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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.

DRAWING*	SUBJECT
M-1	Property Plan
M-7	General Arrangement, Sections "A-A" and "B-B"
M-8	General Arrangement, Sections "C-C" and "D-D"

12.0 RADIATION PROTECTION

The protection of operating personnel from radiation emanating from process equipment, from radioactive materials present on equipment externals in work areas, or from airborne radioactive material particles and gases is accomplished by a combination of the design of the facility's shielding structures, selection and use of appropriate radiation monitoring instrumentation, and the development and implementation of control standards and procedures. The purpose of the following sections is to provide a brief summary of these radiation protection aspects of these units. The shielding and instrumentation are described in greater detail in Sections 11.5, 12.3, and 12.5. A study contained in Appendix 12A details dose rates throughout the plant following a postulated accident.

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

12.1.1 Policy Considerations

This subsection addresses the management policy and organizational structure related to implementation of the "as low as reasonably achievable" (ALARA) program to ensure that occupational exposure for station personnel are maintained ALARA. The ALARA program is part of the station radiation protection program.

12.1.1.1 Management Policy

It is the policy of EGC to maintain the occupational dose equivalents to the individual and the sum of dose equivalents received by all exposed workers to levels that are as low as reasonably achievable (ALARA). This ALARA philosophy is implemented in a manner consistent with station operating, maintenance, and modification requirements, taking into account the state of technology, the economics of improvements in relation to the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

EGC's commitment to this policy is reflected in the EGC ALARA program, in station design, in careful preparation and review of station radiation protection operating and maintenance procedures, and in review of equipment design to incorporate the results of operating experience.

It is the policy of EGC to have all levels of management strongly committed to radiation protection and, specifically, to maintain occupational radiation exposures ALARA. Also, it is recognized that each worker must take personal responsibility for actions necessary to implement successful dose reduction measures.

12.1.1.2 Organizational Structure

The Station ALARA Committee functions as the executive body for the station's ALARA program. The committee is responsible for developing the ALARA goals for the station. The committee provides guidance and recommendations on aspects of radiological protection operations thus ensuring that effective radiation dose reduction measures are applied.

12.1.2 Design Considerations

The initial licensing of Dresden Station predated issuance of 10 CFR 50, Appendix I. Therefore, specific ALARA design considerations were not developed as part of the PSAR or FSAR. The purpose of these considerations is to ensure that design of the plant facilities contributes to keeping occupational exposures ALARA. The ALARA design considerations for plant modifications are addressed at an early stage of the modification planning process as part of the ongoing ALARA program.

The design of modifications is reviewed for their exposure impact. A preliminary review of the exposure impact of the modification to be performed at the station is completed by the responsible engineer in the planning phase shortly after a request to do the modification is submitted. This review is required by engineering procedures. Additional guidance is provided by the ALARA design guide, which includes a detailed compilation of ALARA considerations. The design guide is not a requirement for approval of the modification. Some items covered in the design guide include the following:

A. Layout and radiological boundaries;

B. Activation product concerns;

C. Contamination control;

D. Operability and maintenance concerns;

E. Shielding;

F. Equipment types; and

G. Possible use of robots to minimize personnel exposure.

Station ALARA personnel attend a series of meetings during the planning phases of the modifications to provide information on the exposure impact of the modification. This meeting is the first review of the modification and offers the opportunity to address all ALARA concerns before the procedural process for approval of the modification begins. Written ALARA Action Reviews are required when performing radiation work which has the potential for high cumulative exposures. The ALARA personnel also participate in the walkdown phase of modification planning and in the post-job review process.

12.1.3 Operational Considerations

Plant procedures are required to comply with the ALARA program. Procedures provide the necessary guidance for assuring the radiation protection program addresses ALARA considerations. Aside from procedures which directly implement the ALARA program such as ALARA Action Reviews, other procedures are also associated with the ALARA program. Examples are procedures for exposure review and authorization; respiratory protection; high radiation area control; contamination control; and radiation protection training. The radiation work permit (RWP) procedure also contributes significantly to ALARA. The ALARA actions are determined based on the RWP information and station work activities.

12.2 RADIATION SOURCES

The initial licensing of Dresden Station predated issuance of Regulatory Guide 1.70, Revision 3. Therefore, the identification of radiation sources (beyond those in radioactive waste management systems) was not developed as part of the PSAR or FSAR. The purpose of identifying radiation sources is to permit evaluation of radiation protection design features (described in Section 12.3) in order to provide reasonable assurance that radiation exposure of plant personnel will be within allowable limits. Radiation surveys within the plant are made and evaluated as part of the ongoing as low as reasonably achievable (ALARA) program at Dresden.

Radiation sources generated in the reactor core and transported by the reactor coolant system are described in Section 11.1. Radiation sources from components of the radioactive waste management systems are described in Sections 11.2, 11.3, and 11.4.

Radioactivity is also present in the fuel pool cooling and cleanup systems described in Section 9.1.3, the reactor water cleanup system described in Section 5.4.8, and the condensate cleanup system described in Section 10.4.6.

A complete and current inventory of radioactive sources (i.e., instrument calibration sources, etc.) at the station is maintained. Sealed radiation sources are tested for removable contamination according to Technical Specification requirements. Storage, handling, and use of sealed sources are controlled by plant procedures.

12.3 RADIATION PROTECTION DESIGN FEATURES

This section describes plant design features used to ensure that occupational radiation exposures resulting from the radiation sources within the plant meet the "as low as reasonably achievable" (ALARA) program objectives described in Section 12.1. These features include shielding, ventilation systems, and radiation monitoring instruments.

12.3.1 Facility Design Features

When Dresden Units 2 and 3 were designed, the structures were shaped and arranged on the site to conform to the locations of the previously existing plant (Unit 1), water supply, roads, and railroad. The structures were also arranged to provide the best layout for the equipment. Safety requirements also were met with respect to circulation of contaminants and protection from radiation. Drawing M-1 is a plot plan showing the arrangements of the structures.

Additional information on the design features of Dresden Station that protect personnel from radiation exposure and minimize radiation damage to plant equipment can be found in the following:

A. Sections 11.1 through 11.4 describe radioactive waste processing;

B. Section 11.5 addresses process and effluent radiation monitoring;

C. Section 12.5.3.4 addresses control of access to radiation areas; and

D. Section 12.5.2 describes the radiochemical laboratory facilities.

12.3.2 Shielding

Normal operating conditions determine the major portion of the shielding requirements. Two notable exceptions are the control room, where shielding is determined by the radiation levels produced during the loss-of-coolant accident (LOCA), and the shutdown cooling system, where shielding is determined by shutdown conditions.

12.3.2.1 Design Basis

The basis for the design of the radiation shielding is in compliance with the requirements of 10 CFR 20, which describes the limits of occupational radiation exposures. Compliance with these regulations is achieved in part through shield design, which is based upon occupancy requirements in various areas. A list of generalized occupancy requirements and attendant radiation dose rates are presented in Table 12.3-1. The duration of expected operating personnel occupancy in various areas of each unit was obtained from experience during operation of Dresden Unit 1 and other similar nuclear-powered units.

Radiation areas with dose rates in excess of those listed in Table 12.3-1 were designed to be entered on a controlled time basis. Radiation areas with dose rates in excess of 100 mrem/hr at 30cm from the radiation source are equipped with barriers and administrative controls which prohibit unauthorized entry.

The primary objective of the radiation shielding is to protect personnel against radiation emanating from the reactor, turbine, and their auxiliary systems. Supplemental procedures to control access to radiation areas (see Section 12.5) and to control personnel exposure serve to limit radiation exposure to acceptable levels.
The secondary objective of the radiation shielding is to limit radiation damage to operating equipment. Specific fabrication materials are given individual consideration. Of principal concern are organic materials used in the equipment, such as insulation, rubber tank linings, and gaskets.

In general, it is sufficient to limit the radiation exposure to 10^6 rads for materials of concern over the expected service life of the equipment or of individual parts. For certain materials the exposure must be less or can be greater without significantly affecting serviceability: e.g., a limit of 10^4 rads for teflon and about 10^8 rads for polyurethane.

The shielding materials required to meet the preceding objectives are primarily concrete, water, and steel. High-density concrete, lead, and neutron absorption material may be used as alternates in special applications. The original estimated design dose rate in most areas outside of the drywell in the reactor building was 1 mrem/hr. Consequently, the drywell and its contents are shielded so that most areas outside the drywell and outside the pressure suppression chamber are accessible. Actual dose rate increases resulting from buildup of activated erosion and corrosion products ("crud") are evaluated as part of the station radiation protection program.

12.3.2.1.1 Control Room

The dose in the control room is limited to 0.5 rem in any 8-hour period following a design basis accident in either Unit 2 or Unit 3. During normal operation of either Unit 2 or Unit 3, the total shielding provided for the control room is sufficient to limit the transmission of radiation from the reactor building to less than 0.1% of the above limit. A discussion of control room habitability is contained in Section 6.4.

12.3.2.1.2 Reactor Building

The following regions within the reactor building but outside the primary containment have design dose rates exceeding 1 mrem/hr:

A. Fuel storage pool;

B. Reactor water cleanup (RWCU) system;

C. Shutdown cooling system;

D. The operating floor directly over the drywell shielding plugs above the reactor vessel;

E. The region housing the suppression chamber of the pressure suppression system;

F. Miscellaneous equipment, such as the fuel pool heat exchanger and pumps;

G. High pressure coolant injection (HPCI) system;

H. Building crane cab;

I. Isolation condenser valve rooms;

J. Traversing incore probe (TIP) system; and

K. Control rod drive (CRD) accumulator banks.

12.3.2.1.3 Turbine Building

The major radiation source in the turbine building is N-16. Shielding is provided for the following regions to which access normally is not permitted during full power operation:

A. Main condenser - hotwell;

B. Feedwater heaters (except at tube sheet ends of low-pressure heaters);

C. Air ejector and the gland seal exhauster;

D. Off-gas recombiner and preheater;

E. Condensate demineralizer tanks and associated equipment;

F. Steam piping - moisture separators and high-pressure heaters; and

G. Steam turbine.

H. Condensate prefilter vessels and associated equipment (access permitted up to local shielding during full power operation),

I. 100% Condensate Filtration System.

Most of the turbine operating floor is accessible, and the building crane can be operated remotely.

