to install and remove temporary shielding. Based on these evaluations, the Radiation Protection Department will determine the use of temporary shielding on a case-by-case basis.

12.3.1.5 Laboratory Complex

The station laboratory complex is located on the ground floor of the auxiliary building. The facilities provided within this controlled access area are: hot laboratory, chemistry office, supply room, radiochemist office, cold laboratory, calibration room, counting room, sample preparation laboratory, a personnel decontamination room, a first aid room, a men's toilet, and a women's toilet.

12.3.1.5.1 Hot Laboratory

The hot laboratory is designed to safely analyze radioactive samples as needed for the operation of the plant. The major facilities provided in the laboratory are: fume hoods with HEPA filtered exhaust, sinks (drains routed to radwaste), and sufficient workbench space to allow frequently used equipment to be left in place. To minimize the spread of contamination, floor coverings, surface coatings, and sink and bench materials are chosen to ease decontamination. To minimize the spread of airborne contamination, the room is maintained at a negative pressure with respect to all surrounding areas, and the exhaust from the room is filtered prior to its release to the environment via the station vent stack.

A space is provided for an atomic absorption spectrophotometer.

12.3.1.5.2 Sample Preparation Laboratory

The sample preparation laboratory is located near the hot laboratory. It is intended to be used for standards preparation, special research, or ongoing experiments. Its equipment includes fume hoods, sinks, and workbench space. The contamination control aspects provided in this laboratory are the same as those for the hot laboratory.

12.3.1.5.3 Cold Laboratory

The cold laboratory is located adjacent to the sample preparation laboratory. It is normally used for the analysis of nonradioactive samples, e.g., sewage and makeup water. The equipment provided in this laboratory includes a sink, a fume hood, and workbench space.

12.3.1.5.4 Counting Room

The counting room is located near the radiochemistry laboratories in the auxiliary building. To minimize the background caused by radon daughter products emitted from concrete, the walls of this room are constructed of gypsum drywall. To minimize any influence of in-plant airborne activity, the room is maintained at a positive pressure with respect to surrounding areas by means of fresh filtered and conditioned air. The room is provided with a reliable power source and nonfluorescent lighting to assure optimum performance of the counting experiment.

The original equipment to be provided in the counting room includes:

a. two gamma spectrometer systems, and

b. two gas-flow proportional counter systems.

Localized shielding is provided for this equipment to the extent necessary.

12.3.1.5.5 Chemistry Storage

A chemistry storage room is provided on the ground floor of the auxiliary building between the hot laboratory and the sample preparation laboratory.

12.3.1.5.6 Instrument Calibration Room

The instrument calibration room is located on the Unit 2 side of the ground floor (elevation 710 feet 6 inches) of the auxiliary building. The calibration facility has been positioned in the southeast corner of the room, adjacent to the reactor building and the hot laboratory.

Portable radiation survey instruments are normally calibrated or source checked in this room using a Cs-137 source or other calibration sources stored in the room. Some of the portable sources stored here are used to calibrate the area radiation monitors and the process and effluent radiation monitors (Subsection 12.3.4). Air samplers may be flow calibrated in this room using the equipment normally stored here.

The calibration facility normally utilizes Cs-137 sources of sufficient curie strength to provide required radiation fields for calibration.

The source elevator travels along a vertical shaft (which is approximately 45 feet long) and has a traveling mechanism which permits the source to be positioned at desired distances from an instrument detector to obtain the desired radiation level at the detector. Additional radiation shields extend the range of this facility.

Warning lights are located on the operator's panel and outside of the entrance into the room, and they indicate when the source elevator is stored, moving, or positioned for high level calibration. The source is considered stored when the source elevator is within a few feet of its lowest elevation. An audible alarm sounds when the source is near the top of the elevator, and interlocks prevent shutdown of the equipment unless the source is in the stored position.

Various jigs are used to properly position the portable instrument detectors at the source tube opening.

A Condenser R-meter (or equivalent) is to be used to check and calibrate the calibration facility. This instrument, which is periodically calibrated to NIST standards, can also be used to make accurate radiation measurements as needed around the plant.

An electronic pulse generator is also available to electronically verify pulse counting instruments operability.

12.3.1.6 Decontamination

Equipment decontamination prior to lengthy maintenance and personnel decontamination (when required) are necessary to maintain radiation exposure ALARA. In addition to flushing capabilities on most large (potentially radioactive) components and primary system pumps, the facilities described in the following subsections are available for more extensive decontamination of equipment, clothing, and personnel.

12.3.1.6.1 Equipment Decontamination Facilities

Decontamination pits are designed to aid in the safe removal of radioactive material from station equipment and direct this material to radwaste. LSCS has six pits; one in the solid radwaste building, one in the service building (machine shop), two in the reactor building (cask washdown and reactor pressure vessel), and two in the turbine building (main floor and ground floor). These pits are constructed of concrete lined stainless steel. All drain piping is made of stainless steel.

Almost all equipment decontamination is performed in radiation areas or in the machine shop decontamination room where approved procedures control exposure to maintenance personnel. Work at the other decontamination areas is supervised and monitored in accordance with the health physics program (Section 12.5).

12.3.1.6.2 Personnel Decontamination Facilities

A personnel decontamination facility is supplied to provide for prompt decontamination of plant personnel, if the need should arise.

The personnel decontamination room is provided at the ground floor of the auxiliary building. This facility is available to persons leaving the containment or reactor building areas as well as those leaving the laboratories complex. This room is provided with space for contaminated clothes hampers and has a wash sink and a shower with an associated drying area.

Sinks, eyewashes, and emergency showers have been placed in strategic locations throughout the station. These facilities are primarily intended for emergency chemical removal, but can be used for emergency personnel decontamination. All areas where decontamination is expected to take place have been designed to ease cleanup and include drains which flow to radwaste.

12.3.1.6.3 Mask Wash and Issue Facility

The mask wash and issue facility is located in the Radiologically Controlled Area.

The mask wash and issue facility is designed to receive, dismantle, wash, dry, disinfect, and reassemble respiratory protective devices. Design provides for a point where masks may be dropped off after use, a wash and dry area, and distribution shelves.

12.3.1.6.4 Facility Decontamination

Decontamination of floors and walls, as well as equipment, may be required after servicing and maintaining radioactive equipment. For this reason, the floor and part of the walls of radiation areas are coated with a vitreous coating. The walls surrounding decontamination pits are coated to a height of 8 feet. The rooms which contain equipment which handle radwaste sludge and spent resins, including the radwaste pipe tunnels, are completely coated. Valve aisles, pump aisles, and cubicle entrances were analyzed for wall contamination, and the high maintenance radioactive areas require a minimum 5 foot high wall coating.

The floors of passageways, corridors and general access areas have vitreous coatings to facilitate decontamination due to spillage.

12.3.1.6.5 Decontamination of Primary Coolant Systems

Experience at operating stations has shown an increase in dose rates for passive (drained) primary coolant equipment. The pieces of equipment which show the largest increases are in the nuclear boiler system (NB), the recirculation system (RR), the reactor water cleanup system (RWCU) and the residual heat removal system (RHR). These dose rates could be responsible for the majority of the occupational radiation exposure. If such a state arises at LSCS, the major decontamination of components (and possibly systems) would become advantageous.

Major decontamination is discussed in Subsection 12.4.1.3.

12.3.1.7 Remote Handling Equipment

All but two functions of solid radwaste drumming procedure is performed remotely. One exception is the health physics function of surveying and labeling the drums. The dry waste drumming is the other. These two functions are performed in accordance with the radiation exposure requirement of the health physics program (Section 12.5).

Valves are remotely operated when practicable. Motor-operated valves are controlled from panels, and mechanically operated valves are controlled from valve operating aisles.

Reactor coolant, off-gas, and liquid radwaste samples are obtained from sampling panels, and remote instrumentation readouts, e.g., pressure, temperature, and flow, are located in panels whenever it is practicable. (Note: All the panels mentioned previously are located in general access areas).

12.3.1.8 Equipment Location

The location of equipment within LSCS is based upon ALARA exposure and upon experience from Dresden, Quad-Cities, and Zion Stations. Equipment is located after considering the equipment's function, its relationship to other pieces of equipment, its potential of becoming radioactive, how radioactive pipe and equipment will affect servicing, the difficulty of maintenance, the estimated frequency of possible manual operation, the estimated frequency of maintenance, its accessibility for servicing, the estimated frequency of its removal, the amount and types of instrumentation required, and remote readout and control requirements.

Illustrative examples of equipment layout are given in the following subsections, stressing ALARA occupational dose.

12.3.1.8.1 Heat Exchangers

Heat exchangers are segregated from pumps and are segregated from valves whenever practicable. Some heat exchangers use removable wall plugs instead of the usual removable block wall sections because they have the potential of accumulating large amounts of radioactive crud, thus, they have a high shutdown dose rate. Overhead crane rails are incorporated into the station design to allow remote positioning or removal of heat exchangers. These features have been installed for the purpose of reducing occupational radiation exposure.

The heat exchanger cubicles have ventilation, curbs and floor drains which are designed to minimize the spread of contaminants.

Figure 12.3-3, sheet 1, shows the floor plan of the reactor water cleanup heat exchangers.

12.3.1.8.2 Liquid Filters and Demineralizers

LSCS uses three types of liquid decontamination equipment, e.g., filters, mixed bed demineralizers, and filter-demineralizers.

a. The filters and filter-demineralizers, hereafter referred to as filters, require periodic replacement of the filtering medium by mechanical means. Routinely, the precoat material is backwashed off the elements. Periodically, the filter elements need to be removed from the tubesheet and replaced. For precoatable filter elements, the elements are un-threaded from the tubesheet and processed as solid radwaste. For non-precoatable filter elements, the filter cartridge is removed and processed as solid radwaste. State-of-the-art technology is employed in the design of the filter elements to include ALARA aspects to speed filter changeouts. In order to minimize occupational radiation exposure, each unit has its own cubicle which only has access through the ceiling, or has shielding attached to the filter housing. Special equipment is used to remove and package the old filter. The filters also have flushing and draining capabilities, and all valves, pump, and instrumentation are segregated from them. Figure 12.3-3, sheet 2, shows the cubicle arrangement and the filter removal area for the fuel pool filter-demineralizers and the waste filters.

b. Mixed bed demineralizers do not require manual removal of the filtering material. This material is installed and removed by flowing liquids; therefore, the servicing of demineralizers is very infrequent as long as pump, valves, and instrumentation are segregated from them. Because the servicing of demineralizer cubicles is minimal, two or more demineralizers can be put into one cubicle when access is strictly controlled. Access to demineralizers will be in accordance with the health physics program discussed in Section 12.5. Curbs and floor drains aid in controlling contaminants.

Figure 12.3-3, sheet 3, shows the waste mixed bed demineralizers cubicle.

12.3.1.8.3 Tanks

The servicing requirements for tanks are similar to the ones for mixed bed demineralizers (see Subsection 12.3.1.8.2 item b).

Some tank cubicles (all of them in liquid radwaste) are designed to handle large overflows and spills, and by interconnecting tank cubicles, the entire volume of two highly radioactive liquid radwaste tanks can spill onto the floor without contaminating general access areas. All other tank cubicles are provided with curbs and floor drains. Figure 12.3-3, sheet 4, shows the chemical waste tank rooms.

12.3.1.8.4 Evaporators

The radiation protection design for evaporators is the same as those for heat exchangers except that only removable block wall sections are utilized in lieu of plugs (see Subsection 12.3.1.8.1).

Figure 12.3-3, sheet 5 shows the layout of the steam seal evaporator.

12.3.1.8.5 Recombiners

The recombiner system (e.g. preheater, recombiner, cooler, and valves) is located in its own cubicle which contains a sizable open area for performing maintenance. Temporary shields can be employed to maintain ALARA exposure during maintenance and servicing if it is warranted.

The recombiner room also includes all of the radiation protection design features used for evaporators (Subsection 12.3.1.8.4). Remote manual valve operation is performed in the adjacent valve operating aisle.

Figure 12.3-3, sheet 6, shows the layout of the recombiner system.

12.3.1.8.6 Pumps and Valves

The location of pumps and valves is based upon system requirements, servicing requirements, and radiation protection.

12.3.1.8.6.1 Pumps

LSCS pumps are in two categories, periodic operation and continual operation. Pumps are segregated from the components they service, and valves are segregated from pumps wherever practicable.

a. Continually operating pumps are usually serviced and maintained during shutdowns. Occasionally one of these pumps is serviced during normal operation, but only after it has been isolated from the remainder of the system and flushed. The pump can be removed to an area of lower radiation or temporary shielding can be used to reduce occupational radiation exposure in most cases.

   LSCS has three turbine driven pumps per unit. Two of these pumps are continual operating feed water pumps that require periodic in-service inspection. A steel shield has been located between the pumps and turbines so that the inspection function can be performed in a reasonable radiation field. Figure 12.3-3, sheet 7, shows a portion of a turbine driven feed water pump cubicle. The third turbine-driven pump is the RCIC pump which is operated infrequently.

b. Radwaste pumps are used on a periodic and/or infrequent basis. All of these pumps are separated from the tanks that they service. Several of the pump areas are in isolated open areas for the purpose of reducing servicing and maintenance time. The remainder of the radwaste pumps are in shielded pumprooms and aisles. All radwaste pumps are flushed after they have been used.

Figure 12.3-3, sheet 5, shows the chemical waste pumps.

Figure 12.3-3, sheet 8, shows the waste sample pump area.

12.3.1.8.6.2 Valves

Valves are located in valve rooms and valve aisles whenever practicable. Manual and automatic valve operators are used in valves which are located near highly radioactive components. The manual operators are located in valve operating aisles.

Figure 12.3-3, sheets 1, 3, and 5, show valve aisles and rooms.

Figure 12.3-3, sheet 6, shows a valve operator aisle.

12.3.1.8.7 Ventilation Filters

The functions of the ventilation equipment are discussed in Subsection 12.3.3. The handling of ventilation filters (during removal and while being transported to the radwaste building) will be in accordance with the procedure established in the health physics program.

Figure 12.3-3, sheet 9, shows the ventilation exhaust filtering unit for the radwaste building.

12.3.1.8.8 Sampling Stations

Centralized sampling stations are used on highly radioactive fluid streams, i.e., reactor water cleanup system, off-gas system and liquid radwaste. These stations are located on panels which are in general access areas. For systems which only require one or two sampling stations, the station is located in as low a radiation area as is practicable. The sample stations for the solid radwaste waste concentrator tank is located in a general access area (see Figure 12.3-3, sheet 10).

Figure 12.3-3, sheet 11, shows the reactor building sample panel.

12.3.2 Shielding

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

12.3.2.1 General Shielding Design Criteria

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

12.3.2.1.1 Regulatory Requirements

The shielding design for LSCS is in compliance with 10 CFR 20 and 10 CFR 50, which are concerned with allowable radiation to individuals in restricted and unrestricted areas. The only shielding required to be safety related is the control room and the primary containment shielding; this shielding satisfies the requirements stated in 10 CFR 50.67, and 10 CFR 20.

12.3.2.1.2 Station Operation

LSCS has a cycle of operation modes which include startup, power operation, shutdown, refueling, and hot standby. The station shielding is designed to the mode of operation which produces the most penetrating radiation condition for a given component. In addition to the operation modes, accidents are also considered in the shielding design. The operating conditions have been grouped into the following three classifications:

a. Power Operation - Operation at design power, partial power, hot standby, and startup, including anticipated operation occurrences.

b. Shutdown Operation - includes: radiation from the subcritical reactor core, reactor cooldown; refueling, and fission product and activation product accumulation throughout the plant.

c. Postaccident Operation - includes: accidents analyzed in Chapter 15.0, with special emphasis on control room habitability.

The shielding is designed to one of the above classifications or to testing operations. The operation classification which affects the station shielding is listed in the following on a system-by-system basis.

