# 12.3 RADIATION PROTECTION DESIGN FEATURES

## 12.3.1 FACILITY DESIGN FEATURES

→(DRN 02-110)

Compliance with the design feature guidance specified in 10 CFR 20 and Regulatory Guide 8.8 is discussed in Subsection 12.1.2 in detail. The layout of plant radiation zones is described in Subsection 12.3.2.2 and Section 12.4. The counting room is described in Section 12.5. The locations of sampling ports are discussed in Subsection 9.3.2.

←(DRN 02-110)

### 12.3.1.1 Description of Plant Shielding

Plant layouts and cross sections of buildings containing process equipment for treatment of radioactive fluids, and also a plot plan are shown on Figures 1.2 – 1 through 27 and Drawing G136.

### 12.3.1.2 Primary Shield

The primary shield consists of reinforced concrete which surrounds the reactor vessel. The primary shield is designed to meet the following objectives:

a) to attenuate the core neutron flux in order to limit the activation of component and structural materials,

b) to limit the radiation level after shutdown in order to permit access to the Reactor Coolant System equipment,

c) to reduce, in conjunction with the secondary shield and the neutron streaming shield, the radiation level from sources within the reactor vessel in order to allow limited access to the containment during normal operation, and

d) to permit access during shutdown for inspections required by ASME, Section XI.

For purposes of primary shielding design, the normal full power operation of the core and resultant neutron and gamma fluxes are the controlling factors.

### 12.3.1.3 Secondary Shield

→ (DRN 99-2362)

The secondary shield surrounds the primary shield and reactor coolant loops and attenuates, to a safe level, the radiation originating in the reactor coolant and steam generators. During full power operation, the major radiation source in the Reactor Coolant System is N-16, which is created by neutron activation of oxygen during passage of coolant through the core.

← (DRN 99-2362)

The secondary shield is designed to permit limited access to certain areas within the Reactor Building during full power operation. The secondary shield also serves to reduce the full power radiation levels outside the Reactor Building so that normal continuous occupancy outside the Reactor Building is afforded. In addition, the secondary shield helps in reducing the radiation intensity outside the Reactor Building in the unlikely event of an accidental release of fission products into the containment. This function, however, is primarily accomplished by the Reactor Building concrete structure.

After reactor shutdown, the fission and corrosion product activities in the Reactor Coolant System listed in Tables 11.1-2 and 11.1-10 become the dominant sources.

12.3.1.4 Fuel Transfer Shield

→(EC-14275, R306)

The fuel transfer shield protects plant personnel from fission product gamma radiation emitted from the spent fuel elements during core refueling operations. The fuel is removed from the reactor through a canal to a water filled spent fuel storage area located in the Fuel Handling Building. After sufficient decay, the spent fuel is transferred under water to canisters within the transfer cask.

The fuel transfer shielding consists of five parts:

←(EC-14275, R306)

a) The water and concrete of the refueling cavity, spent fuel storage areas and the transfer canal.

b) The fuel transfer tube shield structure which shields the space between the transfer canal and the steel containment structure, and the space between the steel containment and the concrete Shield Building (annulus).

c) The fuel transfer tube in the gap between the Reactor Building and the Fuel Handling Building is shielded by a crescent shaped lead shield covering the top and sides of the tube.

→(EC-14275, R306)

d) The spent fuel canisters within the transfer cask are loaded underwater.

e) The spent fuel transfer cask contains and shields the loaded canister. The transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. The loaded canister has a steel shielded lid to allow access for sealing operations.

←(EC-14275, R306)

The fuel transfer tube shield structure (part b above) is a combination of concrete, steel, lead and silicone foam. The design objective of the shield structure is to enclose the fuel transfer tube by shielding materials to prevent any inadvertent exposure to this high radiation source. Access openings are provided for periodic leakage inspections and any necessary testing of the fuel transfer tube. An access way below the shield structure in the annulus is incorporated into the design to provide an alternate exit route in the event of a fire or other accident in the annulus. The shield design has provisions for the expansion of the steel containment vessel during normal operations, and also for the maximum expansion during and following a LOCA.

The general dose rate limits used in the shield design are:

a) 5.0 mrem/hr for the areas most likely to be occupied and away from narrow gap areas. These areas include from floor level up to head height on the outside of the shield structure and away from narrow shield gaps.

b) 25 mrem/hr for the areas of infrequent occupancy and narrow exposure areas. These areas include elevation above head height outside the shield structure and the narrow shield gaps. Areas that exceed the 25 mrem/hr (i.e., Shielding Defective Areas) may be posted with radiological signs/barricades to restrict personnel occupancy.