When the hydrogen injection system is operated, the resulting increase in N-16 activity can cause radiation levels in the turbine building to increase. Radiological effects of hydrogen injection are described in Section 11.3.3.3.

12.3.2.1.4 Off-Gas System

The shielding for the off-gas air ejector is based upon N-16 and the noble gases as principal radiation sources. The noble gas component of the combined radiation source is based upon an average annual release rate of 0.7 Ci/s. Shielding for the off-gas filters is based upon the accumulation of particulate radionuclides that are produced by the decay of the noble gases during a 30-minute holdup time. The off-gas system is described in Section 11.3.

Actual holdup time is closer to 7 hours following installation of the modified off-gas system. The amount of off-gas flow leaving the steam jet air ejectors is reduced significantly after passing through a catalytic recombiner which recombines free H_2 and O_2 produced by the radiolytic decomposition of water inside the reactor, therefore taking longer to traverse the holdup volume.

12.3.2.1.5 Radwaste Building

The radwaste building shielding is designed to limit the dose rate in the building control room to approximately 1 mrem/hr. Regions where pumps and valves are located have design radiation levels of 6 to 12 mrem/hr. The solid waste preparation area is shielded to a design dose rate of 1 mrem/hr.

Ample shielding has been provided in the radioactive waste control systems design to maintain personnel exposure well below established limits. Sumps, tanks and other high-activity vessels are housed in limited-access areas or concrete cells. Piping which would contribute significantly to radiation dose rates is shielded or not run in normally frequented areas.

12.3.2.2 Description

This subsection describes the radiation shielding for the reactor vessel, the drywell, the shutdown cooling system components, the control room, the turbine and main steam system, the condensate demineralizers, and the radwaste systems.

12.3.2.2.1 Reactor Shield Wall

Within the drywell, an annular shield wall of concrete is provided between the reactor vessel and the drywell walls to limit gamma heating in the drywell concrete, provide shielding for access in the drywell during shutdown, and limit activation of drywell materials by neutrons from the core.

The reactor shield wall consists of a hollow cylinder of ordinary concrete having a 2-foot thick wall and circumscribing the reactor vessel. The inside and outside surfaces of the reactor shield wall concrete are formed with steel plate which is increased in thickness for extra shielding at the elevation of the core. Reinforcing steel is used in the concrete to give structural strength. The cylinder is supported

on the same structural concrete that supports the reactor vessel. This shield is cooled on both surfaces by circulating air from the drywell cooling system.

The pipes leaving the vessel at elevations below the top of the shield wall penetrate the wall. The penetrations in the vicinity of the core utilize removable shield plugs which fit around the penetrating pipe. The plugs are provided in order to allow access to the pipe welds for purposes of inservice inspections. The Dresden plugs are two 9-inch thick steel plates attached to the shield wall by two 1-7/16-inch diameter vertical hinges, with both halves locked in place by a 1-7/8-inch diameter locking pin. Recirculation piping penetrates this annular shield wall around the reactor vessel. Streaming through these penetrations by radiation from the core is limited by shielding located within the reactor vessel. These penetrations are also provided with removable shielding sections at the annular shield so that access is available for inspection of the connections of recirculation piping to the reactor vessel. The region that houses the control rod drives is shielded against radiation from the recirculation piping. This piping constitutes a radiation source during shutdown as a result of crud buildup.

During reactor operation, the reactor shield wall serves as a thermal shield to protect the containment shield wall outside the drywell from thermal damage. During shutdown, this shield also serves to protect personnel in the drywell from the gamma radiation from the core and the reactor vessel.

12.3.2.2.2 Containment Shield Wall

The primary containment vessel for each reactor is enclosed completely in a reinforced concrete structure (an integral part of the reactor building) having a variable thickness of from 4 to 6 feet. This structure is called the containment shield wall. See Drawings M-7 and M-8. In addition to serving as the basic biological shielding for the containment system, this concrete structure also provides a major mechanical barrier for the protection of the containment vessel and the reactor system against potential missiles generated external to the primary containment. It also serves as a backup for the steel drywell wall in resisting jet forces. Additional information on missile protection is contained in Section 3.5. Jet forces and other effects of pipe breaks are described in Section 3.6.

Bedrock is used as the main support for the concrete containment shield wall which is structurally designed to handle the loads of floors, equipment, and the higher elevations of the shield itself. Reinforcing steel is used to maintain structural integrity under the design basis accident and seismic loading.

Penetrations through the concrete containment shield wall are designed so that they are not aimed directly at the core or major items of equipment in the drywell. In addition, they are either terminated in shielded cubicles or are shielded with steel flanges to reduce radiation levels in accessible areas.

12.3.2.2.3 Shutdown Cooling System

The heat exchangers and pumps of the shutdown cooling system are located in separate shielded cubicles. Gamma radiation from the equipment in these cubicles is reduced by the shield walls to a design dose rate of about 2 mrem/hr or less in the adjacent accessible areas at the time the system is placed in operation.

12.3.2.2.4 Control Room

The shielding for the control room consists of poured-in-place reinforced concrete. The floor and ceiling slabs are 6-inch thick ordinary concrete; whereas, the walls range in thickness from 18 inches of ordinary concrete to 27 inches of magnetite concrete which was used because of space limitations.

Advantage is taken of the shadow shielding offered by other structures to reduce shielding thicknesses and to locate penetrations (see Figures 12.3-1 through 12.3-5). Ordinary concrete shielding for the battery and computer rooms complements the floor and ceiling slabs of the control room.

The shielding factor is based on the expected attenuation of the concrete walls. The control room is designed to provide a minimum of 3 feet of effective concrete shielding. This shielding includes an 18-inch concrete control room wall and an 18- to 36-inch concrete reactor building wall. In addition, the angle of shine is oblique to the control room walls, thus providing greater than 3 feet of effective concrete shielding. Using an average energy of 1.0 MeV, the attenuation through 3 feet of concrete, including buildup, is 3.1×10^{-5}. The peak radiation level in the control room following a LOCA in Unit 2 or 3 is 23 to 30 mrem/hr. This peak value occurs 3 days after the design basis accident.

12.3.2.2.5 Turbine Steam Handling Equipment

The steam handling equipment associated with the turbine-generator unit is shielded with concrete to reduce the radiation levels in accessible areas, (design radiation levels for steam handling equipment are shown in Table 12.3-2). When the hydrogen injection system is operated, the resulting increase in N-16 activity can cause the radiation levels in the turbine building to increase beyond these design levels. Removable shield walls are used where existing shielding is inadequate. As an example, at the end of the Unit 2 high-pressure turbine, a gap between the 3-inch steel shield and the 30-inch concrete wall permits a 60-mrem/hr field to exist. A water shield reduces the dose rate to 10 mrem/hr. Lead blankets are also used to shield steam sample lines, where appropriate.

12.3.2.2.6 Condensate Demineralizer System

For each reactor the demineralizer vessels are located in two cubicles, four service units in one cubicle and three service units in the other cubicle, separated by an operating aisle. The associated tanks are located in one adjoining cubicle.

The radiation penetrating the concrete shielding of these cubicles is designed to be less than 1 mrem/hr, exclusive of radiation streaming. Recycle pumps, valves, some piping, and instrumentation associated with the demineralizer vessels are located in the operating aisle. Piping carrying condensate or demineralized condensate does not require shielding.

12.3.2.2.7 Reactor Water Cleanup System

Most of the reactor water cleanup equipment is shielded with concrete. The bases for shielding design were determined by the estimated frequency of operating, inspecting and maintaining the various equipment and its devices. The shielding is designed to reduce the radiation levels in the valve corridors to 30 mrem/hr or less and the levels in the access corridors around the cleanup system complex to 5 mrem/hr or less. The unshielded equipment is located in controlled access areas.

12.3.2.3 Performance Analysis

The design basis accidents for Units 2 and 3 are defined in the Preliminary Design and Analysis Report (PDAR), Volume I, Section XI.5. The accidents which are significant for control room design are the fuel loading accident, described in PDAR Section XI.5.2, and the LOCA inside the drywell, described in PDAR Section XI.5.4. The radiation sources for control room shielding are the reactor building airborne fission product inventories. The maximum airborne activity for the fuel loading accident is cited as occurring within 1 minute of the accident yielding 1.1×10^4 curies of noble gases and 3.2×10^3 curies of halogens. The maximum reactor building airborne activity for the LOCA is cited as occurring 1 day after the accident, yielding 4980 curies of noble gases, 350 curies of halogens, 86 curies of volatile solids, and 15 curies of other solids. The airborne activities in the reactor building were arrived at using a 10% leak rate from the primary containment, radioactive decay, fallout and plateout, and a reactor building air change rate of 100% building volume per day.

The analysis of the shield design was performed by applying the source as a point source. The point source was located at the nearest approach to the control room that did not require the radiation to penetrate the concrete walls of the reactor building. This point was determined to be a point at columns 39 and K at elevation 636'. The control room point nearest the source point is the southwest corner of the Unit 3 control room 178 feet from the point source. The source was assumed to be 1 gamma ray per disintegration with an average energy of 1.5 MeV. The source was reduced to 12% of the total source to reflect the fractional part of the reactor building volume visible to the control room.

The least shielding path from the point source to the control room requires penetration through a 6-inch floor at elevation 561'-6", the 18-inch west wall of the control room complex, and the 6-inch floor of the battery room. Shielding paths from the refueling floor to the control room which do not pass through the battery room floor must pass through the 3-foot turbine building room floor. The exception is paths through the turbine building floor hatch. In this area, the control room west wall has been constructed of magnetite concrete yielding an effective thickness of 34.8 inches for shielding paths from the refueling floor to the control room.