System	Shielding Design Basis
Fuel Pool Cooling and Cleanup	Shutdown (1 day)
Residual Heat Removal	Shutdown (3 hours)
Reactor Core Isolation Cooling	Test
Main Steam (Blowdown)	Postaccident
Control Room (DBA)	Postaccident
All Others Power	

12.3.2.1.3 Source Parameters for Shielding

The shielding design is based on the quantity and the type of radiation as well as the design dose rate and the shielding material(s). The following list gives the type of radiation used to determine the station shielding along with the additional parameters:

a. The shielding in the turbine building is based on the gamma rays of Nitrogen-16;

MeV/Photon	Photon/Disintegration
2.75	0.01
6.143	0.69
7.112	0.05

b. The shielding for components of the radwaste system, other than the Interim Radwaste Storage Facility (IRSF), is based on the sources provided in Section 11.2.

The IRSF is of a standard design used at a number of EGC nuclear power plants. The IRSF isotopic breakdown assumptions are:

Cs-137	40%
Co-60	30%
Co-58	10%
Cs-134	18%
Mn-54	2%

The isotopic breakdown is conservative relative to that discussed in Section 11.2.

c. The shielding for components of the off-gas system is based on the sources provided in Section 11.3.

d. The shielding for components which handle reactor water is based on 2.26 μCi/g of fission-products and 0.06 μCi/g of corrosion-products (Section 12.2).

e. The dose rates outside cubicles containing multiple sources are determined by the sum of the direct radiation received from all sources through the proposed shielding.

f. Crud buildup is considered, and the shielding is increased by 10% (up to 4 inches) for sources that do not contain high energy gamma radiation (>3 MeV). Crud buildup does not affect shield when sufficient quantities of high energy gammas are present.

g. Credit is normally taken for self-adsorption within a given source geometry.

h. Credit is taken for holdup of Nitrogen-16 in various system components, (i.e., main steam, extraction steam and reactor water cleanup).

12.3.2.1.4 Design Dose Rates

Design dose rates have been chosen on the basis that the entire station design effects the occupational radiation exposure. Because the dose rates within the station span from zero to 10's of rad/hr, the span has been divided into five groups (e.g., <1, <2.5, <15, <100, >100 mr/hr). The highest level of occupancy occurs in the areas with the lowest design dose rate. As the dose rates increase the level of occupancy decreases.

The design dose rate considers all modes of operation and crud buildup. Table 12.3-1 classifies design dose rates for different types of areas and gives the basis for the limits. Design dose rates and administrative controls are selected such that a properly trained staff can maintain radiation exposure ALARA.

12.3.2.1.5 Shielding Materials

The primary shielding material used to meet shielding design criteria is ordinary concrete with a minimum density of 2.2 grams per cubic centimeter. The majority of the concrete shields are poured in place, and the remainder are made up of mortared solid block. Lead, steel, and water are used in certain applications, e.g., reactor shield space limitations, wall plugs, and temporary shielding. Where access must be provided for periodic inspection and maintenance, removable concrete, lead, or steel shields (e.g., plugs, block walls) are used.

Compensatory shielding materials include concrete, steel, water, and lead, and they are incorporated into the station shielding design when the operating staff determines that compensatory shielding is required. The choice of material is selected on the basis of space and structural limitations. Table 12.3-2 gives the shielding properties of the shielding material chosen for LSCS.

12.3.2.1.6 Calculational Methods

To design the reactor shield and primary containment shield, the one-dimensional transport code ANISN (Reference 1) was used to calculate the transport of neutrons and gammas from the core. It also analyzed the subsequent production of capture-gamma rays in regions external to the core. The CASK code (Reference 2) coupled neutron-gamma ray library of cross sections was utilized with the ANISN code to enable all production and loss mechanisms for both neutrons and gamma rays to be handled in a single calculation.

All other shields are designed for only gamma-ray attenuation by the standard point attenuation kernel (buildup factor, exponential attenuation, and geometry factor), numerically integrated over the volume of the source. The buildup factors and gamma-ray attenuation coefficients were obtained from published data (References 3 and 4). ISOSHLD-III (Reference 5) and QAD (Reference 6) are two point-kernel computer codes used in this design effort.

Scattered radiation from labyrinths and penetrations is analyzed by the point-kernel single-scatter computer code GGG (Reference 7) or by the Monte Carlo code OGRE (Reference 8). Penetrations are located such that direct radiation from the source to the dose point is minimized, and the major contribution to the dose rate is from scattered radiation. If possible, wall penetrations are located above head height, and the use of wall and floor penetrations which run between radioactive areas and unlimited-access areas is minimal.

Skyshine radiation from waste stored in the IRSF to onsite and offsite areas is analyzed by the computer code Skyshine II (reference 22) with a check using the computer code GGG.

12.3.2.1.7 Shielding Drawings

The dimensions given in the shielding drawing, Drawing No. M-25, indicate the thickness required to maintain the design dose rate under design-basis operating conditions (which include shutdown, startup, refueling, etc.). The thicknesses were determined by the calculational methods mentioned previously and the source terms given in Section 12.2. Subsection 12.3.2.3 tabulates the shielding design pictured in Drawing No. M-25.

12.3.2.2 Shielding Design Criteria

The shielding design maintains the design dose rates established in Subsection 12.3.1 as much as practicable. The shielding design criteria establishes the requirements and guidelines for shielding, cubical access, radiation streaming, and line routing. For the few cases where the shielding criteria cannot be met, temporary and permanent fencing (or barriers) are used to segregate the area and control occupational radiation exposure. Sign posting and access are in accordance with 10 CFR 20.

12.3.2.2.1 Permanent Shielding

Shielding which affords protection during several operational modes, (e.g., normal operation and postaccident situations) is designed on the basis of the most limiting criterion. Each source of radiation and its surrounding physical environment is modeled mathematically to represent the actual physical arrangement as closely as possible and is subject to the following criteria:

a. General access areas (e.g., aisles, corridors, panel areas, and offices) are shielded from radiation sources using materials of sufficient thickness and height to maintain the design dose rate levels desired.

b. Entrances to shielded cubicles are designed to prevent source radiation from passing directly through an opening (shield door or labyrinth).

c. When possible, a labyrinth entrance is designed to reduce the emerging radiation to the desired level; otherwise, a shield door is used.

d. Steel, lead, or concrete removable plugs are used for equipment replacement and maintenance whenever normal access is determined to be insufficient.

e. Plugs or doors are utilized when access to equipment is expected at a frequency greater than once a year.

f. Mortared block wall sections are incorporated into the cubical design when little or no equipment removal is expected, (i.e., once during the life of a plant).

g. Plugs and doors are considered in lieu of removable mortared block wall sections when the radiation dose rate inside the source cubicle exceeds 3 rads per hour.

h. Unmortared block with special metal framing is used when wall removal is expected once a year or less.

i. The labyrinth roof is designed to control the direct radiation that might affect general access areas, (i.e., galleries, catwalks, and corridors).

12.3.2.2.2 Radiation Streaming

Pipe and duct routing and structural requirements can cause localized "hot spots" (small areas adjacent to shielding walls which have higher radiation levels than the design dose rate of the areas surrounding them). The amount of area affected by "hot spots" is minimized by location and additional shielding. The following list of criteria include location of penetrations through shielding, design streaming ratios, and shield design for imbedded and surface voids:

a. A streaming ratio of up to 5 times the design dose rate is allowed in general access areas where administrative controls are practical.

b. A streaming ratio of up to 5 times the design dose rate is allowed in controlled access areas where the hot spot is zero to 8 feet above the floor.

c. A streaming ratio of up to 10 times the design dose rate is allowed in controlled access areas where the hot spot is 8 to 14 feet above the floor.

d. A streaming ratio of up to 15 times the design dose rate is allowed in controlled access areas where the hot spot is more than 14 feet above the floor.

e. Additional shielding or roping off the areas (with chain or permanent fences) is specified when the above criteria cannot be met.

f. Penetrations are designed so that radiation sources are not in direct view from general access areas.

g. Penetrations are located and orientated to minimize radiation dose levels to general access areas wherever practicable.

h. Imbedded potentially radioactive piping is properly shielded to protect general access areas.

i. Imbedded nonradioactive piping is designed so that the average density of the pipe and its contents are equivalent to the average density of the shielding material which satisfy the streaming ratios mentioned previously. This criterion also applies to structural supports, i.e., diagonal pipe bracing.

j. When the shielding is based on large fluxes of high energy gammas, surface voids (i.e., roof decking) do not effect the design dose level.

k. The surface void streaming ratio due to a large flux of low energy gammas does not exceed 2 times the design dose rate for controlled and uncontrolled access areas.

l. Imbedded voids (i.e., instrument chambers) are designed to the hot spot streaming ratios.

12.3.2.3 <u>Design</u>

LSCS's shielding design is pictured in Drawing No. M-25.

The shielding for the reactor building, turbine building, radwaste building, and remainder of the station is presented in Tables 12.3-3, 12.3-4, 12.3-5 and 12.3-6, respectively. The design data presented in these tables include design sources, shielding material used, required shielding thickness, calculated and estimated maximum dose rates inside the source cubicles, and the design dose rates outside the source cubicles.

12.3.2.4 <u>Evaluation of Shielding</u>

The radiation zone maps (Figures 12.3-1 and 12.3-2) give the radiation dose rates inside and outside source cubicles. The dose rates outside the cubicles are maintained by the shielding design. These dose rates together with administrative controls achieve radiation exposures to station personnel that are ALARA.

The site boundary is protected from direct radiation (scattered and unscattered) by shielding. The shielding surrounds all sources completely, except for the high and low pressure turbines which do not have a ceiling. The skyshine dose at the site boundary was reduced to ALARA levels by making the walls higher (reducing the solid angle).

Unmortared block walls and plugs are used to enhance maintenance and minimize radiation exposure associated with wall section removal.

Labyrinth entrances are used to control radiation streaming. They are designed to reduce unscattered radiation to desirable levels. The resulting scattered radiation flux inside the labyrinth is greater for low energy gamma than for high energy gammas having the same flux. The scattered radiation streaming through the labyrinth entrance way satisfies the design criteria for hot spots.

12.3.2.5 <u>Design Review of Plant Shielding For Post-Accident Operation</u>

The dose information in this section 12.3.2.5 and in Table 12.3-7 and Table 12.3-8 is based on the pre-power uprate power level (3323 MWt) and before consideration of Alternative Source Terms; however, the conclusions of the analyses will not change due to the higher power level (3489 MWt) and the application of Alternative Source Terms. During post-accident conditions the vital areas (Control Room, Auxiliary Electric Equipment Room where the remote shutdown panels are located, and the Technical Support Center) will be accessible for extended or continuous occupancy and access to the areas will be possible from a radiation dose standpoint.

The values in Table 12.3-7 and Table 12.3-8 increase by less than 25% due to a power uprate to 3908 MWt.

A radiation and shielding design review was made for LaSalle County Station using the source terms specified in Regulatory Guides 1.183 and 1.7. The spatial and time-dependence characteristics of the radioactive sources were derived using Regulatory Guide 1.183. The resulting source distribution is as conservative or in some cases more conservative than the NRC-prescribed post-accident source distribution stated in NUREG-0578 or NUREG-0737. For a BWR, this distribution carries considerable radioactivity (concentrations) throughout the plant via piping which contains suppression pool water, and by airborne particulate and noble gas concentrations in secondary containment. However, based on the fact that no operator actions other than those which take place in the control room or at the remote shutdown panel are critical for plant shutdown, only these areas and the sampling stations and Technical Support Center (TSC) are considered to be vital for personnel access for post-accident cases.

In general, the shielding design review shows that the vital areas (the control room, the auxiliary electric equipment room, where the remote shutdown panels are located, and the Technical Support Center) have dose rates which allow continuous occupancy for the accident scenario. The results also show that accessibility to these areas is not a problem during such accidents.

Application of General Design Criterion 19 accident limit of 5 rem whole body (or equivalent) for areas requiring infrequent access indicates that adequate occupancy times are available for typical operator actions. This remains true with application of Alternative Source Terms, with the 10 CFR 50.67 accident limit of 5 rem TEDE. The significant radiological conclusion is that the "less than 15 mr/hr" (averaged over 30 days) criteria is met at LaSalle for plant areas requiring extended or continuous occupancy. Post-accident dose rates from contained sources are shown in Table 12.3-8. Table 13.3-8 gives the post-accident integrated doses for personnel transit between vital reactor shutdown areas and other post-accident essential areas. This table gives a conservative estimate of the travel dose at 1 hour, 1 day, and 1 week after a LOCA due to sources contained within the reactor building. Two traveling speeds and two radiation sources (contained sources with and without airborne sources in secondary containment) are considered. The essential areas including stairways are indicated in Figure 12.3-4.

12.3.2.5.1 Areas of Highly Restricted Access (Reactor Building)

The entire reactor building could experience a high airborne activity if significant primary containment or ECCS leakage occurs. Access time could be limited to 1 minute. Because of the short access time, the post-accident sampling equipment has been located in the upper basement of the auxiliary building.

For the post-accident situation, the major exposure to ECCS equipment located outside primary containment comes from the liquid contained within the equipment. Redundant equipment which is not utilized will experience very little radiation degradation. ECCS equipment which handle post-accident liquids could experience radiation degradation. The ECCS equipment inside the secondary containment are segregated to minimize external sources of radiation. Flushing and draining provisions are available, but significant amounts of airborne activity (which can only be removed by the SGTS) could limit access to this equipment.

The SGTS charcoal filter could become extremely radioactive if the airborne concentration of radioactive halogens is high. SGTS essential equipment that are radiation sensitive have been shielded so that the integrated dose they receive is below the qualification dose requirements. Therefore, the SGTS should not fail due to radiation degradation.

The standby liquid control system will experience almost no dose during the time that it might be required to function (first few minutes following a reactor scram signal). With the post-accident source terms, the primary to secondary containment leakage criterion eliminates any reasonable corrective action to improve access to secondary containment.

12.3.2.5.2 Areas of Restricted Access Following Severe Accident

The pre-TMI or non-HRSS post-accident sampling system, HVAC rooms, waste tanks rooms, and the east end of the radwaste tunnel will require restricted access following an event giving Regulatory Guide 1.3 release. Access into these areas will be performed in accordance with the Health Physics Program described in Section 12.5 of the UFSAR.

In the event that the reactor building receives primary containment leakage, the top floor of the auxiliary building would become a restricted access area. The original stack monitoring panel which was retained on this level will transfer its monitoring functions to the high range stack monitor whenever it measures a prescribed radiation level. The high range monitor is located at 786 ft 6 in. elevation and has a 2-foot concrete ceiling which protects the monitor from the effects of potential airborne radiation sources that could accumulate above the refueling floor.

If the refueling floor volume contains airborne post-accident sources, surrounding buildings will experience a high skyshine radiation dose rate (>100 mrem/hr) in areas where little or no radiation protection exists. This situation can persist for a week or more. Access to these areas will be controlled in accordance with the Health Physics Program.

12.3.2.5.3 Areas of Extended Occupancy

The dose information in this section 12.3.2.5.3 is based on the pre-power uprate original power level (3293 MWt). Use of the design information presented must account for possible effects that a power uprate (up to 3908 MWt) has on plant configuration, design bases functions, and design margins. However, as described in the power uprate Radiological evaluation (Reference 23), the overall conclusions of the analysis are not changed due to the higher power level assumed. During post-accident conditions, operators using the sampling room will be able to gather post-accident samples and not exceed the exposure limits set by NRC guidelines.

Additional shielding has been incorporated into the north, south, and west side entrances of the reactor building in order to reduce radiation streaming and to improve access to the diesel generator buildings, the station laboratories, and the facilities adjacent to the control room.

The post-accident sampling room is designed to limit the integrated dose to the operator to less than 1 rem while taking one set of post-accident samples (which could include an undiluted primary coolant sample, a containment air sample, and a diluted coolant sample). After the first set of samples, the dose to the operator is expected to be less than 100 mrem per set of five samples (a secondary containment air sample, two drywell sump samples and an RHR coolant sample are included). The samples are transported in shielded casks to either onsite or offsite laboratories. Accumulated dose to personnel who are required to spend time in these areas will be controlled so that no individual will exceed the exposure limits set by 10 CFR 20.