→(DRN 99-2362, R11; 06-319, R14-B)

The 5.0 mrem/hr allows 20 hours per week occupancy without exceeding 100 mrem/week and this level is not exceeded in the occupied locations around the fuel transfer tube shielding. The areas that would see infrequent occupancy, and those that have narrow exposure areas are allowed to have higher dose rates. This is a design compromise that reduces the weight of shielding-material (welded to containment vessel), and the volume of shielding material so as not to overly restrict the access opening into the shield structure. It is assumed that a spent fuel assembly will pass through the transfer tube in one minute and that a maximum of 358 assemblies (217 offload + (217 – 76) maximum reload) will be transferred during a refueling outage. A dose rate of 5.0 mrem/hr will result in a total exposure of thirty mrem maximum if a person is present for all transfers. If it is assumed that a maximum of 32 spent fuel assemblies are transferred in an eight hour shift, then the average exposure per shift will be 2.7 mrem for the same conditions. A dose rate of 25 mrem/hr will result in exposures five times greater than those cited above; however, the larger dose rates apply to areas of infrequent occupancy and the narrow gaps. Thus, the actual doses in these areas are expected to be much lower. The narrow gaps are the two in. gaps between the steel containment vessel and the concrete, steel or lead shields. These gaps are shielded by compressable silicone foam and steel. Personnel will have a low exposure at these gaps if they approach the shield structure and press against the steel containment vessel. In these cases the exposures will be to the extremities, as defined in 10CFR20.1003.

←(DRN 99-2362, R11; 06-319, R14-B)

→(DRN 99-1098, R11)

The refueling canal, the fuel transfer tube, the steel containment vessel, the concrete containment building, and the shield structure were originally modelled for dose rate calculations on the SPAN-4 program (Reference 12). This is a point kernel shielding program that uses three dimensional rectangular, cylindrical and spherical geometries in any combination to describe the physical features and radiation sources. For this application the exact geometry was modelled and no compromise of shape was necessary. The program uses Gaussian quadrature, and the Gauss point distributions are determined by Legendre and/or Laguerre polynomials. The data for calculations, such as, cross-sections, buildup factors, material compositions, energy structures, flux to dose conversion factors and Gauss quadratures are contained in the program library, and this data can be changed or added to by the user. For this application, considerable attention was given to the Gauss point distributions for the calculations through the narrow gaps. Trial and error and an excess of Gauss points were used in the narrow gap calculations to assure that the dose rates were adequately calculated.

As part of the spent fuel pool reracking, licensed in 1998, the fuel transfer tube was reanalyzed using the QAD-CGGP program. The results are documented in Reference 13.

←(DRN 99-1098, R11)

The fuel transfer tube shield structure has two parts, which are separated by the steel containment vessel. The first part is between the steel containment vessel and the refueling canal, and it completely blocks off this passageway up to elevation 21.00 feet.

The second part is between the steel containment vessel and the concrete reactor building wall. There is a passageway beneath this shield structure to provide an alternate exit from this annular area. These two parts of the shield structure are not connected at any point, and each is physically separated from the steel containment vessel by a two in. gap. This gap provides space for thermal expansion of the steel vessel.

The first part of the shield structure consists of:

a) a four ft. thick column on the west side between the floor and elevation 17.00 feet.

b) a four ft. thick column on the east side between the floor and elevation 5.50 ft. and 13.50 ft. and 17.00 ft. The space between elevations 5.50 ft. and 13.50 ft. is shielded by stacked lead brick to a thickness of 1.0 ft., which are retained in place by a removable steel frame. The lead brick can be removed to provide access to the interior of this shield structure. The access width is about 33 in.

c) a four ft. thick slab on the top of the structure with a length of 18.5 ft. The top of the slab is at an elevation of 21.00 ft.

d) a bottom shield of concrete filled between the side shield columns (a and b above) to an elevation of 6.00 ft. A one ft. by two ft. lead brick slab is embedded in this lower shield adjacent to the steel containment vessel. This lead brick is needed to shield the diagonal radiation paths through the two in. gaps and the two in. steel containment vessel to the space below the shield in the adjacent annulus.

e) a filler material in the two in. gap to provide shielding and flexibility for thermal expansion of the steel vessel. The material chosen for this application is a cellular polymeric silicone material (silicone foam) which is densified with powdered lead to a total density of 100 pounds per cubic foot. The steel containment vessel is expected to have its minimum radius (at its minimum temperature) at the time of construction, so the gap inside the containment will range from two in. to approximately three in. for a short time following a LOCA. Under normal operating conditions the maximum gap is expected to be approximately 2.25 in. The growth in the gap width can be compensated for by installing the silicone foam with an initial compression so that it will automatically expand as the gap grows. This is accomplished by precasting the silicone foam at an appropriately lower density and installing it against the inside surface of the steel containment vessel. The weight of the wet concrete poured against it will compress it to the desired density and develop the necessary initial compression. For a 1/4 in. deflection in two in. the pressure developed is estimated to be approximately 13 pounds per square inch. It is not considered necessary to compensate for a gap growth of greater than 1/4 in., since the fuel transfer tube will not be used during accident conditions.