Shielding paths which do not penetrate this west wall must penetrate the 24-inch battery room ceiling. Shielding paths from the refueling floor along column line 38 must penetrate the 18-inch turbine building wall along column line H and the 20-inch west guard wall to the control room entrance, or penetrate the south wall of the control room, or penetrate the west magnetite wall of the control room.

For shielding paths below the refueling floor, the radiation reaching the control room must penetrate the walls of the reactor building in addition to penetrating the walls of the control room. The reactor building north wall is 4 feet thick up to elevation 621', and the east wall is 2 feet thick up to elevation 613'.

Table 12.3-3 shows the shielding path thicknesses and the calculated dose rates for several conditions.

12.3.2.4 Inspection and Testing

Visual inspections of the shielding were conducted during the construction phase. The value of those inspections, however, was limited to locating major defects because of the massive nature of the shielding. Thus upon initiation of reactor operation, radiation surveys were performed at various power levels. The purpose of these surveys was to assure that:

A. There were no defects or inadequacies in the shielding, equipment, or operating procedures that might, as a result of reactor operation during initial and subsequent testing up to full power, cause unacceptable levels of radiation or exposure to the public, the operators, or the equipment.

B. Radiation areas were posted, and in addition to posting, high radiation areas were provided with access control provisions in compliance with 10 CFR 20 requirements that were in effect at the time. Current access control practices are described in Section 12.5.3.4.

The surveys consisted of both gamma and neutron monitoring with appropriate portable instrumentation. Gamma surveys were performed on all shielding, while neutron surveys were conducted around the concrete containment shield wall and associated penetrations. Further information about startup testing is contained in Chapter 14.

Surveys of dose rates are routinely conducted throughout the plant. Any significant changes in shielding integrity such as concrete aging may be detected in routine radiation surveys, as described in Section 12.5.3.5.

12.3.3 Ventilation

It is normal practice to flow air from points of less contamination to points of greater contamination. This practice has been used throughout the design of the ventilation systems and is a requirement for design basis as described in Section 15.7.3. In some cases proper air flow direction cannot be maintained. Regardless of air flow direction, station radiological programs (e.g., radiological surveys, plant decontamination) are in place to monitor and limit the spread of contamination. Pressure differentials are maintained to prevent backflow of potentially contaminated air. Additionally, the

control room ventilation system is capable of isolation from the outside air during a radioactivity release.

The plant ventilation systems are addressed in Section 9.4. Control room habitability is addressed in Section 6.4.

12.3.4 Area Radiation Monitoring Instrumentation

The area radiation monitoring system detects, measures, and records the radiation level in various areas of each unit. There are 35 detectors for Unit 2 and 36 detectors for Unit 3. The system actuates alarms if the radiation level exceeds a preset level for each monitor. A description of the use of the area radiation monitors to detect radioactive water leakage is presented in Section 5.2.6.2.

12.3.4.1 Design Objectives

The design objectives of the area radiation monitoring system are to monitor the radiation level in areas where personnel may be required to work, to alarm on a radiation level that exceeds a preset level, and to provide a record of the radiation level as a function of time at the location.

12.3.4.2 System Description

The area radiation monitoring system provides operating personnel with a record of gamma radiation levels at detector locations within the various structures or buildings. All monitors provide continuous indication and intermittent recording of radiation levels and alarm in the control room when radiation levels exceed preselected values or when the monitor has experienced an operational failure. Some monitors also alarm at the detector location. Tables 12.3-4 and 12.3-5 list detector locations. Emergency power is supplied by the safety-related instrument bus.

Each monitor has one channel of instrumentation consisting of a gamma sensitive Geiger-Mueller tube and a dc log radiation monitor, complete with fail-safe operational alarm, electrical test circuitry, appropriate high- and low-voltage power supplies, and control and trip contacts. All channels share a calibration unit and a multi-channel recorder. The monitor readout is logarithmic and covers four decades, except for two 6-decade channels provided for the refueling floor and charcoal absorber vault.

Each channel has a high- and low-trip alarm adjustable over the entire scale. These trip functions operate indicator lights that "seal in" on trip and require a manual reset.

Other monitors on the refueling floor are part of the reactor building ventilation monitoring subsystem described in Section 11.5.2.4 and have the capability of isolating the secondary containment in the event of a refueling accident. These monitors are also used for detection of criticality in the new fuel storage pool.

12.3.4.3 Design Evaluation

Area radiation monitor detectors are distributed (Tables 12.3-4 and 12.3-5) in such a way that radiation detection coverage is provided in most areas where personnel may be required to work for extended periods. Increases in radiation above some preselected level annunciate an alarm. The ranges and sensitivities of the equipment are sufficient to detect increases in radiation level above background level.

It has been determined from shielding calculations and from operating experience on other BWR plants that four ranges of monitoring instrumentation sensitivity are adequate for the radiation areas selected for location of area monitors. These ranges are as follows:

A. 10 to 10^6 mrem/hr (low-low sensitivity);

B. 1.0 to 10^4 mrem/hr (low sensitivity);

C. 0.1 to 10^3 mrem/hr (medium sensitivity); and

D. 0.01 to 10^2 mrem/hr (high sensitivity).

A low-low sensitivity monitor is used for one of the monitors on the refueling floor area. This instrument is intended for post-accident (refueling accident) radiation measurements for use in recovery. Since radiation levels could potentially be very high, a low-low sensitivity instrument is used. More sensitive instruments are also located on the refueling floor to provide capability to ascertain that expected low background radiation does exist.

The other three ranges of instruments are utilized in various areas to assure detection capability as low as expected background radiation levels and up to unlikely maximum levels. Most instruments are high sensitivity monitors since they are located in areas with very low background but with potential for moderate radiation levels. Several instruments are medium sensitivity monitors located in low background areas. A few are low sensitivity monitors in higher background areas (TIP cubicle, torus area, HPCI cubicle).

Local alarms at the detector locations were selected on the basis of personnel safety. In areas normally occupied by personnel, local alarms are installed.

12.3.4.4 Reactor Building Crane Monitor

The reactor building crane monitor is designed to enable the crane operator to monitor refueling floor radiation levels from the crane cab. A sensor/converter unit (Geiger-Mueller tube) is mounted on the overhead crane. An indicator/trip unit is mounted in the crane cab. The monitor has a range of 0.1 to 1000 mrem/hr. High radiation and downscale alarms are provided to the crane operator. The downscale alarm function alerts the crane operator of monitor failure. The upscale alarm function serves to alert the crane operator of elevated dose readings on the refueling floor.

An upscale or downscale trip prevents the crane hoist from being raised. A selector switch is installed in the crane cab to bypass the radiation monitor. The reactor building crane is described in Section 9.1.

Table 12.3-1

OCCUPANCY REQUIREMENTS AND ATTENDANT RADIATION DOSE RATES

Degree of Access Required	Design Radiation Dose Rate at Shield Wall (mrem/hr)
Continuous Occupancy	
Outside controlled areas	0.5
Inside controlled area	1
Occupancy up to 10 hours per week	6
Occupancy up to 5 hours per week	12

Radiation areas with dose rates higher than those listed above were designed to be entered on a time limit basis.

Table 12.3-2

DESIGN RADIATION LEVELS FOR TURBINE AND STEAM HANDLING EQUIPMENT

Equipment	Design Radiation Level Outside Shield
Turbine-generator, stop-intercept valves, and piping	5 mrem/hr in regions beyond the turbine shield wall at the operating floor level (elevation 561'-6")
Moisture separators and piping	1 mrem/hr through walls
Stop valves and piping	1 mrem/hr through building operating floor
Air ejectors, steam packing exhausters, and piping	5 mrem/hr through walls
Low pressure feedwater heaters	1 mrem/hr through walls

Table 12.3-3

CONTROL ROOM SHIELDING AND DOSE RATES FOLLOWING ACCIDENT

	West	South	Top
Perpendicular thickness (in.)	24	21	30
Effective thickness (in.)	33.6	39	42
Eight-hour dose for refueling accident (rem)	0.004	N/A	N/A
Eight-hour dose for LOCA with containment leak at 10% per day (rem)	0.036	N/A	N/A

Table 12.3-4

AREA RADIATION MONITORS — DETECTOR LOCATION AND RANGE

UNIT 2

Station Number	Detector Location	Range (mrem/hr)	Auxiliary Unit and Local Alarm
1	Refueling Floor Low Range	0.1—1000	Yes
2	Refueling Floor High Range	$10—10^6$	
3	Refueling Floor Equipment Hatch	0.01—100	Yes
4	New Fuel Storage Area	0.1—1000	Yes
5	Isolation Condenser Area	$1.0—10^4$	
6	CRD and Repair Room	0.1—1000	
7	RWCU Area	0.01—100	Yes
8	Vessel Instrument Rack Area	0.01—100	
9	TIP Cubicle	$1.0—10^4$	Yes
10	TIP Drive Area	0.01—100	
11	CRD West Module Area	0.01—100	
12	CRD East Module Area	0.01—100	
13	Reactor Building South Access	0.01—100	Yes
14	East Low Pressure Coolant Injection Pump Area	0.1—1000	
15	West Low Pressure Coolant Injection Pump Area	0.1—1000	
16	Torus Area	$1.0—10^4$	
17	HPCI Cubicle	$1.0—10^4$	
18	Turbine Operating Floor Elevator Area	0.01—100	
19	Turbine Operating Floor East End	0.01—100	
20	Air Ejector East Area	0.01—100	
21	Air Ejector West Area	0.01—100	
22	Main Control Room	0.01—100	
23	Feedwater Heater Area	$1.0—10^4$	