12.3.2.5.4 Areas of Continuous Occupancy

The sheilding design of the control room, the auxiliary electric equipment room, and the Technical Support Center satisfies 10 CFR 50.67. Accumulated dose to personnel traveling between these facilities will be within the normal control of the Health Physics Program.

12.3.3 Ventilation

12.3.3.1 Design Objectives

In addition to their primary function of providing suitable environmental conditions that are safe and comfortable for operating personnel and adequate for the functioning of equipment, the plant ventilation system will provide effective protection for operating personnel against possible airborne radioactive contamination. The system is designed to operate such that the in-plant airborne activity levels for normal operation (including anticipated operational occurrences) in normally occupied areas are within the limits of 10 CFR 20. The system operates to mitigate the spread of airborne radioactivity during normal and anticipated abnormal operating conditions. The system provides a suitable environment for equipment and continuous occupancy in the control room under postaccident conditions. Also, the ventilation system operates to ensure compliance with normal operation offsite release limits.

12.3.3.2 Design Criteria

To meet the design objectives, the following radiological safety design guidelines were utilized:

a. The system is designed to maintain air flows from clean areas to potentially contaminated areas and from areas of potentially lower level contamination to areas of potentially higher level contamination (prior to exhaust).

b. The system is designed to ensure that negative pressure differential with respect to surrounding areas is maintained inside potentially contaminated cubicles. Control dampers and seals are provided to assure that airflow patterns can be properly maintained.

c. Fume hoods are utilized in the laboratories to facilitate safe processing of radioactive samples by directing contaminants away from the breathing zone to the filtering and ventilation system.

d. Equipment decontamination facilities are ventilated to assure control of released contamination and prevent personnel exposure and the spread of contamination.

e. Exhaust air is routed through HEPA filters or a combination of HEPA and charcoal filters where necessary before release to the atmosphere to reduce onsite and offsite radioactivity levels.

f. Air is supplied to each principal building via separate supply intakes and duct systems.

g. The fresh air supply to the control room is designed to be operable during loss of offsite power. The air is filtered and can be passed through charcoal adsorbers to prevent contamination of the control room by smoke or excessive radioactivity.

h. Transient airborne contamination may result due to the maintenance. Special procedures, such as: system isolation, the use of flexible vent duct or portable air handling units, and the use of plastic tents will be instituted to minimize the contamination on a case by case basis.

i. All exhaust ventilation systems designed to handle potentially contaminated air in the plant are of similar design. A typical filtration system is equipped with a demister and/or prefilter, a heater for humidity control, a set of prefilters and a set of HEPA filters. Filter systems designed to remove radioiodine are equipped with a charcoal filter bank and an additional set of HEPA filters to collect charcoal fines emerging from the charcoal filters. Dampers are provided before and after the filter train to isolate the train during filter changes.

j. All filter systems in which radioactive materials could accumulate to produce significant radiation fields external to the ductwork are appropriately located and shielded to minimize exposure to personnel and equipment (Table 12.3-6).

k. Filters in all systems are changed based upon 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 test samples or canisters removed and analyzed at intervals.

l. While the majority of the activity in the filter train is removed 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.

These guidelines are incorporated and fully described in Section 9.4.

12.3.3.3 Ventilation Design Features

The ventilation system parameters for radiologically significant areas such as the auxiliary building, off-gas building, radwaste building, reactor building, service building, and turbine building are provided in Tables 12.3-9 through 12.3-14.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Four systems are provided to monitor radiation/radioactivity levels within the plant. These are:

a. the fixed area radiation monitoring system (ARMS),

b. the continuous airborne radioactivity monitoring system (CAM),

c. high range instrumentation, and

d. the fixed air sampling system.

The fixed ARMS is provided to continuously measure, indicate, and record the levels of radiation in general access areas and to activate alarms when predetermined levels are exceeded. The general objective is to keep operating personnel informed of the radiation levels in the selected areas and thus assist in avoiding unnecessary or inadvertent exposure.

A cart system (CAM) is provided to measure, indicate, and record the levels of airborne radioactivity at locations where significant airborne radioactivity is likely. Each cart activates a local alarm when predetermined levels are exceeded.

A high range area radiation monitor is provided for the reactor building crane.

The fixed air sampling system provides the means for sampling areas, cubicles and ducts in the plant to identify trends in air concentration levels and the source of the activity.

12.3.4.1 Fixed Area Radiation Monitoring Instrumentation

The fixed area radiation monitoring system (ARMS) is provided to fulfill the following specific radiological safety objectives:

a. To provide operating personnel with a record and indication in the main control room of gamma radiation levels at selected locations within the various plant buildings (e.g., to warn of excessive gamma radiation levels in areas where nuclear fuel is stored or handled).

b. To contribute radiation information to the control room so that correct decisions may be made with respect to deployment of personnel in the event of a radiation incident.

c. To assist in the detection of unauthorized or inadvertent movement of radioactive material in the plant including the radwaste area.

d. To supplement other systems including process radiation monitoring, leak detection, etc., in detecting abnormal migrations of radioactive material in or from the process streams.

e. To provide local alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area.

f. To assist in maintaining exposure to personnel as low as practicable.

To implement these objectives, area radiation monitors are provided throughout the plant at locations indicated in Table 12.3-15 and as shown on radiation zone maps, Figures 12.3-1 and 12.3-2. Setpoints will be approximately 1.2 times background, but specific monitors have higher setpoints due to unusual requirements.

Energy Dependence

The detector-indicator and trip unit are responsive to gamma radiation over an energy range of 80 keV to 7 MeV. The energy dependence does not exceed ± 20% of indicated point value for a dose rate of approximately 50 mr/hr from 100 keV to 3 MeV and there is response from 80 keV to 7 MeV.

Accuracy

The overall accuracy within the design range of temperature, humidity, line voltage and line frequency variation is such that the actual reading relative to the true reading, including susceptibility and energy dependence (100 keV to 3 Mev) is within 9.5% of equivalent linear full scale recorder output for any decade.

Reproducibility (Precision)

At design center the reading is reproducible within ± 10% of point with constant geometry.

Power Supply

All area radiation monitors receive power from a 120 Vac instrument bus. The audio alarms receive power from a local 120 Vac distribution panel. The ARMS is not a nuclear safety-related system and is not required to be on the emergency power bus.

Calibration

Area radiation monitors will be calibrated at least once per refueling outage. Additionally, source checks are made routinely to validate the operability of these monitors.

Area radiation monitors have indicator/trip units in the main control room for both high and low alarms. Recorders are provided for the ARMS in the main control room. Area radiation monitors are not required to function under all design-basis accident conditions (e.g., LOCA); however, ARMS are designed to fail in the safe (alarm) mode.

A high range area radiation monitor is provided near the reactor on the refueling floor to sense abnormal or accident conditions. The ARMS conforms to Sections 4.2 and 5.3.4.11 of ANSI N13.2-1969. Qualified personnel have been used in the engineering phase and will be used during operation to ensure that radiation exposures will be ALARA. Recorders provided with the ARMS provide records of radiation levels. Calibration sources and a calibration facility are available for source checks of the ARMS systems. Regulatory guidance concerning effluents and solid wastes, and ANSI 13.1-1969 do not directly apply to the ARMS.

12.3.4.2 Continuous Airborne Radioactivity Monitoring Instrumentation

Eleven cart-mounted continuous airborne monitors (CAM's) and a fixed air sampling system are provided for monitoring in-plant airborne activity levels.

The CAM's, whose location is indicated in Table 12.3-17, are normally assigned to monitor the following plant areas and sample points:

a. refueling floor of the unit being refueled (two provided);

b. primary containment - Unit 1;

c. primary containment - Unit 2;

d. off-gas filter building exhaust;

e. reactor building ventilation exhaust - Unit 1;

f. reactor building ventilation exhaust - Unit 2;

g. turbine building ventilation exhaust - Unit 1;

h. turbine building ventilation exhaust - Unit 2;

i. radwaste building ventilation exhaust; and

j. machine shop.

For the monitoring of the exhaust ducts noted in items d, e, f, g and h, the CAM's are coupled to in-duct isokinetic probes, these probes comply to the standards set forth in ANSI N13.1-1969.

As is needed, the refueling floor CAM's may be temporarily removed from their normal assignment for calibration, repair, or to monitor specific work areas where significant airborne radioactivity is probable, or suspected. These areas may be monitored by the direct presence of the CAM or by the use of a vacuum tube extension.

The CAM's have the following specifications:

Flow rate:	2 to 10 cfm (adjustable)
Energy Response Range:	0.08 to 3.0 MeV for gamma
Range:	50 to 50,000 cpm (minimum)
Accuracy:	±20%
Precision (Reproducibility):	± 10% at 95% confidence

Air collected by the CAM is passed first through a fixed particulate filter for continuous monitoring and then through a charcoal canister for iodine sampling. The two CAM's for Units 1 and 2 primary containment also have noble gas monitoring channels. The accumulated particulate deposition is monitored for gross beta/gamma. The output, in counts per minute, is displayed on a count rate meter and recorded on a strip chart recorder. The iodine canister is analyzed in the lab.

AM's are equipped with adjustable high and low radiation alarm setpoints. They are also designed to give alarm in the event of low airflow or a loss of power. The alarm mechanism, which is integral with the CAM cart, produces both audible and visual indication. The setpoints for high radiation alarm are indicated in Table 12.3-17.

These CAM's are normally used as trending devices only. Thus, traceable source calibration is not required. Source checks, however, will be performed as specified by the Radiation Protection Department.

Electric power is provided to the CAM's from local outlets. Emergency power is not provided since CAM's are not a part of an essential safety system.

CAM's facilitate administrative procedural compliance to ANSI N13.2-1969. Guidance in ANSI N13.1-1969 has been utilized as a guide in the design of CAM's and their associated isokinetic probes.

12.3.4.3 High Range Instrumentation

An area radiation monitor with a range of 0.01 mR/hr to 10,000 mR/hr is provided for the reactor building crane. It will prevent upward lifting with the crane upon high radiation alarm.

12.3.4.4 Fixed Air Sampling System

A fixed air sampling system is provided to monitor trends in the station's airborne radioactivity and thereby gives warning of, and identifies increasing levels of activity and leakage. 126 sample lines draw air from cubicles, rooms, ducts, and areas throughout the plant and deliver it to eight local sample panels. A list of the sample panels and their associated sample lines is found in Table 12.3-16.

The appropriate fixed air sampling system may be actuated to detect leaks in the areas where and when one of the CAM's monitoring the ventilation exhaust has alarmed.

From each sample point, air is drawn at a high velocity through 1/2 inch stainless steel piping. At the sample panels each sample stream is passed through replaceable particulate and charcoal filters. These filters are removed periodically for laboratory analysis. The flow rate of each sample stream is monitored, displayed and controlled at the sample panel. The driving force at each panel is a high volume vacuum pump which draws on a manifold into which the sample lines empty after filtration, metering and valving.

The fixed air sampling system is not an essential safety system and is therefore not provided with emergency power.

The fixed air sampling system provides compliance with the requirements of ANSI N13.2-1969 through the utilization of guidance set forth in ANSI N13.1-1969.

12.3.5 References

1. W. W. Engle, Jr., "A Users Manual for ANISN, A One-Dimensional Discrete-Ordinates Transport Code with Anisotropic Scattering," K-1693, Union Carbide Corporation, Nuclear Division, March 30, 1967.

2. RSIC Data Library, "DLC-23/CASK 40 - Group Coupled Neutron and Gamma-Ray Cross Section Data," Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, December 1972.

3. J. H. Hubbell, "Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 keV to 100 GeV," NSRDS-NBS29, August 1969.

4. S. Buscagline and R. Manzini, "Buildup Factors: Coefficients of the J. J. Taylor Equation," ORNL-tw-80, February 1964.

5. R. L. Engle, J. Greenborg, and M. M. Hendrickson, "ISOSHLD - A Computer Code for General-Purpose Isotope Shielding Analysis," BNWL-236, Pacific Northwest Laboratory, Richland, Washington, June 1966; Supplement 1, March 1967; Supplement 2, April 1969.

6. R. E. Malenfant, "QAD: A Series of Point-Kernel General-Purpose Shielding Programs," LA-3573, Los Alamos Scientific Laboratory, April 5, 1967.

7. R. E. Malenfant, "G^3 : A General-Purpose Gamma-Ray Scattering Program," LA-5176, Los Alamos Scientific Laboratory, June 1973.

8. D. K. Trubey and M. B. Emmett, "OGRE-G, A General-Purpose Monte Carlo Gamma-Ray Transport Code," ORNL-TM-1212, Oak Ridge National Laboratory, 1966.

9. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.3, Rev. 2, Directorate of Regulatory Standards, U.S. Atomic Energy Commission, June 1974.

10. "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Practicable," Regulatory Guide 8.8, Directorate of Regulatory Standards, U.S. Atomic Energy Commission, July 1973.

11. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," Regulatory Guide 1.5, Directorate of Regulatory Standards, U.S. Atomic Energy Commission, February 1, 1971.

12. "Fuel Storage Facility Design Basis," Regulatory Guide 1.13, Directorate of Regulatory Standards, U.S. Atomic Energy Commission, February 1, 1971.

13. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, Directorate of Regulatory Standards, U.S. Atomic Energy Commission, March 23, 1972.

14. R. A. Burns, "Radiation Levels and Shielding at the Nine Mile Point BWR," Transactions, American Nuclear Society, Vol. 16, p. 331, 1973.

15. "Program for Testing Biological Shielding in Nuclear Reactor Plants," ANS 6.3, Proposed Standard, American Nuclear Society.

16. Gundremminger Nuclear Power Plant, Annual Report, EUR-4396, 1969.

17. Gundremminger Nuclear Power Plant, Annual Report, EUR-4706, 1970.

18. A. Weckesser and N. Eickelpasch, "Construction and Operating Experience of LWR's in Germany, "Transactions," American Nuclear Society, Vol. 15, p. 842, 1972.

19. M. Cometto et al., "Radiation Protection Experience in Italian Nuclear Power Plants," Fourth United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 1971.

20. "A Compilation of Occupational Radiation Exposure from Light Water Cooled Nuclear Power Plants, 1969-1973," WASH-1311, Radiological Assessment Branch, Directorate of Licensing, USAEC, May 1974.

21. C. A. Pelletier et al., "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants," Atomic Industrial Forum (Prepared by SAI Services), 1974.

22. C. M. Lampley, "The SKYSHINE - II Procedure: Calculation of the Effects of Structure Design on Neutron, Primary Gamma-Ray, and Secondary Gamma-Ray Dose Rates in Air," NUREG/CR-0781 (RRA-T7901), Oak Ridge National Laboratory-Radiation Shielding Information Center, Radiation Research Associates, May 1979.

23. NDIT LAS-ENDIT H040, Task Number: 21E: LaSalle Power Uprate Project, Sargent & Lundy Engineering Evaluation for Radiological / PASS (Post Accident Sample System), Engineering Task Evaluation 21E, Control No. H040, Revision 2, approved 12/27/99.

TABLE 12.3-1

CLASSIFICATION OF RADIATION ZONES
(FOR SHIELD DESIGN AND ACCESS CONTROL)

ZONE DESIGNATION	DESIGN DOSE RATE* (mR/hr)	AREA CONTROL DESIGNATION ** ***	USNRC POSTING REQUIREMENTS
I	< 1	Uncontrolled access; personnel monitors may be necessary***	None****
II	< 2.5	Uncontrolled or Controlled; personnel monitors may be necessary***	None****
III	< 15	Controlled (if greater than 5 mrem/hr); use of direct-reading instruments by workers or visitors.	Radiation Area sign
IV	< 100	Controlled; prior survey by Radiation Controlled (if greater than 5 mrem/hr); use of	Radiation Area sign
V	> 100	Controlled with barricade or locked door; intermittent monitoring as necessary by Radiation Protection Technician	High Radiation Area sign or Very High Radiation sign

* For a given operation mode, the maximum dose rate calculated in a given region, excepting localized radiation streaming paths, will satisfy the design dose rate chosen for that region.

** Access control definitions are given in the American Nuclear Society's Proposed Standard, *Program for Testing Biological Shielding in Nuclear Reactor Plants," July 1967, Volume 5-1, ANS 6.1, Page 5 and subsequent revisions.

*** The 10 CFR 20.1502 requires a radiation monitor be worn by any adult individual who are likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits in 10CFR20.1201 (a).

**** This zone may require posting with a "RADIATION AREA" sign, in accordance with 10 CFR 20.1003 if an area, accessible to individuals, exists in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

TABLE 12.3-2

SHIELDING PROPERTIES OF MATERIALS

	LEAD	STEEL	CONCRETE (TYPE 04-A)	WATER
Density at 70° F (gm/cc)	11.34	7.92	2.35	1.0
Fast Neutron Removal Cross Section Σ_r (cm^{-1})	0.116	0.17	0.079	0.1
Gamma Ray Attenuation Coefficients				
at 1 MeV	0.797	0.472	0.149	0.0707
at 6 MeV	0.505	0.269	0.063	0.0274
Thermal Neutron Absorbtion Cross Section Σ_a (cm^{-1})	0.00648	0.24	0.0086	0.0226
Capture Gamma Energy Spectra Range, MeV	7-9	0.1-10	1-6	2-3
* Principal Activation Isotopes	Pb209; Pb210	Fe-59; Co-60; Cr-51; Mn-56; Ni-65	Na-24; Si-31; Fe-59; Ca-47	H-3; 0-19 N-16
γ Activation Energies (MeV)	0; 0.05	1.1, 1.29; 1.33, 1.17; 0.32; 0.84, 1.81; 1.49, 1. 11	2.75, 1.37; 1.27; 1.1, 1.29; 1.3, 0.5	0; 0.2, 1.36
β Activation Average Energies (MeV)	0.2; 0.02, 0.004	0.17, 0.095, 0.62; 0.106; 0; 1.22, 0.39, 0.27; 0.87, 0.38, 0.23	0.56, 0.6; 0.095; 0.62; 0.81, 0.24	0.006; 1.49, 2.2

* For isotope half lives and yields consult Table of Isotopes by C. M. Lederer, J. M. Hollander, and I. Perlman.

TABLE 12.3-3
(SHEET 1 OF 2)

REACTOR BUILDING DESIGN DATA*

COMPONENT**	SOURCE (TABLE)	SHIELD MAT'L	SHIELD THICKNESS (inches) W	C_l	F	DOSE RATE (mrem/hr) INSIDE	OUTSIDE
[redacted]	12.2-6	C	44	40a	36	1.0×10^3	1.0
	12.2-6	C	24a	24	30	$<5.0 \times 10^2$	1.0
	12.2-6	C	44	X	32	5.3×10^2	1.0
	11.2-5	C	44a	44a	44	$<2.0 \times 10^4$	1.0
	11.2-5	C	24	24	44	5.0×10^2	1.0
	12.2-6	L C	6 36	 X	 42	 2.7×10^3	 1.0
	12.2-6	C	32	X	42	6.5×10^2	1.0
	12.2-6	C	24	X	32	3.8×10^2	2.0
	12.2-6	C	36	42	42	1.1×10^3	1.0
	12.2-6	C	33a	42	33	1.1×10^3	2.0
	12.2-4	C	72			NA	1.0
	12.2-4	C	24	12a	X	2.1×10^2	1.0
	12.2-4	Controlled Access			X	$<8.0 \times 10^1$	1.0
	12.2-5	C	56	56a		$<3.0 \times 10^3$	1.0
	Quad Cities Station Experience	C	36	36	36	2.0×10^4 Point Source	2.5
	12.2-4	C	52		36	$<2.0 \times 10^3$	1.0
	12.2-5	C	36			$<5.0 \times 10^2$	1.0

TABLE 12.3-3
(SHEET 2 OF 2)

COMPONENT	SOURCE (TABLE)	SHIELD MAT'L	SHIELD THICKNESS (inches)			DOSE RATE (mrem/hr)	
			W	C_l	F	INSIDE	OUTSIDE
[redacted]	12.2-5	C	36			<3.0 x 10^2	1.0
	12.2-4	C	36			<5.0 x 10^2	1.0
	12.2-4	C	36a	36	36	6.0 x 10^2	1.0
	12.2-9	C	28	27		1.3 x 10^3	1.0
	12.2-9	C	24		18	1.3 x 10^3	1.0
	12.2-4	C	24	X		3.0 x 10^2	1.0
	12.2-5	C	24	28		1.0 x 10^2	1.0
	12.2-4	C	24			<3.0 x 10^1	1.0

* Abbreviations used:

C	=	concrete
Chem	=	chemical
Demin	=	demineralizer
I	=	Iron
L	=	Lead
Tk	=	Tank
a	=	shield contains a removable wall section or plug
X	=	design of shield is determined by other sources
W	=	Wall
C_l	=	Ceiling
F	=	Floor

** Component name, floor elevation, column coordinates

TABLE 12.3-4

TURBINE BUILDING DESIGN DATA*

COMPONENT**	SOURCE (TABLE)	SHIELD MAT'L	SHIELD THICKNESS (inches) W	C_l	F	DOSE RATE (mrem/hr) INSIDE	OUTSIDE
[redacted]	12.2-8	C	42	36a	36	<3.0 x 10^2	1.0
[redacted]	12.2-5 12.2-8	C	52	 48	48	3.3 x 10^3	1.0 2.5
[redacted]	11.3-3	C	52	42	48	<1.0 x 10^5	1.0 2.5
[redacted]	12.2-8	C	44		X	<3.0 x 10^3	1.0
[redacted]	12.2-8	C	44		X	<3.0 x 10^3	1.0
[redacted]	12.2-8	C	30a			<2.0 x 10^3	1.0
[redacted]	12.2-8	C	56a	48	64	1.7 x 10^3	1.0
[redacted]	12.2-8	C C (Plugs)	48 24	36	48	<1.0 x 10^3	1.0 2.5
[redacted]	12.2-8	C	48a	48	60	<5.0 x 10^3	1.0
[redacted]	11.2-5	C	36a	30	48	1.0 x 10^4	1.0
[redacted]	12.2-5	C	56a	60	60	2.0 x 10^3	1.0
[redacted]	12.2-8	C	24	X	39	2.0 x 10^2	1.0
[redacted]	12.2-5	C	24	X	39	5.0 x 10^1	1.0
[redacted]	12.2-5	C	20	X		<1.0 x 10^1	1.0
[redacted]	12.2-8	C	52a			5.0 x 10^3	1.0
[redacted]	12.2-5	C	20	24		1.0 x 10^1	1.0

*Abbreviations used:

C = concrete
Chem = chemical
Demin = demineralizer
I = Iron
L = Lead

Tk = Tank
a = shield contains a removable wall section or plug
X = design of shield is determined by other sources

TABLE 12.3-5
(SHEET 1 OF 2)

RADWASTE BUILDING DESIGN DATA*

COMPONENT**	SOURCE (TABLE)	SHIELD MAT'L	SHIELD THICKNESS (inches)			DOSE RATE (mrem/hr)	
			W	C_l	F	INSIDE	OUTSIDE
[redacted]	11.2-5	C	32	42		1.0×10^3	1.0
[redacted]	11.2-5	C	40	42		6.0×10^3	1.0
[redacted]	11.2-5	C	20	42		3.5×10^1	1.0
[redacted]	11.2-5	C	48	48a	48	3.8×10^4	1.0
[redacted]	11.2-5	C	48	48	48	2.4×10^5	2.5
[redacted]	11.2-5	C	20	48		6.5×10^2	1.0
[redacted]	11.2-5	C	48	48a	48	3.8×10^4	1.0
[redacted]	11.2-5	C	24a	26	X	1.0×10^2	1.0
[redacted]	11.2-5	C	40	42		5.2×10^4	1.0
[redacted]	11.2-5	C	40	X		2.3×10^3	1.0
[redacted]	11.2-5	C	20	42		1.3×10^2	1.0
[redacted]	11.2-5	C	20	26	42	3.5×10^2	1.0
[redacted]	11.2-5	C	20	X		4.5×10^1	1.0
[redacted]	11.2-5	C	32	42		2.1×10^3	1.0
[redacted]	11.2-5	C	20	X		>1.0	1.0
[redacted]	11.2-5	C	X	X		>2.5	2.5
[redacted]	11.2-5	C	40		36	2.4×10^4	1.0

TABLE 12.3-5
(SHEET 2 OF 2)

COMPONENT**	SOURCE (TABLE)	SHIELD MAT'L	SHIELD THICKNESS (inches) W	C_l	F	DOSE RATE (mrem/hr) INSIDE	OUTSIDE
[redacted]	11.2-5	C	40		36	2.4 x 10^4	1.0
[redacted]	11.2-5	C	48		36	4.7 x 10^4	1.0
[redacted]	11.2-5	C	48	54		4.3 x 10^4	1.0
[redacted]	11.2-5	C	44	54		2.2 x 10^3	1.0
[redacted]	11.2-5	C	44	48		6.1 x 10^4	1.0
[redacted]	11.2-5	C	40	54		2.5 x 10^3	1.0
[redacted]	11.2-5	C	32	48		2.5 x 10^3	1.0
[redacted]	11.2-5	C	36	36		6.0 x 10^4	1.0
[redacted]	11.2-5	C	36	36		5.0 x 10^2	1.0
[redacted]	11.2-5	C	36	36		2.0 x 10^2	1.0
[redacted]	11.3-3	C	48	48		1.0 x 10^3	1.0
[redacted]	11.3-3	C	24	24		<5.0 x 10^2	1.0
[redacted]	11.3-3	C	57	48		2.0 x 10^4	1.0
[redacted]	11.3-3	C	48	54	48	<2.0 x 10^3	1.0
[redacted]	11.3-3	C	18	24	X	<5.0 x 10^2	1.0

* Abbreviations used:
C = concrete
Chem = chemical
Demin = demineralizer
I = iron
L = lead
Tk = tank_
a = shield contains a removable wall section or plug
X = design of shield is determined by other sources
**Component name, floor elevation, column coordinates

TABLE 12.3-6

DESIGN DATA FOR REMAINDER OF STATION*

COMPONENT**	SOURCE (TABLE)	SHIELD MAT'L	SHIELD THICKNESS (inches)			DOSE RATE (mrem/hr)	
			W	C_l	F	INSIDE	OUTSIDE
[redacted]	DBA	C	56	36	6	<5 rem TEDE for duration of accident	
	11.2-5	C	36			1.0 x 10^3	1.0
	12. 2-8	C	56			1.0 x 10^3	
	12.2-8	C	56			2.0 x 10^3	1.0
	(a)	C	12			2.0 x 10^1	1.0
	(a)	C	12			2.0 x 10^1	1.0
	Calibration Source	C	12	24		8.6 x 10^1	1.0
	11.2-5	C	18		12	3.0 x 10^1	1.0
	12.2-8	C	56a	60	60	2.0 x 10^3	1.0
	(b)	C	8			5.0	1.0
	(c)	C	12			5.0	1.0
	(d)	C	12			2.0 x 10^1	1.0
	12.4.3.1	C	30	12		5.0 x 10^3	1.0
	(e)	C	N/A	~28	N/A	N/A	(f)

* Abbreviations used:
C = concrete
Chem = chemical
Demin = demineralizer
I = Iron
L = Lead
QCS = Quad-Cities Station
Tk = Tank
a = shield contains a removable wall section or plug
X = design of shield is determined by other sources
(a) Small sealed volume of reactor water or calibration source.
(b) Shielding based upon 2.5 mR/h contact dose on stored tools.
(c) Shielding is provided for expected lint buildup in the dryers.
(d) Shielding is provided to protect machine shop workers from radioactive components and tools stored in the decontamination room.
(e) Calculation No. L-003748.
(f) Contact dose rate per Calculation No. L-003748.

**Component name, floor elevation, column coordinates.

TABLE 12.3-7
(SHEET 1 OF 2)

POST-ACCIDENT DOSE RATES PROM PIPE CONTAINED SOURCES
(in rad/hr)+

LOCATION	1-HOUR	1-DAY	1-WEEK	CRITERION
Control Room (a)(b)	<0.001	<0.001	<0.001	10 CFR 50.67
Remote Shutdown Panel (b)	<0.001	<0.001	<0.001	10 CFR 50.67
Technical Support Center (b)(c)	<0.001	<0.001	<0.001	10 CFR 50.67
Post-Accident* Sampling Room	0.015-0.1	0.015-0.1	0.015-0.1	10CFR20**
Diesel Generator Buildings*** (b, e)	<0.015-0.1	<0.015-0.1	<0.015	10CFR20 and <100 mrem/hr in NUREG-0578
Laboratories (d, f)	<0.015	<0.015	<0.001	10CFR20 and <15 mrem/hr in NUREG-0578
Pathway to TSC	<0.015-0.5	<0.015-0.5	<0.015-0.1	10CFR20
Stack Monitors at el. 786'-6"	0.015-0.1	0.015-0.1	0.015-0.1	<25 rads/hr background NUREG-0578

+The dose rate information presented in this Table increases by less than 25% due to a power uprate up to 3908 MWt.

* These are the values resulting from the expected radiation source contained in the panels. The unshielded samples or undiluted primary coolant in the panels would produce a much higher radiation field.

** The design dose of less than 1.25 rem for obtaining the initial samples is below the 10CFR20 limits and is achievable with the sampling equipment installed at LSCS assuming the R.G. 1.183 sources are present in the primary coolant samples. This is less than the NUREG-0737 II.B.3 criterion.

*** Ground level. When RHR is operating.

TABLE 12.3-7
(SHEET 2 OF 2)

(a) The operational support center and the master security center have the same dose rates as the control room.

(b) These areas contain instrument panels and/or MCCs.

(c) The radwaste control panel(s) are located in area(s) where the dose rate due to post-accident sources is much less than 1 mr/hr.

(d) The switchgear rooms have dose rates that are much less than the conservative estimate given for the laboratories.

(e) This area requires access on an irregular basis, not continuous occupancy. Per NUREG-0737 Section II.B.2, shielding is provided to allow access at a frequency and duration estimated by the licensee. Specific post-accident tasks will be evaluated on an individual basis in accordance with the Station's Health Physics Program (UFSAR Section 12.5). This area may therefore require posting as a high radiation area (.> 100 mrem/hr) for a short period of time post-accident. The guidelines in 10 CFR 50.67 (5 rem TEDE whole body dose during the course of the accident) will not be exceeded.

(f) Per NUREG-0737 Section II.B.2, the dose rate criterion of 15 mrem/hr (0.015 rad/hr) is an average over a 30-day period.

TABLE 12.3-8

POST-ACCIDENT INTEGRATED DOSES FOR ESSENTIAL POST-ACCIDENT PATHS (3
(DOSES ARE IN MILLIREMS FOR SELECTED POST-ACCIDENT TIMES)++

PATHWAY		WALK (1)			RUN (2)			NO AIRBORNE (3)	
		1 HOUR	1 DAY	1 WEEK	1 HOUR	1 DAY	1 WEEK	WALK	RUN
TSC TO C.R.		3.6	4.0	3.0	1.8	2.0	1.5	0.04	0.02[d]
RWCR TO C.R.		0.30	0.30	0.22	0.14	0.14	0.10	0.03	0.01[d]
LABS TO C.R.		0.03	0.02	0.01	0.02	<0.01	<0.01	<0.01	<0.01
HRSS TO LABS	NO CART	0.06	0.01	0.01	0.03	<0.01	<0.01	<0.01	<0.01
	W/CART	0.15	0.04	0.04	0.08	0.02	0.02	a	a
HRSS TO UNIT #2 TRACKWAY	NO CART	1.7	1.8	1.3	0.82	0.89	0.64	0.040c	0.02
	W/CART	3.3	3.6	2.6	1.6	1.8	1.3	c	c
C.R. TO D.G. ROOMS		0.21	0.03	0.03	0.11	0.02	0.02	b	b
C.R. TO DIV. 1 SG (4)		0.07	0.02	0.02	0.04	<0.01	<0.01	b	b
		0.04	0.03	0.02	<0.02	<0.02	<0.02	b	b
C.R. TO HPCS SG	UNIT #1	0.25	0.20	0.15	0.18	0.15	0.12	b	b
	UNIT #2	1.4	0.35	0.15	0.90	0.21	0.12	b	b
C.R. TO HVAC (SGTS SYSTEM SAMPLING AREA)		0.54	0.68	0.68	0.30	0.37	0.37	<0.01	<0.01

C.R. – Control room, TSC – technical support center, RWCR – radwaste control room, Labs – high level laboratory and counting room
HRSS – post-accident sampling room, D.G. – Diesel Generator, SG – switch gear.

a. 50% of the doses determined for the airborne cases.
b. 100% of the doses determined for the airborne cases.
c. Walk: 0.09 at 1 hour; 0.05 at 1 week; Run: 0.05 at 1 hour, 0.03 at 1 week.
d. Less than 0.01 for 1 day and 1 week (both walking and running).