Table 12.3-4

AREA RADIATION MONITORS — DETECTOR LOCATION AND RANGE

UNIT 2

Station Number	Detector Location	Range (mrem/hr)	Auxiliary Unit and Local Alarm
24	Feedwater Pump Area	0.01—100	Yes
25	Auxiliary Electrical Room	0.01—100	
26	Access Control Area	0.01—100	
27	CRD Feed Pump Area	0.1—1000	
28	Main Condenser Area	0.01—100	
29	Radwaste Conveyor	0.01—100	Yes
30	Radwaste Pump Room	0.1—1000	
31	Radwaste Control Room	0.01—100	Yes
32	Radwaste Storage and Shipping	0.01—100	Yes
33	Not Used		
34	Not Used		
35	Charcoal Adsorber Vault	1.0—10^6	Yes
36	Recombiner Level 1	0.01—100	
37	Recombiner Level 2	0.01—100	

Table 12.3-5

AREA RADIATION MONITORS — DETECTOR LOCATION AND RANGE

UNIT 3

Station Number	Detector Location	Range (mrem/hr)	Auxiliary Unit and Local Alarm
1	Maximum Recycle Chemical Addition Room	0.01—100	
2	Maximum Recycle HVAC Area	0.01—100	
3	Refueling Floor Low Range	0.1—1000	Yes
4	Refueling Floor High Range	$10—10^6$	
5	Refueling Floor Equipment Hatch	0.01—100	Yes
6	Isolation Condenser Area	0.1—1000	
7	RWCU System Area	0.01—100	Yes
8	Vessel Instrument Rack Area	0.01—100	
9	TIP Cubicle	$1.0—10^4$	Yes
10	TIP Drive Area	0.01—100	
11	West CRD Module Area	0.01—100	
12	East CRD Module Area	0.01—100	
13	East Low Pressure Coolant Injection Pump Area	0.1—1000	
14	West Low Pressure Coolant Injection Pump Area	0.01—100	
15	Torus Area	$1.0—10^4$	
16	HPCI Cubicle	$1.0—10^4$	
17	Turbine Operating Floor West End	0.01—100	
18	Air Ejector East Area	0.01—100	
19	Air Ejector West Area	0.01—100	
20	Standby Gas Treatment System	0.01—100	
21	Condensate Demineralizer Area	0.01—100	
22	Unit 2/3 Cardox System Tank Area	0.1—1000	
23	Feedwater Heater Area	0.1—1000	
24	Feedwater Pump Area	0.01—100	Yes
25	CRD Feed Pump Area	0.1—1000	
26	Main Condenser Area	0.01—100	
27	Charcoal Adsorber Vault	$1—10^6$	Yes

(Sheet 1 of 2)

Table 12.3-5 (Continued)

AREA RADIATION MONITORS — DETECTOR LOCATION AND RANGE

UNIT 3

Station Number	Detector Location	Range (mrem/hr)	Auxiliary Unit and Local Alarm
28	Recombiner Level 1	0.01—100	
29	Recombiner Level 2	0.01—100	
30	Filter Building Level 1	0.01—100	Yes
31	Filter Building Level 2	0.01—100	Yes
32	Filter Building Level 3	0.01—100	Yes
33	Concentrator Instrument Rack Area	0.01—100	
34	Maximum Recycle Pump Room	0.01—100	
35	Maximum Recycle Distillate Tank Area	0.01—100	
36	Maximum Recycle Demin. Instrument Rack	0.01—100	

12.4 DOSE ASSESSMENT

Dose assessment is a continuing program for the Dresden Station as a part of the radiation protection program. The station collects and evaluates radiation dose data in accordance with the radiation protection program. Dose assessment information was not developed as a part of the FSAR and its amendments during initial licensing. Design radiation dose rates for Dresden Station are presented in Table 12.3-1.

12.5 HEALTH PHYSICS PROGRAM

This section describes the organization, equipment, and procedures utilized by the radiation protection program, which includes the health physics program.

12.5.1 Organization

A portion of the administrative organization of the health physics program is described in Section 12.1, which describes the "as low as reasonably achievable" (ALARA) program. The health physics program at Dresden Station is administered by the Radiation Protection Manager. The experience and qualification requirements of the Radiation Protection Manager are provided in Section 13.1.8.

12.5.2 Equipment, Instrumentation, and Facilities

Section 12.3 contains a discussion of plant design features which ensure that occupational exposures are maintained ALARA.

12.5.2.1 Decontamination and Change Room Facilities

All changing into or out of protective clothing or equipment is performed at various designated locations within the station perimeter or security fenced area. Provisions are made for collection, transfer, cleaning, and monitoring of all potentially contaminated protective equipment.

The general arrangement of the plant is designed to provide adequate change areas and a personnel decontamination area. The service building is provided with a special shower facility and sink for the decontamination of personnel. Local change areas are provided for control of radioactive contamination at the work areas. Radiation monitors are provided so that personnel may check for contamination.

12.5.2.2 Laboratories

The facilities provided for processing chemical and radiochemical samples, measuring radioactivity, and for other related health physics activities were originally provided for Unit 1 and are further augmented to meet the requirements of Units 2 and 3. The original facilities have been described in detail previously as part of the application for Dresden Station Byproduct Material License Number 12-05650-01, issued September 1959. Laboratory instrumentation is listed in Table 12.5-1.

12.5.2.3 Portable Instrumentation

Battery-powered portable radiation instrumentation is provided for use by all qualified personnel.

12.5.2.3.1 Design Basis

Portable radiation survey instruments are available for the measurement of the alpha, beta, gamma, and neutron radiations expected in normal operation and emergencies. Appropriate instruments and auxiliary equipment are available to detect and measure radioactive contamination on surfaces, in air, and in liquids.

12.5.2.3.2 Description

A list of the instrumentation is given in Table 12.5-2. Friskers and/or personnel contamination monitors (PCMs) are provided at exits from radiologically posted areas.

Personnel dosimetry is provided to be worn by persons in areas where monitoring is required by 10 CFR 20 regulations or station radiation protection program procedures.

Laboratory radiation measuring instruments are provided for alpha, beta, and gamma radiations and for gaseous, liquid, and solid samples.

Secondary calibration sources and check-test sources for the various instruments are provided.

12.5.2.3.3 Inspection and Testing

Proper operation of survey and laboratory instruments is checked frequently by built-in testing circuits and/or radiation sources. Measuring instruments are periodically calibrated with secondary calibration sources.

12.5.2.4 Personnel Protective Equipment

Special protective equipment is provided to minimize the possibility of transfer of radioactive materials to the hands, head and face, or personal clothing of any individual. This equipment includes coveralls, shoe covers, gloves, glove liners, waterproof clothing, boots, and head covers. When engineering controls are not feasible, or cannot be applied, respiratory protective equipment is used to limit the intake of airborne radioactive materials, consistent with the principle of minimizing the total effective dose equivalent. Commonwealth Edison's authority to use these respirators was granted in February 1965. The nature of the work to be done and the contamination levels present are the governing factors in the

selection of protective clothing to be worn. In all cases, radiation protection personnel shall evaluate the radiological conditions and specify the required items of protective clothing.

12.5.3 Procedures

The radiation protection procedures and policies are designed to provide protection of personnel against exposure to radiation and radioactive materials in a manner consistent with applicable regulations. It is the policy of EGC to maintain personnel radiation exposure within the regulations, and further, to reduce such exposure to as low as reasonably achievable via the ALARA program. Individuals are trained to minimize their exposure consistent with discharging their duties and are responsible for observing rules adopted for their safety and that of others.

Radiation protection personnel evaluate radiological conditions and establish the procedures to be followed by all personnel. They ensure that all RP program elements are in compliance with all applicable regulations and that the required radiation protection records are adequately maintained.

Training of operations, maintenance, support, and technical personnel, as well as contractors, in radiation protection principles and procedures is completed before the beginning of their work assignments.

Section 12.1.3 addresses radiation protection provisions of other station procedures.

12.5.3.1 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.

12.5.3.2 Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by each individual is normally obtained from the interpretation of the dosimeter of legal record (DLR). Secondary dosimeters provide day-by-day indication of external radiation exposure.

All employees, contractors, and visitors are issued and required to wear appropriate dosimetry when entering, working in, or visiting radiation areas. In accordance with station procedures.

Under normal conditions each person leaving the plant is required to pass through a portal monitor in the main access facility. Multiple portal monitors are installed to facilitate egress during times of high traffic, such as end of normal workday.

12.5.3.3 Visitors Monitoring

All visitors to the station who enter a radiation area are monitored by appropriate dosimetry or are provided with an escort having such monitoring devices.

12.5.3.4 Bioassay and Medical Examination Program

EGC provides whole body radiation counting service for employees and contractors at Dresden Station in compliance with 10 CFR 20 requirements and in accordance with station procedure.

Special medical examinations are given for authorization to use respiratory equipment (e.g., face masks for areas with airborne radioactive contamination). These examinations (or medical physicals) are performed to meet the requirements of 10 CFR 20.

12.5.3.5 Access Control

Plant areas can be classified as radiation areas, high radiation areas, airborne radioactivity areas, or radioactive materials areas. Areas so classified are posted to warn personnel approaching the area from any direction. Access to posted areas for all work is controlled by plant procedures and by the use of the station radiation work permit (RWP) program.

Control of access to radiation areas is provided through the detailed design of equipment location, shielding, access doors, and passageways. In addition, procedural control is achieved through administrative control of radiation exposures and the control of the concentrations of radioactive material concentration present in various areas of each unit structure. Commonwealth Edison has extensive experience in the application of access control principles.

Accessible areas in which the radiation levels could result in dose rates greater than 100 mrem in 1 hour (at 30 cm from the radiation source) are posted as high radiation areas. Access to these areas is controlled with barriers which prohibit unauthorized entry. Administrative controls are in the Technical Specifications.

12.5.3.6 Radiological Surveys

Radiation surveys of plant areas are performed for a number of reasons, including:

A. Establishment of representative radiation levels;

B. Identification and characterization of contaminated areas;

C. Verification of clean areas;

D. Evaluation of airborne radioactivity concentrations; and

E. Providing pre-job and post-job data as part of the ALARA program.

F. Identification of localized "hot spots" and areas where radiation streaming may occur.

Routine survey frequencies for a given plant area are based on considerations such as area occupancy, the potential for dose rate change due to potential contamination, and the extent of these considerations. Survey schedules are

prepared by the Radiation Protection Department in accordance with station procedures.