Notes:
1. Walking Speeds: 300 ft/min. horizontal, 50 ft/min. up stairs, 90 ft/min. down stairs, 150 ft/min. with carts-horizontal.
2. Running Speed: 600 ft/min.. horizontal, 80 ft/min. up stairs, 120 ft/min. down stairs, 300 ft/min. with carts-horizontal.
3. Sources: Regulatory guide 1.3 and 1.7 sources were used to calculate contained sources and secondary containment airborne sources (refueling floor).
   This table does not include any doses due to a post-accident plume.
4. Dose while traveling between Control Room and Remote Shutdown panel is always less than 0.01 millirems.

[+] The dose rate information presented in this Table increases by less than 25% due to a power uprate up to 3908 MWt.

++ At worst, the pathway doses (Control Room Filter and refueling floor activity shine) with Alternative Source Terms utilization would increase proportionally to the increase in primary containment leakage utilized for Alternative Source Term evaluations (i.e., [0.0100 / 0.00635] = 1.575), in combination with the above power uprate multiplier.

TABLE 12.3-9

AUXILIARY BUILDING VENTILATION SYSTEM

SYSTEM	RADIOLOGICAL SAFETY FEATURES*	EXHAUST AIR FLOW RATE (cfm)#	FRACTION OF MPC FOR IODINE ***
Control Room	2 x 100% redundant supply fans and filter	24,840 (common)	The airborne concentration in the control room is much less than the levels in other general access areas.
Hot Laboratory	Separate ventilation exhaust system consisting of the filter system, hood, and two 50% capacity fans.**	13,820 (common)	6.8×10^{-6}
Counting Room	Counting room is provided with a positive pressure relative to surrounding areas. Back flow dampers are provided.	1,000 (common)	9.1×10^{-5}

* Detailed description of ventilation system including safety features for auxiliary building is given in Subsection 9.4.3.

** HVAC system for hot laboratory is fully described in Subsection 9.4.3.3.

*** The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 1/1/94.

The airflows shown are original design values. For required airflows refer to latest test and balance data and/or associated engineering evaluation.

TABLE 12.3-10

OFF-GAS BUILDING VENTILATION SYSTEM

SYSTEM	RADIOLOGICAL SAFETY FEATURES*	EXHAUST AIR FLOW RATE (cfm)#	FRACTION OF MPC FOR IODINE **
Off-gas (HVAC)	An exhaust system for the off-gas charcoal room which consists of an exhaust fan and dampers.	7,800 (common)	Iodine emission from the off-gas building exhaust is minimal.

* Complete description of the off-gas HVAC system including safety features is given in Subsection 9.4.8.

** The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 1/1/94.

\# The airflows shown are original design values. For required airflows refer to latest test and balance data and/or associated engineering evaluation.

TABLE 12.3-11

RADWASTE BUILDING VENTILATION SYSTEM

SYSTEM **	RADIOLOGICAL SAFETY FEATURES*	EXHAUST AIR FLOW RATE (cfm)#	FRACTION OF MPC FOR IODINE ***
Radwaste Building HVAC	Filter system for supply and exhaust air and three 50% capacity supply air fans.	98,150 (common)	1.6×10^{-2}

* Complete description of radwaste building HVAC including safety features is given in Subsection 9.4.3.

** All drumming equipment (e.g., drum processor tank, drum processor, and dry waste hydraulic compactor) have their exhaust air filtered near or in the unit, and the filtered air is ducted directly to the radwaste building exhaust. The filtering units are designed for easy filter packaging. All dry waste is packaged immediately and drummed frequently, so that only negligible quantities of contaminant become airborne.

*** The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 1/1/94.

\# The airflows shown are original design values. For required airflows refer to latest test and balance data and/or associated engineering evaluation.

TABLE 12.3-12

REACTOR BUILDING VENTILATION SYSTEM PER UNIT

SYSTEM	RADIOLOGICAL SAFETY FEATURES*	EXHAUST AIR FLOW RATE (cfm)#	FRACTION OF MPC FOR IODINE ***
Reactor Building HVAC	Three 50% supply fans, three 50% exhaust fans, filters for supply air, and two isolation valves in the common exhaust air duct for secondary containment during the abnormal operating conditions.	110,800	2.8×10^{-2}
High Pressure Core Spray Pump	Fan coil** unit/cubicle	40,000	During normal operation, the area is not contaminated.
Low Pressure Core Spray Pump	Same as HPCS**	5860	During normal operation, the area is not contaminated.
Residual Heat Removal Pump	Same as HPCS**	4600	9.4×10^{-4}

* Complete HVAC description including safety features is given in Section 9.4.

** Full description of HVAC system is given in Section 9.4.

*** The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 1/1/94.

\# The airflows shown are original design values. For required airflows refer to latest test and balance data and/or associated engineering evaluation.

TABLE 12.3-13

SERVICE BUILDING VENTILATION SYSTEM

SYSTEM	RADIOLOGICAL SAFETY FEATURES*	EXHAUST AIR FLOW RATE (cfm)#	FRACTION OF MPC FOR IODINE **
Machine Shop	Work is done under local ventilation and contaminated air is exhausted to the main stack through particulate filters.	9,050 (common)	6.4×10^{-6}
Cave	Contaminated air is exhausted to the main stack through particulate filters.	1,350 (common)	6.7×10^{-5}
Hot Tool Room	Air is exhausted to the cave.	700 (common)	1.3×10^{-4}
Decontamination Room	Contaminated air is exhausted to the main stack through particulate filters.	4,000 (common)	2.3×10^{-5}

* The detailed description of service building HVAC is given in Subsection 9.4.11.

** The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 1/1/94.

The airflows shown are original design values. For required airflows refer to latest test and balance data and/or associated engineering evaluation.

TABLE 12.3-14

TURBINE BUILDING VENTILATION SYSTEM

SYSTEM	RADIOLOGICAL SAFETY FEATURES*	EXHAUST AIR FLOW RATE (cfm)#	FRACTION OF MPC FOR IODINE **
Turbine Building HVAC	Three 50% capacity fan and filter units for outside air. Three 50% exhaust fans.	300,000 (per unit)	$1\ 0 \times 10^{-2}$

* The detailed description of turbine building HVAC including safety features is given in Subsection 9.4.4.

** The Maximum Permissible Concentration (MPC) value is based on 10 CFR 20 prior to 1/1/94.

The airflows shown are original design values. For required airflows refer to latest test and balance data and/or associated engineering evaluation.

TABLE 12.3-15
(SHEET 1 OF 8)

AREA RADIATION MONITORS
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
1-1	Standby Gas	820 ft 6 in. Col 12, Row CD	10^{0}-10^{4}		25.0	Bkg <5 mR/hr
1-2	RWCU Phase Sep.	807 ft 0 in. Col 10, Row H	10^{0}-10^{4}		25.0	Bkg <5 mR/hr
1-3	Rx Bldg Sample Sink	786 ft 0 in. Col 10-11, Row DE	10^{-1}-10^{3}	Yes	25.0	Bkg varies
1-4	Containment Purge	786 ft 6 in. Col 10-11, Row L	10^{0}-10^{4}		25.0	Bkg < 5 mR/hr
1-5	North HCU Modules	761 ft 0 in. Col 14, Row CD	10^{-1}-10^{3}		2.5	Bkg < 1 mR/hr
1-6	South HCU Modules	761 ft 0 in. Col 10, Row CD	10^{-1}-10^{3}		2.5	Bkg < 1 mR/hr
1-7	Off-Gas Equipment And Sample	754 ft 0 in. Col 13, Row Y	10^{-1}-10^{3}	Yes	2.5	Bkg < 1 mR/hr
1-8	Tip Room	754 ft 0 in. Col 10, Row DE	10^{0}-10^{4}	Yes	100.0	Bkg varies
1-9	Rx Bldg Mezzanine Floor	740 ft 0 in. Col 10, Row C	10^{-1}-10^{3}		5.0	Bkg <2.5 mR/hr
1-10	CRD Storage and Repair	740 ft 0 in. Col 15, Row H	10^{-1}-10^{3}		10.0	Bkg varies

* Including Internal "Bug" Source. '

TABLE 12.3-15
(SHEET 2 OF 8)
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
1-11	NW RHR HX	710 ft 6 in. Col. 14-15, Row H	10^{0}-10^{4}		100.0	Bkg High During RHR Use
1-12	SE RHR Hx	710 ft. 6 in. Col 9-10, Row D	10^{0}-10^{4}		100.0	Bkg High During RHR Use
1-13	Turbine Bldg Sample Sink	687 ft 0 in. Col 8-9, Row W	10^{-1}-10^{3}	Yes	2.5	Bkg < 1 mR/hr
1-14	Cond. Demin Regen Valve Aisle	687 ft 0 in. Col 12, Row WX	10^{0}-10^{4}		100.0	Bkg varies
1-15	URC Valve Aisle	687 ft 0 in. Col 13-14, Row WX	10^{0}-10^{4}		100.0	Bkg varies
1-16	RCIC Turbine	673 ft 4 in. Col 14-15, Row C	10^{0}-10^{4}	Yes	30.0	Bkg <15 mR/hr
1-17	HPCS Pump	673 ft 4 in. Col 9, Row H	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
1-18	Cond Booster Pump	663 ft 0 in. Col 9, Row WX	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
1-19	Aux. Equip. Room	731 ft 0 in. Col 15, Row JL	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
1-20	Spare					
1-21	Spare					
1-22	Spare					
1-23	Spare					

* Including Internal "Bug" Source

TABLE 12.3-15
(SHEET 3 OF 8)
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
1-24	Spare					
1-25	Spare					
1-26	Spare					
1-27	Spare					
1-28	Spare					
1-29	Spare					
1-30	Spare					
2-1	Standby Gas	820 ft 6 in. Col 18, Row CD	10^{0}-10^{4}		25.0	Bkg <5 mR/hr
2-2	RWCU Phase Sep.	807 ft 0 in. Col 10, Row H	10^{0}-10^{4}		25.0	Bkg <5 mR/hr
2-3	Rx Bldg Sample Sink	786 ft 6 in. Col 16-17, Row DE	10^{-1}-10^{3}	Yes	25.0	Bkg varies
2-4	Containment Purge	786 ft 6 in. Col 20, Row L	10^{0}-10^{4}		25.0	Bkg <5 mR/hr
2-5	North HCU Modules	761 ft 0 in. Col 20, Row CD	10^{-1}-10^{3}		2.5	Bkg < 1 mR/hr
2-6	South HCU Modules	761 ft 0 in. Col 16, Row CD	10^{-1}-10^{3}		2.5	Bkg < 1 mR/hr

*Including Internal "Bug" Source.

TABLE 12.3-15
(SHEET 4 OF 8)
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
2-7	Off-Gas Equipment And Sample	754 ft 0 in. Col 16-17, Row Y	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
2-8	Tip Room	740 ft 0 in. Col 16, Row DE	10^{0}-10^{4}	Yes	100.0	Bkg varies
2-9	Rx Bldg Mezzanine Floor	740 ft 0 in. Col 16, Row C	10^{-1}-10^{3}		5.0	Bkg <2.5
2-10	CRD Storage and Repair	740 ft 0 in. Col 21, Row G	10^{-1}-10^{3}		10.0	Bkg varies
2-11	NW RHR Hx	710 ft 6 in. Col 20-21, Row H	10^{0}-10^{4}		100.0	Bkg high during RHR Use
2-12	SE RHR Hx	710 ft 6 in. Col 16, Row C	10^{0}-10^{4}		100.0	Bkg high during RHR Use
2-13	Turbine Bldg Sample Sink	687 ft 0 in. Col 20-21, Row W	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
2-14	Cond. Demin. Regen Valve Aisle	687 ft 0 in. Col 18, Row WX	10^{0}-10^{4}		100.0	Bkg varies
2-15	Spare					
2-16	RCIC Turbine	673 ft 4 in. Col 20-21, Row C	10^{0}-10^{4}	Yes	30.0	Bkg <15 mR/hr
2-17	HPCS Pump	673 ft 4 in. Col 15, Row H	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr

*Including Internal "Bug" Source.

TABLE 12.3-15
(SHEET 5 OF 8)
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
2-18	Cond Booster Pump	663 ft 0 in. Col 21, Row WX	10^{-1}-10^{3}		2.5 Bkg	< 1 mR/hr
2-19	Aux. Equip. Room	731 ft 0 in. Col 15-16, Row L	10^{-1}-10^{3}		2.5 Bkg	< 1 mR/hr
2-20	Spare					
2-21	Spare					
2-22	Spare					
2-23	Spare					
2-24	Spare					
2-25	Spare					
2-26	Spare					
2-27	Spare					
2-28	Spare					
2-29	Spare					
2-30	Spare					
3-1	Refuel Flr High Range	843 ft 6 in. Col 15-16, Row G	10^{2}-10^{6}	Yes	1000	Bkg < 500 mR/hr
3-2	Refuel Flr Low Range	843 ft 6 in. Col 15-16, Row G	10^{-1}-10^{3}	Yes	2.5	Bk< 1 mR/hr

*Including Internal "Bug" Source.

TABLE 12.3-15
(SHEET 6 OF 8)
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
3-3	New Fuel Storage Vault	843 ft 6 in. Col 15, Row DE	10^{-1}-10^{3}		Alert 25.0 Criticality 50.0	Bkg <10 mR/hr
3-4	Refuel Flr Equipment Hatch	843 ft 6 in. Col 14-15, Row C	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
3-5	Vent Stack Sample	815 ft 0 in. Col 15, Row L	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
3-6	Main Control Rm	768 ft 0 in. Col 15, Row L	10^{-2}-10^{2}		2.5	Bkg <.5 mR/hr
3-7	HP Turbine	768 ft 0 in. Col 17, Row R	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr >
3-8	Turbine Bldg Decon Pit	768 ft 0 in. Col 13-14, Row W	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
3-9	Rx Bldg Trackway	710 ft 6 in. Col 13, Row A	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
3-10	Hot Lab Corridor	710 ft 6 in. Col 15, Row L	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
3-11	Turbine Bldg Bsmt Elevator	663 ft 0 in. Col 14, Row W	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
3-12	O.G. HVAC Exhaust Area	710 ft 6 in. Col 13, Row AB	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
3-13	O.G. Upper Bsmt	690 ft 0 in. Col 12-13, Row AB	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr

*Including Internal "Bug" Source.

TABLE 12.3-15
(SHEET 7 OF 8)
(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
3-14	O.G. Char Ads Valve Aisle	674 ft 0 in. Col 12-13, Row AC	10^{0}-10^{4}	Yes	30.0	Bkg <15 mR/hr
3-15	Service Bldg Office Corridor	726 ft 6 in. Col 27-28, Row AA	10^{-2}-10^{2}		2.5	Bkg < 1 mR/hr
3-16	Laundry	710 ft 6 in. Col 27, Row WX	10^{-1}-10^{3}		2.5	Bkg < 1 rnR/hr
3-17	Machine Shop	710 ft 6 in. Col 28, Row Y	10^{-1}-10^{3}	Yes	2.5	Bkg <1 mR/hr
3-18	Service Bldg Lunch-room Corridor	710 ft 6 in. Col 27, Row C-C	10^{-2}-10^{2}		2.5	Bkg <1 mR/hr
3-19	Spare					
3-20	Spare					

*Including Internal "Bug" Source.

TABLE 12.3-15
(SHEET 8 OF 8)

(Historical)

CHANNEL IDENT	NAME OR SERVICE	LOCATION	RANGE mR/hr	LOCAL INDICATION AND ALARM	INITIAL SETPOINT mR/hr	EXPECTED INITIAL BACKGROUND*
		RADWASTE CONTROL ROOM**				
4-1	Conc Waste Tanks (pump room)	734 ft 6 in. Col 15-16, Row Y_{AB}	10^{-1}-10^{3}		100.0	Bkg varies
4-2	Unit 1 Fl. Dr. Conc. Pump & Valve Rm	734 ft 6 in. Col 13-14, Row Y_B	10^{-1}-10^{3}		100.0	Bkg varies
4-3	Unit 2 Fl. Dr. Conc. Pump & Valve Rm	734 ft 6 in. Col 17, Row Y_B	10^{-1}-10^{3}		100.0	Bkg varies
4-4	Chem Wst Conc Pump & Valve Rm	734 ft 6 in. Col 18, Row Y_B	10^{-1}-10^{3}		100.0	Bkg varies
4-5	Radwaste Control Rm	706 ft 6 in. Col 13, Row Y_B	10^{-2}-10^{2}		2.5	Bkg <1 mR/hr
4-6	Drum Labeling Stn	706 ft 6 in. Col 15-16, Row Y_B	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
4-7	Radwaste Compactor	706 ft 6 in. Col 19, Row Y_B	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr
4-8	N. High Level Drum Storage	706 ft 6 in.	10^{0}-10^{4}		100.0	Bkg varies
4-9	S. High Level Drum Storage	706 ft 6 in.	10^{0}-10^{4}		100.0	Bkg varies
4-10	Service Bldg - Technical Support Center	694 ft 6 in. Col 29-30, Row C-C	10^{-1}-10^{3}		2.5	Bkg <1 mR/hr

* Including Internal "Buy" Source.

** Readouts, recorders and alarms to be in radwaste control room.

TABLE 12.3-16
(SHEET 1 OF 2)

FIXED AIR SAMPLING PANELS
AND AREAS FROM WHICH AIR IS SAMPLED

1. Reactor Building Air Sampling Panel (One for each Unit) (1PL83J & 2PL83J)*

   a. RWCU Recirculation Pump A-Cubicle
   b. RWCU Recirculation Pump B-Cubicle
   c. RWCU Heat Exchanger A-Valve Room
   d. RWCU Heat Exchanger B-Valve Room
   e. RWCU Phase Separator Cubicle
   f. RWCU Filt. Demin. Valve Room
   g. RWCU Demineralizer A-Cubicle
   h. RWCU Demineralizer B-Cubicle
   i. RWCU Demineralizer C-Cubicle
   j. RHR Heat Exchanger A-Room
   k. RHR Heat Exchanger B-Room
   l. Steam Tunnel
   m. RWCU Recirc. Pumps Valve Room

2. Turbine Building Air Sampling Panel (One for each Unit) (1PL16J & 2PL16J)

   a. Air Ejector Room A
   b. Air Ejector Room B
   c. Turbine Pedestal - Near Stop Valve
   d. Above Turbine Bypass
   e. Turbine Pedestal - Near Low Pressure Cylinder
   f. H.P. Heater A – Compartment
   g. H.P. Heater B – compartment
   h. Condensate Demineralizer Area – E
   i. Condensate Demineralizer Area – W
   j. Steam Tunnel
   k. Beneath Turbine Pedestal in Moisture Separator Reheater Area
   l. L.P. Turbine Area
   m. L.P. Turbine Area
   n. L.P. Turbine Area

3. Containment Air Sampling Panel (One for each Unit) (1PL15J & 2PL15J)

   24 Sample Points

4. Radwaste Area Air Sampling Panel (Common to Units 1 and 2) (0PL45J)*

   a. Ventilation Duct of Radwaste Exhaust
   b. Radwaste Area
   c. Radwaste Area

*Sampling panels are abandoned in accordance with Engineering Change 370987

TABLE 12.3-16
(SHEET 2 OF 2)

d. Radwaste Area
e. Radwaste Area
f. Radwaste Area
g. Off-gas Condenser and Recombiner A
h. Off-gas Condenser and Recombiner B

5. Off-gas Filter Building Air Sampling Panel (Common to Units 1 & 2) (0PL44J)*

a. Off-gas Prefilter, A-Cubicle
b. Off-gas Prefilter, B-Cubicle
c. Charcoal Adsorber Cubicle
d. Charcoal Adsorber Valve Aisle
e. Cooler Condenser A Cubicle
f. Cooler Condenser B Cubicle
g. Off-gas After Filter 1A-Cubicle
h. Off-gas After Filter 1B-Cubicle
i. Off-gas Prefilter 2A-Cubicle
j. Off-gas Prefilter 2B-Cubicle
k. Charcoal Adsorber Cubicle Unit 2
l. Charcoal Adsorber Valve Aisle Unit 2
m. Cooler Condenser 2A-Cubicle
n. Cooler Condenser 2B-Cubicle
o. Off-gas After Filter 2A-Cubicle
p. Off-gas After Filter 2B-Cubicle

*Sampling panels are abandoned in accordance with Engineering Change 370987

TABLE 12.3-17

CONTINUOUS AIRBORNE RADIOACTIVITY

GENERAL ARRANGEMENT ITEM NO.	NAME OR SERVICE	EQUIPMENT NUMBER	ELEVATION	RANGE COUNTS/MIN	SETPOINT
1100	Refueling Floor	0PLA1J	843 ft. 6 in.	10^1-10^7	*
1101	Refueling Floor	0PLE5J	843 ft. 6 in.	10^1-10^7	*
742	Primary Containment Unit 1 – Particulate - Gas	1PL75J	761 ft. 0 in.	10^1-10^7 10^1-10^7	** *
742	Primary Containment Unit 2 – Particulate - Gas	2PL75J	761 ft. 0 in.	10^1-10^7 10^1-10^7	** *
721	Off-Gas Filter Building Exhaust	0PL99J	690 ft. 0 in	10^1-10^7	*
685	Reactor Building Ventilation Exhaust Unit 1	1PL99J	786 ft. 6 in.	10^1-10^7	*
685	Reactor Building Ventilation Exhaust Unit 2	2PL99J	786 ft. 6 in.	10^1-10^7	*
684	Turbine Building Ventilation Exhaust Unit 1	1PLA1J	786 ft. 6 in.	10^1-10^7	*
684	Turbine Building Ventilation Exhaust Unit 2	2PLA1J	786 ft. 6 in.	10^1-10^7	*
745	Radwaste Building Ventilation Exhaust	0PLB2J	754 ft. 0 in.	10^1-10^7	*
798	Machine Shop	0PLB3J	710 ft. 6 in.	10^1-10^7	*

- *Set at 2 times 100% power background
- **Set as determined by Radiation Protection

12.4 DOSE ASSESSMENT

12.4.1 Estimated Occupancy of Plant Radiation Zones

The occupancy of any one zone is unpredictable on an average weekly basis, especially if the data is to be broken down into work function. Even to take the zone occupancy on an average yearly basis (if such data existed) would lead to erroneous conclusions because the exposure is related to the operating history.

The types of data that do exist are:

a. the total person-rem per year per station for each operating year; and

b. the percentage of the total exposure received by each type of work function (e.g., routine operations, routine maintenance, refueling, radwaste, and inservice inspection).

Much of the exposure on operating stations is due to backfitting which is not expected on the LaSalle County Station (LSCS), Units 1 and 2. In addition, LSCS was designed to reduce routine exposure especially in radwaste handling. Equipment has been chosen to reduce maintenance. These changes will cause an increase in the portion experienced by refueling because the exposure due to refueling is expected to remain constant.

Table 12.4-1 presents the fractional distribution of exposure by work function and gives the annual person-rem breakdown for the Quad-Cities Station.

12.4.1.1 Estimated Occupancy

Unavailable.

12.4.1.2 Estimated Annual Person-Rem Doses Associated with Major Function

In order to determine the person-rem doses associated with major functions, an estimate of the rem per person-year and the number of station employees receiving exposure was made.

Table 12.4-2 presents the expected total annual person-rem based on Industry's BWR operating history. The data indicates that the annual exposures increase with the operating history. There is insufficient data to determine where exposure levels off, because data include some backfitting exposure that cannot be separated out. Therefore, any decision on person-rem exposure must be based upon routine functions and the design improvements incorporated into LSCS's design. The rem per person-year for LSCS is expected to be similar to Monticello as shown in Table 12.4-2. Assuming a major decontamination of the station every 14 years, the average individual person-rem exposure is not expected to exceed 2 on an annual work basis.

If 250 employees are assumed, the maximum person-rem will be less than 500 for permanent station personnel. The annual exposure to contractors and temporary workers is expected to be less than 300 person-rem. The estimated person-rem exposure is given at the bottom of Table 12.4-2.

12.4.1.3 Major Decontamination of Equipment

Recent studies of occupational radiation exposure at operating U.S. light water reactors show an increase in the yearly average exposure. Operating ComEd stations have also shown constant increasing in occupational radiation exposure.

The increasing radiation exposure is directly related to the increasing shutdown dose rate levels of primary components.

Analysis of sludge and metal coupons from Dresden-1 show that the increases in dose levels are caused primarily by buildup of Cobalt-60.

Engineering and procedures for decontaminating the primary system components on Dresden-1 have been completed. The plans for the Dresden-1 decontamination were submitted to the NRC for review and approval (Docket #50-10, Dresden-1 Chemical Cleaning Licensing Submittal, 12-16-74). If similar decontamination becomes necessary for LSCS, EGC can use its experience and the existing decontamination technologies to keep radiation exposure during the decontamination process ALARA.

12.4.1.3.1 Frequency of Major Decontamination

The decision to perform a major decontamination on the primary components of Dresden-1 was based on the steady increase of occupational radiation exposure. ComEd health physics and management personnel determined that sometime after the 14th year of operation, Dresden-1 would require major decontamination to remove the majority of builtup Cobalt-60. This 14-year period may or may not be applicable to LSCS.

LSCS has improved cleanup systems and lower nickel and cobalt impurities, and will start with an experienced staff. These characteristics could delay major decontamination for five or more years. Whether one or two major decontaminations occur during the design life of LSCS will depend in part upon the occupational radiation exposure experienced during the 10 years of operation.

12.4.1.3.2 Equipment to be Decontaminated

Major decontamination is anticipated for many of the components which contact reactor water directly, i.e., equipment in the following systems: NB, RR, RWCU, and RHR. The types of equipment include pumps, heat exchangers, piping and the reactor pressure vessel. Major decontamination of other components will be made on an individual basis (i.e., components which cause considerable occupational exposure).

12.4.1.3.3 Radiation Exposure and Sources

The dominant radionuclide in equipment during shutdown is Cobalt-60. The activation product is deposited in the form of a thin film on the interior of piping and equipment. The decontamination process is expected to remove several thousand curies of Cobalt-60.

LSCS is expected to have a higher volume to surface area ratio than Dresden-1, resulting in lower activity of the decontamination fluid (i.e., less than the 10 μCi/cc expected at Dresden-1).

The majority of the radiation exposure during decontamination occurs while connecting and disconnecting the decontamination equipment to the system or component being decontaminated. Surveillance and monitoring account for the remainder of the radiation exposure. Leaks and emergency draining could result in additional exposure (these occurrences are discussed in Docket #500-10). The total person-rem for major decontamination is dependent upon the equipment, number of monitoring stations, and the number of systems and components to be decontaminated.

Since none of the above is known at the present time, no meaningful estimate of the total person-rem is available. The experience gained from the decontamination of older BWR's will be used to produce ALARA occupation radiation exposure at LSCS.

12.4.2 Estimates of Inhalation Doses

Small airborne concentrations of radionuclides are expected within the various plant structures. Implementation of the health physics program (Section 12.5) mitigates against any significant ingestion doses to plant personnel.

Radionuclides are a potential hazard because they may be present and released in significant quantities from fluids. For this reason, maximum design-basis radioiodine concentrations have been calculated in the buildings most susceptible to airborne contamination. The assumptions used in calculating these airborne concentrations are presented in Table 12.2-12. The resultant design-basis calculated concentrations are also tabulated in Table 12.2-12, and the thyroid dose acquired from their inhalation is tabulated in Table 12.4-3.

Limited data have been found in the literature (References 1 and 2) concerning measured airborne activity in the reactor, turbine, and radwaste buildings of operating BWR stations. The data is consistent with the calculated design-basis concentrations of Table 12.2-12.

In comparing typical airborne concentrations of I-131 in the reactor building of seven operating units, they are found to be similar, ranging generally from 1×10^{-10} to 1×10^{-12} $\mu Ci/cm^3$ with peaks for one plant (Reference 2) on the order of 4×10^{-9} $\mu Ci/cm^3$. Measurements of the other radioiodines are limited because of their shorter half lives and generally lower concentrations.

Measured average I-131 concentrations in the turbine building were found to vary from plant to plant, but in most cases the values are smaller than those in the reactor building. The limited data (from two plants) for the radwaste building shows the I-131 concentration varying from 1×10^{-9} to 1×10^{-10} $\mu Ci/cm^3$. Thus, the conservative nature of the assumptions used for the design-basis concentrations provide a value which agrees with this experience; i.e., 1.4×10^{-10} $\mu Ci/cc$.

Furthermore, plant personnel will be working in these areas for only a portion of their work year (i.e., < 10%). In the unusual event that airborne concentrations exceed or are likely to exceed the published DAC values, corrective action, controlled occupancy, engineering/process controls, or respiratory protective equipment is worn by station personnel as dictated by the TEDE ALARA evaluation.

Thus, when the occupancy factor and respiratory equipment protection factor are taken into account, station personnel are likely to receive only a fraction of the calculated design-bases thyroid inhalation dose.

12.4.3 Estimated Annual Dose at Restricted Area Boundary

12.4.3.1 Skyshine Doses

The maximum annual continuous exposure at the site boundary due to normal operation of LSCS, including anticipated operational occurrences, has been calculated to be less than 20 mrem at a distance of 500 meters. A large contributor to this dose is N-16 gamma radiation which penetrates the casing of turbine system components, emerges from the turbine building structure, and is scattered through the air to the dose point (skyshine).

This average annual skyshine dose is reduced to less than 4 mrem/yr for 100% occupancy at an average yearly fraction of full-power operation (80% plant availability) by the turbine shielding walls.

The offsite skyshine dose rate from a filled IRSF has been calculated to be less than 1 mrem/yr at the receiver identified for 40 CFR 190 purposes.

12.4.3.2 Cycled Condensate Storage Dose

The continuous exposure rate from two cycled condensate storage tanks (350,000 gallons each) is calculated to be less than 0.2 mrem/yr at a distance of 500 meters. The average annual site boundary dose from these tanks is calculated to be less than 0.05 mrem/yr after accounting for the occupancy factor (0.25).

12.4.4 Estimated Dose to Construction Workers

The estimated person-rem exposure to construction workers after both units are generating power are stated in Subsection 12.4.1.2.

The estimated person-rem exposure to construction workers due to testing and operating Unit 1 while constructing Unit 2 is analyzed in the following paragraphs.

During this period, facilities common to both units will be complete and operational. Only the facilities that are unique to Unit 2 will be under construction. The areas occupied by these incomplete facilities include portions of the reactor building, turbine building, and auxiliary building.

The onsite radiation sources considered for these areas are: radiation from the effluent cloud; scattered radiation from the Unit 1 turbine; and direction radiation from miscellaneous Unit 1 holding tanks, Unit 1 turbine, Unit 1 reactor, and associated piping. Since the system will be new and operating for a short period of time, it is assumed that cycled condensate storage tanks and stored radwaste will be negligible radiation sources.