Table 12.5-1

LABORATORY INSTRUMENTATION

Radiation Type	General Monitor Type	Detector Type
Alpha	Gross Activity	Ionization or Scintillation
Beta	Gross Activity	Ionization or Scintillation
Gamma	Specific Activity	Hp Ge Ionization
Tritium	Specific Activity	Scintillation

Notes:

The above instrument list is intended to be typical of in-service instrumentation.

Table 12.5-2

HEALTH PHYSICS MONITORING INSTRUMENTS

Radiation Monitoring System	General Monitor Type	Detector Type	Approximate Ranges	Radiation Alarm Types
Alpha	Survey meter	Scintillation or Ionization	0 - 500,000 cpm	None
Beta/Gamma	Survey meter	Ionization chamber	0 - 5 rem/hr 0 - 50 rem/hr	None
Beta/Gamma	Survey meter	G-M tube	0 - 100,000 cpm	None
Beta/Gamma	Extendable survey meter	G-M tube	0 - 999 rem/hr	None
Neutron	Survey meter	BF_3 tube	0 - 5 rem/hr	None
Air Particulate	High/low volume sampler	G-M tube	NA	Local
Gamma	Portal monitor	Plastic scintillator	50 - 200 nCi	Contaminated
Beta/Gamma	Personnel contamination monitor	Gas proportional	2 - 200 nCi	Contaminated

Notes:

The above instrument list is intended to be typical of in-service instrumentation.

APPENDIX 12A POST-ACCIDENT RADIATION LEVELS[1]

The material in this Appendix is from a report prepared for CECo by Sargent & Lundy, in response to Item 2.1.6.b of NUREG-0578. This item required a review of plant shielding for areas requiring post-accident accessibility. This requirement was a result of recommendations of the NRC's Three Mile Island Unit 2 (TMI-2) Lessons Learned Task Force. The material reflects plant conditions as of the time it was written and has been left in the future tense.

12A.1 Introduction - Report Objectives

The review of the TMI-2 accident by the NRC's TMI-2 Lessons Learned Task Force disclosed a number of actions in the areas of design, analysis, and plant operation to be addressed in the short term. These actions are deemed to provide substantial additional protection for the public health and safety. The recommendations and implementation schedule for licensee action were published in NUREG-0578[2] and later clarified at NRC regional meetings with licensees.[3]

The primary objective of this report is to document a design review of plant shielding for areas requiring post-accident accessibility (this task is Item 2.1.6.b of NUREG-0578). The radiation environment due to both contained and airborne radioactivity is considered. If given access requirements are known, then ingress, operation, and egress doses may be determined and compared to applicable limits.

Radiation protection design changes are not limited to simply providing extra shielding. Exposed piping may be relocated in shielded pipe tunnels or raceways; equipment operating stations may be moved to more favorable locations; draining and flushing features may remove offending sources. In areas where airborne radioactivity dominates the radiation field, added shielding offers no protection; enhanced ventilation or protective clothing providing beta radiation protection may be advisable.

A secondary objective of this report is to provide detailed information on the post-accident radiation environment in the vicinity of sampling stations and effluent radiation monitors. The radiation environment information will be used in evaluations required by NUREG-0578 for:

A. Item 2.1.8a, "Post-Accident Sampling Capability" and

B. Item 2.1.8b, "Radiation Monitors."

The introduction of SPC fuel (ATRIUM-9B) does not invalidate the results of the report because the reactor core inventory and the potential radioactive releases from the core are not changed significantly from that obtained with GE fuel.

The use of different fuel types (GE14, Westinghouse Optima2 and AREVA ATRIUM-10XM) and a core uprate to 2957 MW_t does not invalidate the results of this report. The expected increase in radiation levels was determined to be offset by the conservatism in the analytical techniques used by Sargent & Lundy in the development of this report.

12A.2 Source Terms

The source terms (releases from the reactor core) for use in the plant shielding design review were specified by the NRC[3] and reproduced here as Table 12A-1. The ubiquitous noble gas definition is considered to postulate two scenarios: one, a major line break or other uncontrolled release in the

containment, and the second, a major release from the core with the primary coolant pressure boundary remaining intact.

Refer to the discussion in Section 12A.1 for the application of these values to Siemens fuel.

For the present analysis, then, the following source terms are derived from the NRC dictated values.

12A.2.1 Line Break or Other Uncontrolled Release to Containment

An instantaneous release of radioactivity from the core results in:

A. Liquid systems:

1. Noble gases: none
2. Halogens: 50% of core inventory
3. Others: 1% of core inventory

B. Containment air:

1. Noble gases: 100% of core inventory
2. Halogens: 25% of core inventory
3. Others: none

Radioactivity in liquid systems is assumed to be uniformly mixed in a volume of water corresponding to:

A. PWR - primary coolant volume plus contents of a refueling water storage tank, a boron injection tank, and the accumulator tanks; and

B. BWR - reactor vessel and recirculation system volume plus suppression pool water volume.

Radioactivity in the containment air is assumed to be uniformly mixed in the PWR containment volume or BWR drywell volume.

12A.2.2 Core Release - Non-Line Break Scenario

An instantaneous release of radioactivity from the core constitutes:

A. Liquid systems:

1. Noble gases: 100% of the core inventory
2. Halogens: 50% of the core inventory
3. Others: 1% of the core inventory

B. Containment or drywell air: none.

The radioactivity in the liquid systems is assumed to be uniformly mixed in a volume of water corresponding to:

A. PWR - primary coolant volume

B. BWR - reactor vessel and recirculation system plus safety system piping volume.

12A.2.3 Method of Source Evaluation

The major assumptions of source evaluation as dictated by NUREG-0578 are discussed above. Their specific application to the Dresden Station, Units 2 and 3 design is discussed below. Initial core sources were evaluated in accordance with Regulatory Guide (R.G.) 1.3 assumptions.

Refer to the discussion in Section 12A.1 for the application of these values to Siemens fuel, GE14 fuel and core uprate.

12A.2.3.1 Liquid Sources - Non-Line Break Scenario

The reactor core radiation sources described above are instantly and uniformly distributed throughout the dilution volume (9500 cubic feet). A water density of 1 g/cc is assumed throughout, as are standard temperature and pressure (STP) conditions for gaseous sources. These sources were then evaluated in time, taking radioactive decay and radionuclide daughter buildup into full account. No credit was taken for loss terms other than decay. Plateout, filtration and leakage effects were ignored.

12A.2.3.2 Liquid Sources - Line Break Scenario

These sources were handled just as those of Section 12A.2.3.1 above with the exception that the dilution volume has now been expanded to include the torus water volume. The water dilution volume in this case is 122,000 cubic feet.

12A.2.3.3 Reactor Building Airborne Sources - Non-Line Break Scenario

Airborne sources in this scenario result from an assumed failure of a pump seal on a shutdown cooling stream within the reactor building. It is assumed that leakage of 50 gal/min occurs for 30 minutes before isolation is achieved. This results in an airborne radiation source which permeates the affected reactor building and the refueling floor. (This event is being evaluated to ascertain its effect and perhaps support specific leak rates determined in Item 2.1.6.a of NUREG-0578, "Integrity of Systems Outside Containment Likely to Contain Radioactive Materials." The assumptions of this failure accompanying massive core damage [simultaneous double failure] is an unreasonable basis for a safety system design. Nevertheless the event is presently considered.)

Specifically this scenario was modeled by the instantaneous release of 1500 gallons of non-line break case liquid sources into the reactor building. From this the

radionuclides are assumed to evolve immediately from the fluid and to be distributed evenly throughout the total air volume of 9.08×10^{10} cc according to the following partition factors (PF):

A. Noble gases: PF = 1.00

B. Halogens: PF = 0.10

C. Others: PF = 0.01

For conservatism, the leak was always assumed to occur at the time for which the radiation zones were being evaluated. In doing this, any removal due to standby gas treatment system (SBGTS) cleanup or other loss factors, which would have an effect on the airborne concentrations due to leaks at earlier times, were ignored.

12A.2.3.4 Reactor Building Airborne Sources - Line Break Scenario

Airborne sources due to equipment leakage in the line break scenario are handled exactly as they were in the non-line break scenario with the exception that in this case it is the line break scenario liquid sources which are assumed to leak.

In addition to those from equipment leakage the reactor building receives a major contribution to its airborne source, in the line break case, from postulated drywell leakage. Modeling assumptions for this contribution are as follows:

A. Core source release:

1. Noble gases: 100%

2. Halogens: 25%

3. Others: 0%

B. Drywell to reactor building leak rate: 0.5% per day

C. SBGTS exhaust rate: 1 Reactor building volume per day

D. Other loss factors: none

Core source release into the drywell is assumed to be instantaneous and complete at time zero. Source distributions in the drywell and reactor building volumes are assumed to be homogeneous. Drywell leakage and SBGTS cleanup are assumed to start and continue from time zero.

Source term accounting took the above mentioned parameters plus the effects of radionuclide decay and daughter buildup into full account.

12A.2.3.5 SBGTS Filters and Effluent Sources

In both scenarios, the contribution of airborne sources from engineered safety features (ESF) or shutdown cooling equipment leakage to the SBGTS filters was evaluated by conservatively assuming that the entire resulting airborne source had been processed through the filters. The following decontamination factors (DF) were assumed:

A. Noble gases: DF = 0.0

B. Halogens: DF = 10.0

C. Particulates: DF = 20.0

In the line break case the contribution of sources from drywell leakage, which subsequently was processed through the SBGTS, was determined and added to those from ESF equipment leakage.

In both cases the SBGTS effluent source concentrations were evaluated by applying the given decontamination factors to the source concentrations in the reactor building atmosphere (see discussion in Sections 12A.2.3.3 and 12A.2.3.4).

12A.3 Determination of Radiation Environment

12A.3.1 Overview

The determination of the total radiation environment at any point requires consideration of all of the many sources expected to contribute. These sources include:

A. Direct radiation from the airborne and liquid-borne radioactivity in the primary containment.

B. Direct radiation from contained radioactivity in a PWR auxiliary building or BWR reactor and auxiliary building - for example, from pipes and equipment like heat exchangers and tanks, and heating, ventilation, and air conditioning (HVAC) filters.

C. Direct radiation from airborne radioactivity in the BWR reactor building (secondary containment).

D. Immersion dose from airborne radioactivity due to primary containment leakage, emergency core cooling system (ECCS) or shutdown cooling equipment leakage.

E. Radiation from the effluent cloud outside of buildings.

The total radiation dose rate is determined by summing over all of the above individual components. The final dose rates were then assembled and presented as sets of radiation zone maps for the station. Refer to the discussion in Section 12A.1 for the application of these values to Siemens fuel, GE14 fuel and core uprate. Separate zone map sets corresponding

to the two scenarios (line break and non-line break) are prepared for times of 1 hour, 1 day, and 1 week after the onset of the accident. Radiation zone maps were then prepared based on the following zone designations.

ZONE DESIGNATIONS (See Figures 12A-1 through 12A-7F)	RATIONALE
less than 15 mrad/hr	Continuous occupancy
15 - 100 mrad/hr	Possible frequent access; less than 10 CFR 20 high radiation area
100 - 1000 mrad/hr	Accessible for more than 3 hours before exceeding 10 CFR 20 quarterly dose
1 - 10 rad/hr	Limited access - 15 minutes to 3 hours before exceeding 10 CFR 20 quarterly dose
10 - 100 rad/hr	Restricted access - 2 to 15 minutes before exceeding 10 CFR 20 quarterly dose
100 - 500 rad/hr	Restricted access - 20 to 100 seconds before exceeding 10 CFR 20 quarterly dose
greater than 500 rad/hr	Access for life saving only

It is assumed for this analysis that radioactivity is confined to systems which are intended for post-accident operation such as the ECCS. It is assumed that isolation precludes radioactive contamination of such nonsafety systems as PWR letdown, chemical and volume control system (CVCS), waste gas processing, BWR reactor water cleanup, turbine-related systems, or the radwaste systems.

12A.3.2 Method of Analysis

External dose rates due to direct radiation from the drywell, the torus, the airborne and contained sources in the reactor building, the SBGTS and its effluent were determined by the use of a computer code, ISOSHLD,[4] which calculates the decay gamma-ray and bremsstrahlung dose rate at the exterior of a shielded radiation source. The attenuation calculations were performed by the point kernel integration technique using the Simpson's rule of numerical integration. This numerical integration technique consists of dividing the source volume into a number of differential volumes. The source energy is also divided into a number of energy groups narrow enough to consider the buildup factor, attenuation coefficients, and dose conversion factors as constants over the energy range of the group. Each monoenergetic differential volume source is then treated as a point source and the dose rate from each of these point sources is calculated. The total dose rate is calculated by summing up the dose rates for each point for every energy group.

Buildup factors are calculated by the code based on the number of mean free paths of material between the source and detector points, the effective atomic number of a particular shield region, and the point isotropic buildup data available as Taylor coefficients in the effective atomic number range of 4 - 82. Other data needed to
solve most isotope shielding problems of practical interest are linked to the computer code in various libraries internal to the code.

12A.3.2.1 Direct Radiation from the Drywell

There are two radiation sources which contribute to the direct shine from the drywell: liquid and airborne. The activities of these source contributions and the assumptions used for their determination are given in Section 12A.2. These two source contributions require separate evaluation. The results are then added.

The two paths for shine out of the drywell are through the drywell shield wall and shield plug, and through the various drywell penetrations where attenuation may be less than that through the bulk shield. Both of these paths result in radiation fields in the reactor building.

The direct shine from the drywell was not evaluated in detail for several reasons. First and most significantly the evaluation of the various source contributions (other than that from the drywell) had already pushed the radiation zone designation in the reactor building to the greater than 500 rad/hr range for all times considered. This range is considered totally uninhabitable, and evaluation of additional dose contributors becomes moot. Secondly, the same gaseous and liquid sources which cause the shine through the drywell shield and penetrations are circulating, unshielded, in various pipes routed through the reactor building. Any contribution from the sources behind the drywell shield would be a second order effect.

12A.3.2.2 Direct Radiation from the Torus

For the line break scenario, radiation from the torus fluids are the dominant radiation source in the reactor building basement. The assumptions used in determining the liquid sources in the torus are as given in Section 12A.2.

Representative radiation levels near and around the torus were evaluated by modeling a local sector of the torus as a large cylindrical source, 50 feet long and 30 feet in diameter, filled with the liquid source described in Section 12A.2.3.2.

12A.3.2.3 Reactor Building Airborne Sources

Reactor building airborne sources are significant in both the line break and the non-line break case. There are four contributions from these sources:

A. Dose in the reactor building;

B. Shine through the reactor building walls;

C. Shine through reactor building HVAC penetrations into the turbine building; and

D. Shine from the essentially unshielded refueling floor to various places inside and outside of the station.

The method applied in each of these cases is considered below.

12A.3.2.3.1 Radiation Dose in the Reactor Building Due to Immersion in the Airborne Source

Immersion doses in the reactor building are unique in that they are the only case in which beta radiation dose must be considered in addition to the gamma dose. Beta skin dose was evaluated by the ISOSHLD computer code which operates under the assumption of a semi-infinite cloud geometry. For purposes of this study, this assumption is adequately accurate, and when inaccurate will result in a slight conservatism. As a consequence of this assumption, the beta skin dose is independent of the size of the area within which the immersion dose is evaluated.

Conversely, the whole-body gamma dose rate depends on the size of the source cloud in which immersion occurs and was therefore evaluated by the ISOSHLD computer code for four representative cases:

A. A small cubicle;

B. A large cubicle;

C. A residual heat removal (RHR) cubicle; and

D. The refueling floor.

In each case the source cloud was modeled conservatively as a cylinder with the same height and volume as the area being evaluated.

12A.3.2.3.2 Airborne Shine Through the Reactor Building Walls

Shine through the reactor building walls was evaluated with the ISOSHLD computer code for three representative source geometries:

A. 294 feet long by 117.5 feet wide by 45.5 feet high;

B. 20 feet long by 20 feet wide by 22.5 feet high; and

C. 20 feet long by 5 feet wide by 12.5 feet high.

The first source geometry was evaluated for concrete wall thicknesses of zero and 1.5 feet; the second for zero, 1.5 feet, and 3 feet; and the third for zero and 1.5 feet. The sources were modeled as homogeneous rectangular parallelepipeds in contact with the shield under consideration. Here, as elsewhere throughout this study,

140 lb/ft^3 ordinary concrete was assumed to be representative of the shield wall makeup.

12A.3.2.3.3 Airborne Shine Through Reactor Building HVAC Penetrations

Several sizable HVAC penetrations exist between the reactor and turbine buildings. Dose points in the turbine building see the airborne source in the reactor building through the aperture formed by the penetration. This situation was modeled in the ISOSHLD code by using the truncated cone source geometry. With the dose point at the apex, the top of the source volume was assumed to have the same circular area as that of the actual penetration. The depth of the source region was as determined by inspection of the general arrangement drawings.

12A.3.2.3.4 Refueling Floor Shine

The refueling floor is sheltered by metal siding whose attenuation of gamma rays is negligible. The source was therefore modeled as a bare rectangular source sitting on top of the reactor building. For correct modeling, the refueling floor slab and the reactor building walls, and other concrete structures significant to dose points in the yard, turbine building, counting room, and radiation monitor dose points had to be taken into account. Such geometric modeling is beyond the capabilities of ISOSHLD. QAD,[5] a computer code with greater geometric versatility, was therefore put to use. QAD is a point-kernel evaluation code similar to ISOSHLD in almost all respects except for its greater geometric versatility. By means of this more complex modeling, it was possible, for example, to evaluate the dose rate at ground level which is low near the reactor building but rises as the dose point moves away (due to more of the refueling floor source being visible from the ground) and then begins to fall as the dose point moves further out and distance becomes the dominant effect.

12A.3.2.4 Shine from Reactor Building ECCS Piping

Evaluating dose contributions from the large number of pipes carrying radioactive fluid through the reactor building, on a case-by-case basis, was deemed to be impracticable. As an alternative, a set of curves was generated to show the dose rate due to pipes filled with torus fluid sources as a function of several variables. Pipes with nominal diameters of 2 inches, 4 inches, 8 inches, 10 inches, 12 inches, 14 inches, 18 inches, and 24 inches were considered. All pipes were taken to be 20 feet long and assumed to be of Schedule 40 wall thickness, except for the 18-inch and 24-inch lines which were taken to be Schedule 20. The pipes were filled with sump fluid sources homogeneously mixed in water having a density of 1.0 g/cc. The specific activity in the water was based on considerations discussed in Section 12A.2.

Evaluations were made for post-accident times of 1 hour, 1 day, and 7 days. For all of the above parameter variations, dose rates were determined at 10 feet from the center of the pipe as a function of shield thickness (for 1 foot to 4 feet of 140 lb/ft^3 concrete). These curves allowed for a quick but accurate evaluation of the dose rate

from any given pipe at a location of concern. With the aid of a set of marked-up piping and instrument diagrams (P&IDs) which showed the pipes assumed to be carrying radioactive fluids, a set of mechanical piping drawings was marked up to show the pipes of concern; the contribution of each pipe to the dose rate at the points of concern was then determined from the dose rate curves.

12A.3.2.5 Reactor Building Equipment Containing Recirculated Fluids

The modeling of the low pressure coolant injection (LPCI) heat exchangers was set up for the ISOSHLD code as a homogeneous right circular cylinder filled with source terms of either liquid with a specific activity equal to the primary containment sources discussed in Sections 12A.2.3.1 and 12A.2.3.2.

12A.3.2.6 Shine from the SBGTS Filters

Radiation sources held on the SBGTS filters were described in Section 12A.2.3.5.