During the construction period, a background dose rate of 100 mrem/year is assumed to apply to the entire construction crew. For a 40-hour workweek, this is approximately 6 mrem/quarter. The direct dose rate from the Unit 1 reactor, external to the Unit 1 reactor building, is expected to be less than the background dose rate. Therefore, this will present a negligible contribution to the dose received by the construction crew.

Because the Unit 1 and Unit 2 facilities are physically separate, most of the work on Unit 2 will not be performed next to Unit 1 operating equipment. There are some areas, however, where the equipment for the two units is shared. To account for this, and the possibility of rework and hook-up under adverse conditions, it is assumed that 25% of the work effort in the turbine and auxiliary buildings will take place in areas where the direct dose rate can be as high as 0.5 mrem/hr.

Based on calculations, the scattered radiation from the Unit 1 turbine will result in an onsite dose rate of less than 0.1 mrem/hr. For the purposes of calculating the person-rem dose from this source, it was assumed that half of the person-hours are spent in the areas of the plant shielded form turbine scattered radiation (i.e., inside the reactor building, auxiliary building, etc.).

The dose rate for gaseous radioactive effluents is calculated by use of the releases described in Regulatory Guide 1.112 (Nuclear Regulatory Commission 1976), and the conservative assumption that the release is at ground level. In calculating doses from the effluent cloud, no credit is taken for any shielding afforded to the work crew by the Unit 2 structures.

Current estimates indicate that approximately one million person-hours will be expended by the construction work force between fuel loading of Unit 1 and fuel loading of Unit 2.

For the sake of conservatism in calculating the total person-rem dosage, no credit was taken for Unit 1 availability. That is to say, all calculations were based on 100% power for the period between loading fuel into Unit 1 and loading fuel into Unit 2. The estimated doses from contained sources, gaseous effluents, and nominal background radiation are presented along with the total person-rem in Table 12.4-4. Table 12.4-5 gives the estimated person-hour breakdown per radiation level due to contained sources and skyshine.

12.4.5 <u>References</u>

1. Docket RM-50-2 Effluents from Light Water Nuclear Power Reactor, AEC Staff Exhibit #21, "Results of Independent Measurements of Radioactivity in Process Systems and Effluents at Boiling Water Reactor," USAEC, May 1973.

2. N. Eickelpasch, "Radioactivity and Radiation Protection in an LWR Installation," <u>Atom and Strom 18</u>, 122, 1972 (In German).

3. D.E. Harmer et al., "Developing a Solvent, Process Equipment, and Procedures for Decontaminating the Dresden-1 Reactor," 37th Annual Meeting of American Power Conference, April 21-23, 1975, The Dow Chemical Company, Midland, Michigan.

4. Commonwealth Edison Company, Docket 50-10, "Dresden-1 Chemical Cleaning Licensing Submittal," December 16, 1974.

TABLE 12.4-1
(SHEET 1 OF 2)

WORK FUNCTION EXPOSURE DATA FROM QUAD-CITIES

1973

Work Function	Number of People	Total Exposure (REM)	REM/ Person-year
Radiation Protection	10	17.53	1.75
Maintenance	69	52.43	0.76
Fuel Handling	10	8.05	0.81
Station Operation	85	46.81	0.55
Technical Staff	48	6.15	0.13
Management and Others	35	4.60	0.13
TOTAL	257	135.57	
Average REM/Person-year			0.53

1974

Radiation Protection	16	57.40	3.59
Maintenance	81	151.40	1.87
Fuel Handling	15	21.24	1.42
Station Operation	89	84.22	0.95
Technical Staff	63	18.12	0.29
Management and Others	45	9.02	0.20
TOTAL	309	341.76	
Average REM/Person-Year			1.11

TABLE 12.4-1
(SHEET 2 OF 2)

To 12/1/75*

Work Function	Number of People	Total Exposure (REM)	REM/ Person-year
Radiation Protection	17	73.71	4.34
Maintenance	87	224.33	2.58
Fuel Handling	16	25.85	1.62
Station Operation	88	132.85	1.51
Technical	74	56.32	0.76
Management and Others	50	14.68	0.29
TOTAL	332	527.74	

* Average REM/Person-11 months --------------------------------1.59
  Average REM/Person-Year --1.73

TABLE 12.4-2

EXPECTED ANNUAL PERSON-REM BASED ON OPERATING BWR STATIONS

STATION	YEAR	PERSON-REM FOR STATION PERSONNEL	REM/PERSON-YEAR FOR STATION PERSONNEL	PERSON-REM FOR CONTRACTORS AND TEMPORARY WORKERS
Nine-Mile Point	1970	26	0.38	36
	1971	106	1.56	100
	1972	218	1.68	88
	1973	310	2.28	284
Oyster Creek	1969	7	0.10	6
	1970	48	0.55	16
	1971	140	1.43	109
	1972	399	3.69	252
	1973	551	3.88	898
Dresden	1970	127	0.63	16
	1971	315	1.40	400
	1972	368	1.54	360
	1973	576	2.41	333
	1974	801	2.99	818
Quad-Cities	1972	26	0.07	38
	1973	136	0.53	59
	1974	342	1.11	101
	1975*	528	1.50	1166
Monticello	1971	27	0.33	2
	1972	63	0.76	2
	1973	91	0.87	65
Millstone	1971	31	0.13	18
	1972	255	1.10	340
	1973	225	1.28	395
Average		241	1.35	250

LSCS Station Estimated Annual Person-Rem	
Station personnel (250 x 1.4 REM)	350
Contractors (including temporary workers) (1000 x 0.3 REM)	300
Total	650

* Data supplied to Nov. 30, 1975, the data was multiplied by 12/11 before averaging was performed.

TABLE 12.4-3

ANNUAL THYROID DOSES RESULTING FROM CALCULATED DESIGN-BASIS AIRBORNE CONCENTRATION IN REMS/YR

ISOTOPE	REACTOR BUILDING	TURBINE BUILDING	RADWASTE BUILDING
I-131	[redacted]		
I-132			
I-133			
I-134			
I-135			

The above dose rates are based on 13.3 hr/wk exposure of personnel in each building, of which 50% is spent in clean areas, 35% in general areas with potential airborne, 10% in pump room and valve aisle, and 5% in radiation areas.

TABLE 12.4-4

ESTIMATED RADIATION DOSES IN PERSON-REM TO THE CONSTRUCTION WORK FORCE AFTER UNIT 1 FUEL LOADING

TOTAL PERSON-HOURS	1 X 10^6 person-hours
DOSE TO THYROID	3.9 person-rem
WHOLE BODY DOSE	
From effluent cloud	3.9 person-rem
From contained sources*	125 person-rem**
From natural background	11.3 person-rem

* Includes shyshine dose.

** A location breakdown is given in Table 12.4-5.

TABLE 12.4-5

ESTIMATED RADIATION DOSES TO CONSTRUCTION WORKERS[*]

LOCATION (UNIT 2)	PERSON-HOURS	AVERAGE DOSE RATE (mrem/hr)	PERSON-REM
Reactor Building	350,000	0.01	3.5
Turbine Building[**]	300,000	0.10	30.0
Turbine Building[**]	100,000	0.60	60.0
Auxiliary Building	40,000	0.50	20.0
Auxiliary and Service Buildings	110,000	0.01	1.1
Grounds**	100,000	<0.10	<10
			<124.6

[*] From loading fuel into Unit 1 to startup of Unit 2.

[**] Includes the direct dose from contained sources plus skyshine.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 Organization

The administrative organization of the health physics program and personnel responsibilities are referenced in Subsections 12.1.1.1 and 12.1.1.2.

The experience and qualification of all station personnel are given in Subsection 13.1.3.

12.5.2 Equipment, Instrumentation, Facilities

Table 12.5-1 lists the normal storage location of respiratory protective equipment, protective clothing, and portable and laboratory technical equipment and instrumentation.

Detectors and monitors of the appropriate quantity, sensitivity, and range for the measurement of gamma, beta, alpha, and neutron radiation shall be available based on current Health Physics practices. The frequency and methods of calibration are described in the applicable procedures. Examples of types of Health Physics and Chemistry Equipment include: Spectrometer Multichannel Analyzer, Gas Flow Proportional Counter, Electronic Dosimeter, Area Radiation Monitor (ARM), Air Ion Chamber, Dose Rate Meter, High Range GM Dose Rate Meter, Neutron Monitoring Equipment, Whole Body Contamination Monitor, Personnel Portal Monitor, GM Survey Count Rate Instrument, Alpha Monitoring Equipment, Small Articles Monitor, Air Sampler, and Continuous Air Monitor.

Health physics and radiochemist facilities are described in Table 12.5-2.

12.5.3 Procedures

Radiation control procedures shall be maintained, made available to all Station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

The health physics procedures have been developed to implement EGC's commitment to "As Low As Reasonably Achievable" (ALARA) as stated in Subsection 12.1.1.

The Health Physicists continuously review the effectiveness of the ALARA program as implemented by the Health Physics Program (Section 12.5). Review of health physics procedures is performed in accordance with Section 13.5.

12.5.3.1 Administrative Program

Strict administrative control of radiation exposure includes those methods described in Subsections 12.5.3.2, 12.5.3.3, and 12.5.3.5. Other administrative controls used included locked high radiation areas greater than 1 rem/hr, radiation worker permits, timekeeping of personnel in high radiation areas, where applicable, and security measures including escorts for visitors within the plant security area.

12.5.3.2 Personnel External Exposure Program

The personnel external exposure program consists of multiple methods of reviewing external radiation levels and controls within the plant. These provide plant and personnel status information required to maintain an ALARA program (Subsection 12.1.1).

Area radiation monitors (ARM's) are located throughout the plant and provide general area indication of gamma radiation levels. These levels are continuously monitored and are alarmed in the control room. Some monitors also have local indication and alarm at certain in-plant locations. Process radiation monitors with control room indication and alarms also provide for immediate recognition of significant increases in in-plant dose rate levels.

Routine beta-gamma dose rate evaluations are made on at least an annual basis for general access areas of the plant. This provides detailed dose rate information for normal in-plant exposure evaluation. The survey sheets are reviewed to note unusual values for determination of additional controls that may be required due to new or increased radiation dose rates.

Special beta-gamma dose rate surveys are made on an as needed basis for jobs that take place in normally inaccessible (i.e., high radiation) areas. These areas are not normally surveyed on a routine basis due to the required dose commitment being inconsistent with the ALARA program. Continuous or intermittent surveys are provided on an as needed basis as determined by radiation protection for radiation work permits (Subsection 12.5.3.1).

Personnel entering controlled radiation areas onsite are provided with personnel radiation monitoring devices in accordance with 10 CFR 20.1502 to measure their radiation exposure. These devices consist of DLR (Dosimeter of Legal Record), electronic dosimeters, or other suitable devices. Daily, the electronic dosimeters readings (or equivalent) and timekeeping results (if applicable) are normally recorded, and are routinely reviewed by radiation protection management and by management in the individual's work group, if applicable. DLR (or equivalent) are changed at the frequency specified by the Radiation Protection Manager. DLR results are entered in the EGC computerized radiation exposure records system. These official and permanent records furnish the exposure data for the administrative control of radiation exposure. Required reports are made by radiation protection management through the use of this records system.

General area neutron dose rate measurements are made during startup after initial fuel loading and following refueling outages to verify neutron dose rates. Special neutron surveys and neutron dosimetry is used when entrance is made into neutron areas as required by 10 CFR 20 (i.e., inside the drywell during power operation). The LSCS procedures will conform to Regulatory Guide 8.14, Rev. 1, August 1977, "Personnel Neutron Dosimeters."

Radioactive materials and special nuclear materials are handled and stored under the direction of personnel as specified in Subsection 12.1.1.2.

12.5.3.3 Personnel Internal Exposure Program

The personnel internal exposure program consists of multiple methods of reviewing airborne radiation levels and controls within the plant. These provide plant and personnel status information required to maintain an ALARA program (Subsection 12.1.1).

The station vent stack gas monitor provides control room indication and an alarm of high level noble gas activity from plant ventilation and of process air (i.e., off-gas and gland seal systems).

The station vent stack particulate and charcoal continuous air sample provides radioactive particulate and radioactive iodine concentrations for plant ventilation and process air. These samples are taken as specified in the ODCM.

The gross gamma reactor building process monitors provide control room indication and alarm of radiation level from each reactor building ventilation exhaust duct.

Continuous air monitors containing particulate and charcoal filters with local readout and alarm normally monitor each turbine building ventilation exhaust, each reactor building ventilation exhaust, each drywell, the radwaste building ventilation exhaust, the off-gas filter building ventilation exhaust, and the refueling floor.

The turbine building air manifold system sample results are periodically reviewed by health physics personnel. High volume particulate and charcoal grab samples are normally taken periodically in accessible areas of the plant. Special samples are taken as required by radiation protection for radiation work permits and other jobs as deemed necessary. These results are reviewed by health physics personnel and are used to determine respiratory protective equipment requirements per the TEDE ALARA evaluation in accordance with the radiation protection program.

A personnel bioassay program is administered by the health physicists. Bioassays (in vivo measurements and/or measurements of radioactive material in excreta) are conducted as necessary to aid in determining the extent of an individual's internal exposure to concentrations of radioactive material.

A permanent onsite whole body counter with an associated multichannel analyzer system. This equipment is operated by site radiation protection personnel.

Measurement of radioactive material in excreta is normally performed by an independent contractor.

All results of bioassay determinations are recorded as required.

12.5.3.3.1 Personnel Internal Exposure Program During Accident Conditions

Particulate, iodine, and noble gas air monitoring is provided for air sampling plant areas where personnel may be present during accident conditions. Grab samples are used as an alternate means of sampling.

Grab samples are obtained using the equipment specified in LSCS-UFSAR, Section 12.5.2. During accident conditions, Silver Zeolite cartridges will be used for radioiodine analysis in accordance with Corporate and/or Station Emergency Plan Implementing Procedures and analyzed with a gamma spectrometer multichannel analyzer system with a germanium detector.

The analysis of iodine cartridges will be performed in a low background low contamination area. During accident conditions, an area such as the low storeroom elevation of the service building or the radwaste control room can be used for this purpose. The post-accident radiation levels in these areas are projected to be at normal shutdown levels.

Station procedures are provided for obtaining and evaluating both routine and non-routine air samples. In addition to initial training provided for radiation protection personnel, periodic drills are conducted in accordance with Generating Stations Emergency Plan (GSEP) Section 8.3 (refer to Section 13.3).

12.5.3.4 Contamination Control Program

The contamination control program consists of multiple methods of controlling the spread of contamination to personnel and equipment within the plant.

Routine smear surveys are periodically made of normally accessible areas of the plant and are recorded on survey sheets. These results are reviewed by health physics personnel. Special smear surveys are performed on an as needed basis for radiation work permits and for unconditional release of equipment, tools, and materials being removed from radiation posted areas. Items which are contaminated are required to be decontaminated to within release limits or packaged and tagged in accordance with the station radiation protection program.

Personnel working in areas contaminated with radioactive materials are required to wear protective equipment as specified by radiation protection. Workers in significantly contaminated areas are required to check out with a G-M type instrument or portal monitor following removal of protective clothing. In addition, all personnel leaving potentially contaminated areas (including controlled areas of the reactor and turbine buildings) are required to check out with a G-M type instrument or portal monitor. A final check out with a scintillation type instrument (normally a portal monitor) is typically required of all plant personnel prior to leaving the site.

12.5.3.5 Training Program

The radiation protection training programs are described in Section 13.2. This program covers the following:

a. general employee health physics,
b. general employee respiratory protection,
c. contractor health physics,
d. contractor respiratory protection,
e. general employee retraining,
f. radiation protection technician training, and
g. radiation protection technician retraining.