The dose rates resulting from these sources were evaluated with the ISOSHLD computer code. The filters were modeled as rectangular parallelepipeds 4 1/2 feet by 4 1/2 feet by 2 1/2 feet on a side. The source region was assumed to be filled with carbon at a density of 0.5 g/cc. No other credit for shielding was taken.

12A.3.2.7 Shine from the SBGTS Exhaust Line

Radiation sources in the SBGTS exhaust line were described in Section 12A.2.3.5. Dose rates due to these sources were evaluated in the same manner as described in Section 12A.3.2.4, except that the pipe was assumed to be filled with STP air.

12A.3.2.8 Shine from the Stack Plume

Radiation sources applied to the stack plume were determined by correcting those in the SBGTS exhaust line for a 4000 ft^3/min exhaust rate and an assumed 1 m/s wind speed. The plume was modeled as an air-filled cylinder 20 meters in diameter with its centerline 100 meters above the ground. Dose rates determined assume that the plume is directly overhead.

12A.4 Results

Refer to the discussion in Section 12A.1 for the application of these values to Siemens fuel, GE14 fuel and core uprate.

The results of this study are shown on six sets of radiation zone maps included with the figures at the end of this report. The six sets of radiation zone maps are designated A, B, C, D, E or F corresponding to the Line Break Case (LB) at 1 hour, LB at 1 day, LB at 1 week, Non-Line Break Case (NLB) at 1 hour, NLB at 1 day, and NLB at 1 week, respectively. Each set comprises seven figures which show the radiation dose rates expected on the property plat and the various elevations of the

station. The radiation zones and the sources from which they result are discussed below.

12A.4.1 Radiation Zones in Figures 12A-1A - 12A-1F

The radiation zones in Figures 12A-1A - 12A-1F are those for the property plat. The radiation fields depicted on these zone maps are those due solely to the airborne cloud on the refueling floor. The dose rates on ground level were evaluated in the worst direction, due south along the centerline between Units 2 and 3. The radiation zone boundaries were then drawn at a constant perpendicular distance from the surface of the refueling floor. Details of radiation zones on the ground level are shown on Figure 12A-3A - 12A-3F.

12A.4.2 Radiation Zones in Figures 12A-2A - 12A-2F

The radiation zones on the basement floor are due to the airborne cloud in the reactor building and equipment such as the torus, LPCI heat exchangers, and core spray pumps handling highly radioactive fluids in the line break scenario. Either of these sources are sufficient to yield dose rates greater than 500 rad/hr. In the NLB scenario, the shutdown cooling system was assumed to handle primary coolant with no torus dilution. The airborne cloud in the reactor building due to equipment leakage results in a dose rate greater than 500 rad/hr.

12A.4.3 Radiation Zones in Figures 12A-3A - 12A-3F

The radiation zones in Figures 12A-3A - 12A-3F are for the ground floor. The radiation sources on this level for the reactor building are the airborne cloud and pipes carrying post-accident fluids. Far out from the building, the radiation source is the airborne cloud on the refueling floor. Closer to the building localized radiation fields higher than the refueling floor exist. These higher dose rates are due to the airborne cloud in the reactor building and highly radioactive pipes in proximity to the airlock doors, railroad entrance, and to relatively thin (1.5 feet) reactor building walls. In the NLB scenario, the LPCI and core spray lines were assumed to be carrying relatively clean torus water. In this case, the main sources of radiation near the reactor building are the airborne cloud in the reactor building and control rod drive (CRD) equipment near the air locks.

The radiation zones on the ground level of the turbine building are determined by the airborne cloud on the refueling floor and the SBGTS filter train on elevation 534'0". The north-south corridor between the main condensers is greatly affected by these filters. Examination of the zone maps shows that a portion of the main corridor going east-west is also affected by the SBGTS filter train. The radiation levels due to the SBGTS filters, for all practical purposes, separate the east and west wings of the turbine building. On the ground floor, the wings of the turbine building, along with the north-south corridor between the main condensers have 12 - 14 inches of concrete between them and the refueling floor. For conservatism, these areas were zoned on the basis of 12 inches of concrete. The high-pressure heater bays are exposed to the airborne cloud in the reactor building through

penetrations in the turbine building/reactor building wall. In the NLB scenario, the shine through the wall due to shutdown cooling piping dominates the dose rate from the penetrations. Because of the shielding, the accident sources do not affect the condenser areas or radwaste areas. They, therefore, were zoned normal shutdown.

12A.4.4 Radiation Zones in Figures 12A-4A - 12A-4F

Figures 12A-4A - 12A-4F show the radiation zones on the mezzanine floor. The sources are the airborne cloud in the reactor building and the unshielded LPCI and core spray piping. In the NLB scenario, the shutdown cooling heat exchangers and piping will be carrying the post-accident fluid. The dose rates on the mezzanine floor in the area of the moisture separators and low-pressure heaters of the turbine building are due to penetrations in the turbine building/reactor building wall. In the corridor between the two moisture separators, the two radiation sources are the shine from the refueling floor and the SBGTS filter train. In the short term, the dose rates due to the filter train increase with time while those from the refueling floor decrease.

The dose rates in the control room and adjacent areas are due to the shine from the refueling floor airborne sources.

12A.4.5 Radiation Zones in Figures 12A-5A - 12A-5F

Zones shown in Figures 12A-5A - 12A-5F are the reactor building main floor. The radiation levels in these zones are due to the airborne radiation, the isolation condenser lines, and the possible contamination of the pressure suppression pipe out to its second isolation valve.

The dose rates on the main floor of the turbine building are due to shine of the refueling, penetrations in the turbine building/reactor building wall, an access doorway to the reactor building, and the SBGTS filter train on the mezzanine floor.

The SBGTS filter train is the largest perturbation to the radiation field on the turbine building main floor. Because of shielding, the radwaste and off-gas cubicles are zoned normal shutdown.

12A.4.6 Radiation Zones in Figures 12A-6A - 12A-6F

The radiation zones shown in Figures 12A-6A - 12A-6F are elevations 589' and 613' of the reactor building and the turbine building fan floors near these elevations. The radiation levels in the reactor building are due to mainly airborne sources. In the NLB scenario, the isolation condenser may be used before going to shutdown cooling. If so, the isolation condenser and associated pipes may handle significant quantities of noble gases from the reactor vessel. The radiation levels on the fan floors are due to shine from the refueling floor and the airborne cloud in the reactor building.

12A.4.7 Radiation Zones in Figures 12A-7A - 12A-7F

The high pressure coolant injection (HPCI)/diesel generator building was zoned on the basis that its atmosphere is the same as that of the reactor building and that there exists potential for contaminating unshielded condensate piping. In the NLB scenario, the reactor steam used to drive the HPCI turbine could have a large quantity of noble gases in addition to the 0.2% carryover of halogens from the reactor water. The personnel access way into the HPCI building has both a partially shielded and unobstructed view of a 16-inch LPCI line in the Unit 2 reactor building.

12A.5 Addendum A-Radiation Environment At Sampling Stations

A.1 Introduction

Closely coupled to the objectives of the post-accident zone maps generated in this document are the concerns of Item 2.1.8.a of NUREG-0578, which involves post-accident sampling capability. It requires a review of sampling capability to insure that a post-accident sample can be obtained within an hour post accident without any individual exposures in excess of 3 rem to whole body and 18-3/4 rem to the extremities. The two types of samples in question are a primary coolant sample and primary containment atmosphere.

Refer to the discussion in Section 12A.1 for the applicaiton of these values to Siemens fuel, GE14 fuel and core uprate.

A.2 Results

At Dresden Units 2 and 3, the primary coolant sample sinks are located on the main floor (elevation 561') of the reactor building and the drywell air sampling racks are on the mezzanine floor (elevation 538') of the reactor building. As presently designed, an attempt at post-accident sampling would probably start with an entry into the reactor building through the airlocks. Upon entering either reactor building one would encounter potentially contaminated CRD equipment. A contaminated 4-inch CRD sparger (line 0308) can read 300 rad/hr at 10 feet at 1 hour. Dose rates due to radionuclides uniformly spread throughout the reactor building and refueling floor atmosphere are on the order of 1100 rad/hr in large open areas, of which approximately 800 rad/hr is due to beta particles and 300 rad/hr is due to gamma radiation. Traveling through the reactor building exposes one to varying gamma whole body dose rates depending on the size of the area confining the airborne cloud. The beta dose rate is insensitive to these considerations. The value 1100 rad/hr at 1 hour can conservatively be applied throughout the reactor building.

An individual taking a primary liquid sample could be exposed to a dose rate of 12 rad/hr at 3 feet from a typical unshielded sample line. An unshielded 10-cc vial of primary coolant could have a contact dose of 2000 rad/hr which could be reduced to 700 mrad/hr by 3 inches of lead, all at 1 hour.

An individual taking a primary containment atmospheric sample could be exposed to 8 rad/hr at 3 feet from a typical sample line. A 10-cc vial at one hour could have a contact dose rate of 1300 rad/hr and one of about 1 rad/hr through 3 inches of lead. The Unit 3 containment air sampling panel has an extra source of radiation. It is in close proximity to a 14-inch core spray line which can read 2300 rad/hr at 10 feet 1 hour post-accident.

It is obvious from the dose rates quoted above that it is not feasible for an individual to enter the reactor building, obtain either type of sample and leave the reactor building while still receiving less than 3 rem whole body and 18 3/4 rem to the extremities.

12A.6 Addendum B-Radiation Environment At Radiation Monitors

B.1 Introduction

Closely coupled to the objectives of the post-accident zone maps generated in this document are the concerns of Item 2.1.8.b of NUREG-0578, which involves functioning of radiation monitors in the post-accident environment.

Refer to the discussion in Section 12A.1 for the application of these values to Siemens fuel, GE14 fuel and core uprate.