All personnel must understand how radiation protection relates to their jobs and have reasonable opportunities to discuss radiation protection safety with a member of the radiation protection department whenever the need arises. Personnel are made aware of EGC's commitment to keep occupational radiation exposure as low as reasonably achievable (Subsection 12.1.1). A minimum goal of this program is that workers shall be sufficiently familiar with this commitment that they can explain what the management commitment is, what "As Low As Reasonably Achievable" means, why it is recommended, and how they have been advised to implement it on their jobs.

Qualifications of personnel, including training requirements for radiation protection personnel are described in Subsection 12.5.1.

12.5.3.6 In Plant Radiation Monitoring

This program provides controls which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

- Training of personnel
- Procedures for monitoring, and
- Provisions for maintenance of sampling and analysis equipment

TABLE 12.5-1

STORAGE LOCATION OF EQUIPMENT

EQUIPMENT	NORMAL STORAGE LOCATION
Self-Contained Breathing Apparatus (Pressure Demand)	Various locations in the plant
Protective Clothing	[redacted]
β - γ Air Ionization Chambers	
G - M Survey Instruments	
Neutron Detector	
Gamma Spectrometer	
α β Sample Changer	
Chemical Analysis Equipment	
Respiratory Protection Equipment	

TABLE 12.5-2

HEALTH PHYSICS AND CHEMISTRY FACILITIES

NAME	LOCATION	PRIMARY FUNCTION
Calibration Facility	[redacted]	Calibration of Gamma Dose Rate Instruments, Storage of Routine Survey Instruments
Hot Laboratory	[redacted]	Chemical Analysis and Radiochemical Separations
Cold Laboratory	[redacted]	Chemical Analysis
Sample Preparation Room	[redacted]	Standards Preparation and Special Projects, Chemical Analysis
Supply Room	[redacted]	Storage of Chemicals, Glassware and Laboratory Equipment
Counting Room	[redacted]	Radioactivity and Radiological Determination of Samples
Laundry	[redacted]	Storage of Protective Radiological Clothing
Mask Issue and Cleaning Facility	[redacted]	Cleaning, Inspection, and Storage of Respiratory Protective Equipment
Radiation Protection Office	[redacted]	Location of Survey Information

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 1 of 11)

[redacted]

LEGEND FOR RADIATION ZONES

Area	Maximum Dose Rate (mr/hr)
	>100
	100
	15
	2.5
	1.0
	< 0.5 Normal Operation < 5000 Post-Design-Basis Accident

LEGEND: SEE M-4

GRAPHIC SCALE

NORTH

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 2 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 3 of 11)

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS
REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION
(SHEET 4 OF 11)

REVISION 13

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 5 of 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 6 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION
(SHEET 7 of 11)

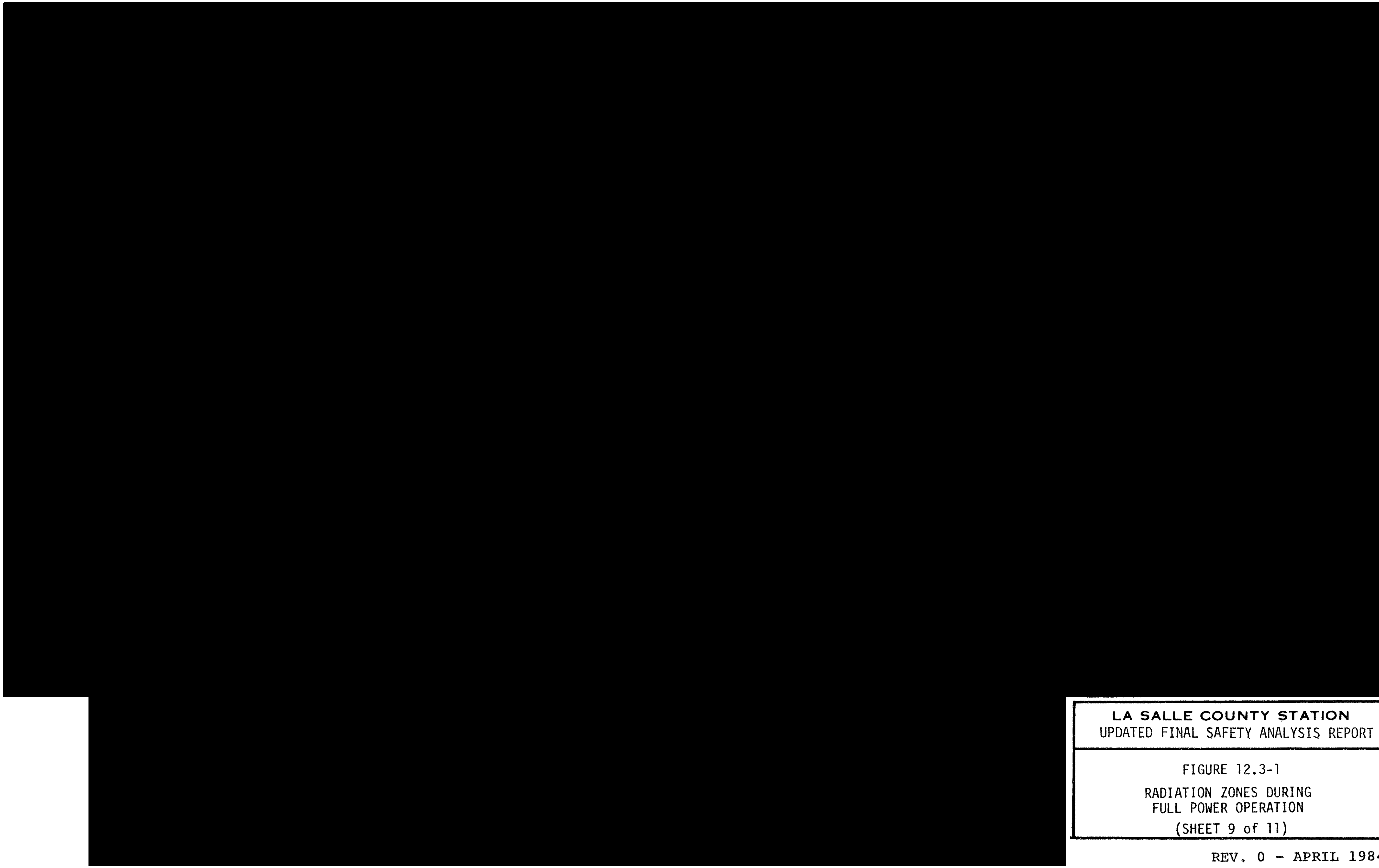
[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 8 of 11)



LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION
(SHEET 9 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 10 of 11)

REV. 0 - APRIL 1984

[redacted]

REV. 14

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS
REPORT

FIGURE 12.3-1

RADIATION ZONES DURING
FULL POWER OPERATION

(SHEET 11 OF 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN

(SHEET 1 of 11)

[redacted]

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN
(SHEET 2 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN
(SHEET 3 of 11)

REV. 0 - APRIL 1984

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN
(SHEET 4 OF 11)

REVISION 13

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN
(SHEET 5 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN

(SHEET 6 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN

(SHEET 7 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN
(SHEET 8 of 11)

REV. 3 - APRIL 1987

[redacted]

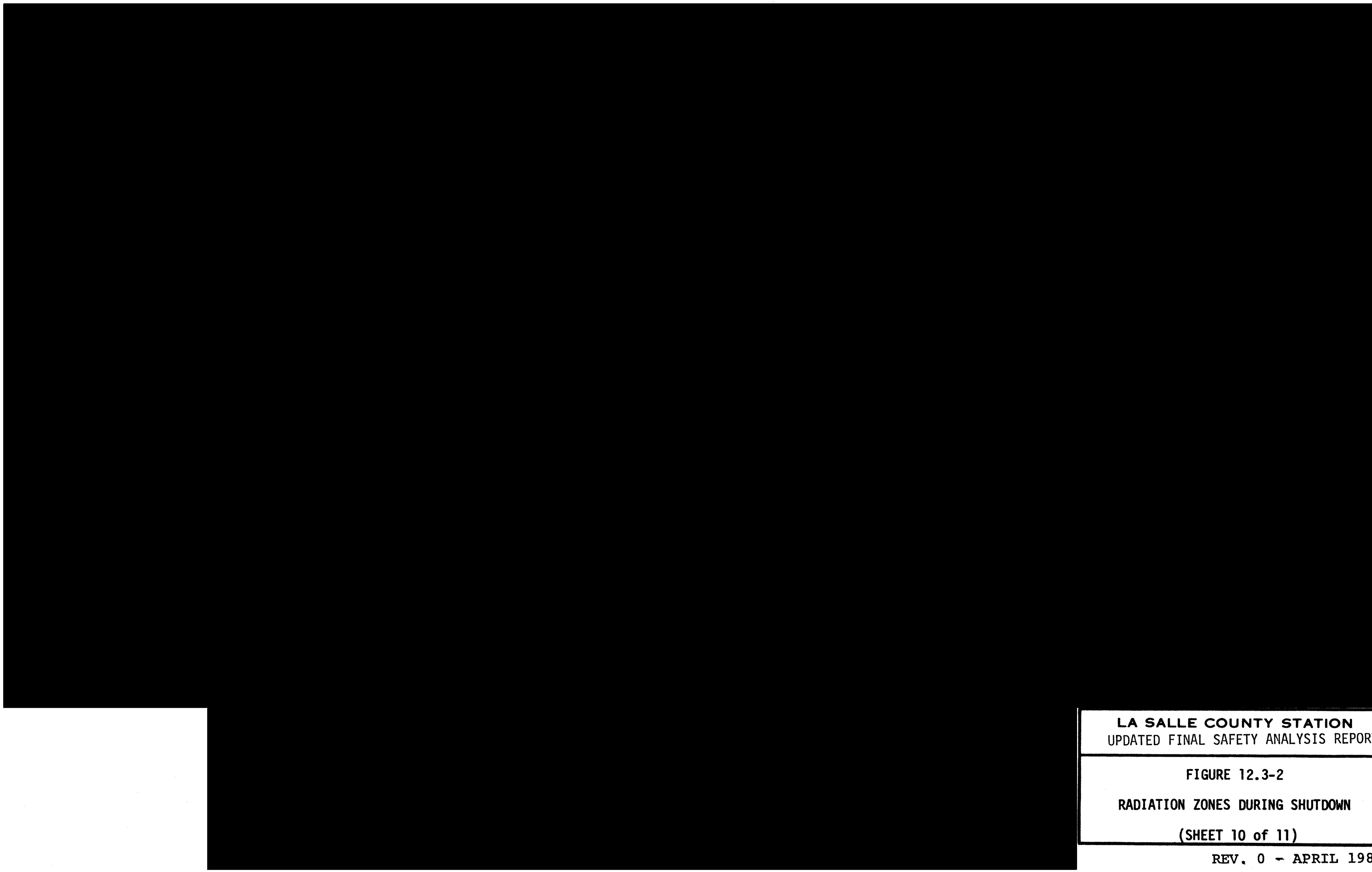
LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN

(SHEET 9 of 11)

REV. 0 - APRIL 1984



LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN

(SHEET 10 of 11)

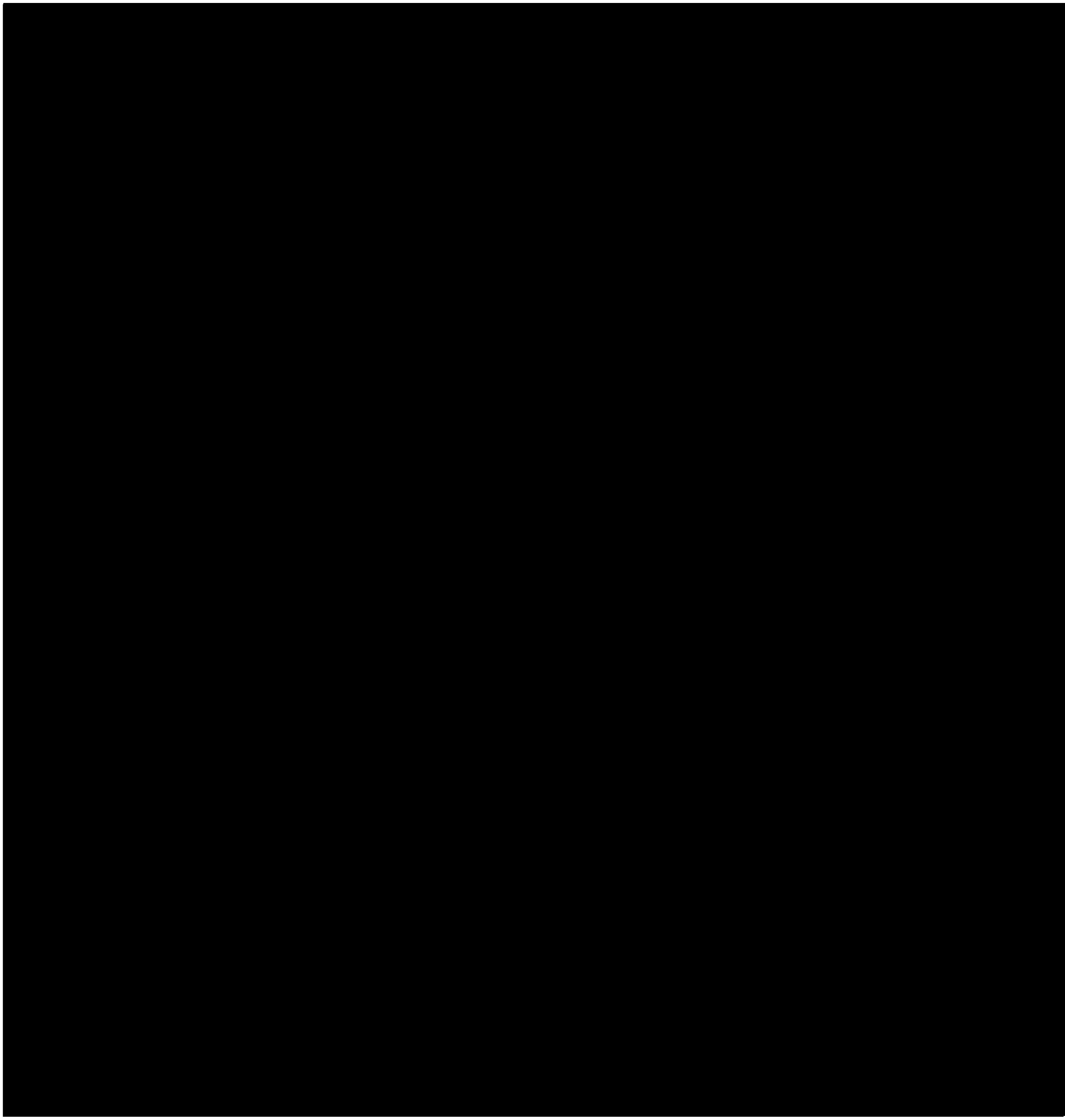
REV. 0 - APRIL 1984

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS
REPORT

FIGURE 12.3-2

RADIATION ZONES DURING SHUTDOWN
(SHEET 11 OF 11)



LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 1 of 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 2 of 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 3 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 4 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 5 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 6 of 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 7 of 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 8 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 9 of 11)

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 10 of 11)

REV. 0 - APRIL 1984

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-3

RADIATION PROTECTION DESIGN FEATURES

(SHEET 11 of 11)

[redacted]

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-4

POSTACCIDENT ESSENTIAL AREAS
(SHEET 1 of 6)

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-4

POSTACCIDENT ESSENTIAL AREAS
(SHEET 2 OF 6)

REVISION 13

[redacted]

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-4

POSTACCIDENT ESSENTIAL AREAS

(SHEET 3 of 6)

8 7 6 5 4 3 2 1

F

E

D

C

B

A

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS
REPORT

FIGURE 12.3-4

POSTACCIDENT ESSENTIAL AREAS
(SHEET 4 OF 6)

8 7 6 5 4 3 2 1

[redacted]

NORTH

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 12.3-4

POSTACCIDENT ESSENTIAL AREAS
(SHEET 5 of 6)

[redacted]

LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS
REPORT

FIGURE 12.3-4

POSTACCIDENT ESSENTIAL AREAS
(SHEET 6 OF 6)