B.2 Results

Particulate and iodine monitors 2/3 1788 A and B and noble gas monitors 2/3 1789 A and B are located on the fan floor at elevation 581'4". They are unshielded at present. The radiation field has been conservatively calculated to be 2.33 rad/hr at 1 hour and 1.36 rad/hr at 1 week. An erroneous readout due to refueling floor shine would indicate that the reactor building has not isolated and the short stack is an uncontrolled release point.

Radiation monitor 2-1774 is located in the gas sample house. Examination of the property plot indicated that the stack is approximately 225 feet (69 meters) from the refueling floor using the shortest path. The slant distance would be greater since the monitor is at the back of the radwaste building and some shielding would be provided by the floors of the turbine placing the stack monitor 69 meters on the ground level from the refueling floor unshielded would yield a dose rate of 0.6

rad/hr at 1 day from the refueling floor shine in the line break scenario. This is well below the 25 rad/hr limit of the proposed monitor.

B.3 Recommendations

If the radiation levels on the fan floor at elevation 581'4" are too excessive for the unshielded particulate and iodine and noble gas monitors, the possibility of local shielding should be explored.

12A.7 Addendum C-Doses to the Control Room, Support Centers and General Assembly Areas

C.1 Introduction

Several areas onsite have been identified as requiring continuous habitability or pre-evacuation occupancy. These include:

A. Control room;

B. Technical support center (TSC);

C. Onsite operational support center (OOSC);

D. Generating Station Emergency Plan (GSEP) assembly areas;

E. Gate house;

F. Visitors' center; and

G. Offsite emergency center

The first four of these are addressed below to provide support for the January 1, 1980, response to NUREG-0578. The remaining three will be considered at a later date.

The development of integrated doses on a case-by-case basis was deemed to be impracticable. As an alternate, a set of normalized integral curves were developed for representative sources and representative shielding thicknesses. These allowed an integrated dose to be approximated based on the known dose rate at 1 hour post-accident.

C.2 Result

The results of the investigation are discussed below and are given in Table 12A-2.

Refer to the discussion in Section 12A.1 for the application of these values to Siemens fuel, GE14 fuel and core uprate.

C.2.1 Control Room

The control room is located in the east portion of the turbine building at elevation 534'0". The dose rate to control room personnel is from refueling floor shine. Radiation from this source must pass through a 1.5 foot control room shielding envelope. Using a conservative assumption that the concrete envelope is 140 lb/ft^3, the highest dose rate calculated is 3.39 mrad/hr due to refueling floor shine which implies a 30-day integrated dose of 0.101 rad. Further information is available in Table 12A-2.

C.2.2 Technical Support Center

The TSC is to be located on the west side of the access control building. This location has an unobstructed view of the refueling floor. Based on unshielded 1-hour ground level dose rates, the 30-day integrated dose is 75 rad. See Table 12A-2.

C.2.3 Onsite Operational Support Center

The OOSC is to be located under the Dresden Unit 1 control room. Based on the assumption that it is in the control room shielding envelope, it is conservative to use the control room 1-hour dose rate of 3.39 mrad/hr and a 30-day integrated dose of 0.101 rad. See Table 12A-2.

C.2.4 GSEP Assembly Areas

There are four GSEP assembly areas:

A. The turbine building main hall, near the reactor feed pumps;

B. The warehouse near the flume;

C. The warehouse west of the Unit 3 turbine building; and

D. The new administration building.

Based on the 1-hour dose rates given in Table 12A-2, all the above areas exceed a 30-day integrated dose of 15 rad from the refueling floor shine. See Table 12A-2.

12A.8 Addendum D-Dose to Reactor Building Equipment

D.1 Caveat

The information below should be used as a qualitative indicator of dose to equipment. IT SHALL NOT BE CONSTRUED TO BE A QUANTITATIVE REVIEW OF EQUIPMENT QUALIFICATION REQUIREMENTS, NOR SHALL IT BE APPLIED AS SUCH. The bases for the evaluation in this section are those given earlier in this report. They were developed and intended for the purpose of satisfying the requirements of NUREG-0578. They do not conform to the guidance of Regulatory Guide 1.89 ("Qualification of Class IE Equipment for Nuclear Power Plants"), or Draft NUREG-0588 ("Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment").

Specific differences in the sources and their assumed migration exist. Also, consideration of plateout sources and proper beta-depth dose have been ignored in this report. This report considers only the gamma shine dose from primary water and the gamma immersion doses integrated over a 30-day post-accident period.

This may or may not be applicable depending on the period of time for which the equipment must operate.

Refer to Chapter 3.11.5, "Estimated Chemical and Radiation Environment" for a detailed discussion of dose to reactor building equipment in post accident conditions.

D.2 Discussion

Sargent & Lundy agreed to perform a "rough-cut" review of the potential dose to equipment based on the information generated earlier in this report. It was decided that Sargent & Lundy should perform an evaluation to allow the identification of areas which will exceed 10^5 rad. (This is with exception to the NLB case with an immediate 1500-gallon leak into the reactor building. In this case, the gamma immersion dose is on the order of and possibly slightly higher than 10^5 rad.) On the other hand, dose points in the proximity of pipes or equipment carrying post-accident primary water can exceed 10^5 rad. Our findings on this are given in Tables 12A-3 and 12A-4.

Tables 12A-3 and 12A-4 give the radial distance from a pipe to which the 30-day post-accident dose will exceed 10^5 rad. Table 12A-3 is for the LB case. Table 12A-4 is for the NLB case. Both tables consider pipes of various sizes. The information on these tables can be conservatively applied by setting up corridors of dose greater than 10^5 rad around the pipes and equipment shown on Figures 12A-2 through 12A-7 based on the radial distances given. Such an application, however, would ignore the three dimensional orientation of the pipes with respect to the dose points of concern. The information in these tables is best used for determining areas requiring detailed evaluation in the future.

One generalization which can be drawn from this review is that the basement areas of the reactor building can be expected to receive greater than 10^5 rad in the 30-day post-accident period. Equipment in the vicinity of the SBGTS filter train may be expected to receive 10^5 rad or greater since the filter has a maximum contact dose rate of 10^7 rad/hr.

12A.9 Addendum E-Recommendations

None.

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12A.10 References

1. Sargent & Lundy Project 5954-00 for Dresden Units 2 & 3, "Post-Accident Radiation Levels," January 2, 1980.

2. United States Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, dated July 1979.

3. United States Nuclear Regulatory Commission, Handout at a Region III meeting concerning NUREG-0578, held September 25, 1979, in Rosemont, Illinois.

4. Engel, R. L., Greenborg, J., and Handrickson, M. M., "ISOSHLD - A Computer Program for General Purpose Isotope Shielding Analysis," BNWL-236, June 1966.

5. RCIC Computer Code Collection, "QAD - Point-Kernel General Purpose Shielding Codes" CC-48, LA-3573, NASA TM X-1397.

6. DRE00-0073, Rev. 1 “Dose and Dose Rate Scaling Factors to Evaluate Impact of EPU on Radiological Equipment Qualification and Vital Access.”

7. DRE00-0087, Rev 0 “Impact of Extended Power Uprate on the Post LOCA Radiation Dose Rate Zone Maps and Vital Access.”

Table 12A-1

DESIGN REVIEW OF PLANT SHIELDING

Shielding Source Term:

A. Liquid systems:

1.	Noble gases:	100% of core inventory
2.	Halogens:	50% of core inventory
3.	Others:	1% of core inventory

B. Containment air:

1.	Noble gases:	100% of core inventory
2.	Halogens:	25% of core inventory

Radiation Level Guidance:

A.	Areas requiring continuous occupancy: (e.g. control room)	Less than 15 mrem/hr
B.	Areas requiring possible frequent access: (e.g. radwaste panel)	Less than 100 mrem/hr
C.	Others:	Shielding as required to keep exposures less than 10 CFR 20 limits

Note:

Assure adequate shielding of radiation from systems outside containment which may contain primary coolant or gases to assure access to vital equipment. (Field run piping is to be evaluated.)

Table 12A-2[2]

RADIATION ENVIRONMENT AT TSC, CONTROL ROOM, ASSEMBLY AREAS, AND OOSC

Location	1-Hour Dose Rate (rad/hr)	12-Hour Integrated Dose (rad)	24-Hour Integrated Dose (rad)	30-Day Integrated Dose (rad)
TSC (Access Control Building)	0.515	6.70	12.36	75.2
Control Room	0.0039	0.0468	0.0620	0.101
Assembly Areas				
1. Warehouse Near Flume	0.255	3.315	6.12	37.2
2. Warehouse West of Turbine Building	0.535	6.96	12.84	78.1
3. New Administration Building	0.535	6.96	12.84	78.1
4. Turbine Building Main Hall (Near Reactor Feed Pumps)	0.183	2.26	3.90	19.4
OOSC[1] (Under Unit 1 Control Room)	0.0039	0.0468	0.0620	0.101

Note:

1. Area assumed to be within the control room boundary.

2. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are unchanged.

3. Use of different fuel types (GE14, Westinghouse Optima2 and AREVA ATRIUM-10XM) and core uprate does not invalidate the values on this table. The expected increase in radiation levels was determined to be offset by the conservatism in the analytical techniques used by Sargent & Lundy in the development of these values.

Table 12A-3[1]

RADIAL DISTANCE FROM A PIPE FOR 10^5 RAD IN LINE BREAK CASE

Pipe Size (in)	Radial Distance (ft)
2	0.3
4	1.0
8	4.0
10	6.0
12	8.0
14	8.5
18	12
24	17

1. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are unchanged.

2. The values in this table are provided for guidance only and will not be significantly impacted by the use of different fuel types and core uprate.

Table 12A-4[1]

RADIAL DISTANCE FROM A PIPE FOR 10^5 RAD IN NON-LINE BREAK CASE

Pipe Size (in.)	Radial Distance (ft)
2	8.0
4	29
8	103
10	150
12	200
14	220
18	321
24	488

1. The values in this table were determined to be unchanged due to the introduction of ATRIUM-9B fuel since the source terms are unchanged.

2. The values in this table are provided for guidance only and will not be significantly impacted by the use of different fuel types and core uprate.