f) Two silicone foam blocks adjacent to the stacked lead bricks and the steel containment vessel. The smaller block fills the outer portion of the W8x24 post and the larger block is adjacent to the west side of the W8x24 post and the steel containment vessel. The latter block is fastened to the containment vessel so as to move with its expansion and contraction. The two blocks extend from elevations 6.00 ft. to 13.00 ft. It is not necessary to fill the inner portion of the W8x24 post with silicone foam since its contribution in reducing dose rates is negligible. The silicone foam material for this application is lead loaded to a total density of 150 pounds per cubic foot, and they are not subject to compressive loads. The size of the blocks are limited so as not to overly restrict the access opening size or the maneuvering space inside the shield structure.

The second part of the shield structure consists of:

a) a four ft. thick column on the west side between elevations 4.625 ft. and 17.00 ft.

b) a four ft. thick column on the east side between elevation 13.50 ft. and 17.00 ft. The space between elevations 5.50 ft. and 13.50 ft. is shielded by stacked lead bricks to a thickness of 1.0 ft. Its function and dimensions are similar to item b) under the first part of the shield structure.

c) a four ft. thick slab on the top of the structure with a length of 18.5 ft. The top of the slab is at an elevation of 21.00 ft.

d) a 1.5 ft. thick steel slab on the bottom with its top elevation at 6.00 ft. It is supported by steel columns, which provide a passageway beneath the shield with a width of approximately 25 in. and a height of 6.0 ft.

e) five steel slabs with cross sectional dimensions of 3 x 9 in. These slabs are retained against the containment vessel by steel brackets which are welded to the steel containment vessel, and as close to the main shield structure as construction will permit. The nine in. dimension is adjacent to the steel vessel. The slab on the west side of the shield structure extends from elevation 3.00 ft. to 21.75 ft. Two of the slabs are on the east side of the structure between elevation 3.00 ft. and 4.63 and elevation 13.50 ft. and 21.75 feet. A third slab in the east side is adjacent to the stacked lead bricks between elevation 4.63 ft. and 13.00 ft. The fifth slab is on the top and it fills the space between the side slabs. The two in. gaps between the concrete shields and the steel containment vessel are air gaps, the silicone foam filler material cannot be used on this side of the containment vessel since the approximate one in. maximum expansion of the containment vessel would result in excessive pressures (buckling forces) that were not included in the vessel design.

f) The two silicone foam blocks on the outside of the containment vessel and adjacent to the stacked lead brick serve the same function as the two blocks in item f) under the first part of the shield structure. These blocks are also lead loaded to a density of 150 pounds per cubic foot.

g) A silicone foam slab with a thickness of three in. is placed below the 18 in. thick steel slab. This shield is used to reduce the dose rates in the passageway below the steel slab in the annulus. The silicone foam for this application is 150 pounds per cubic ft., and it is not subject to compressive forces. It is shaped around the support posts to avoid compression when the containment vessel is expanded to its maximum dimension. This shield extends from the floor to elevation 4.50 ft., and azimuthally along the containment vessel for the same length as the steel slab.

All of the shield components are shaped to conform to the curvature of the steel containment vessel. Following an accident inside the Containment Building it would be necessary to inspect these shields to determine if any have shifted position.

→ (DRN 99-1098)

Gamma-ray dose rates from the spent fuel in transit in the fuel transfer tube were calculated with the QAD-CGGP program (Reference 14). QAD is a combinatorial geometry point kernel code system that uses the geometric progression buildup factor. The source-term input for QAD was determined with SAS2H-ORIGEN-S (Reference 15), which gave the 18-group gamma-ray source strengths shown in Table 12.2-17. The gamma intensities shown in the table are those for a single, average assembly with an initial fuel enrichment of 5.5% (Note: enrichment used for conservative calculation purposes only; licensee can not posess fuel of this enrichment level) and a fuel exposure of 70,000 Mwd/mtU. In calculating the dose rates, a peaking factor of 1.8 was used in the QAD program.

The fuel carrier, which holds the spent fuel assemblies as they move through the fuel transfer tube, is equipped to carry two assemblies simultaneously; the assemblies are essentially “piggy-backed” one

← (DRN 99-1098)

→ (DRN 99-1098, R11)

above the other, forming a gamma source that is long horizontally, tall vertically, and narrow laterally. This two-assembly source assumed for the dose rate calculations is shielded by the water in the inner steel tube of the transfer-tube assembly, the inner tube itself, and the outer steel tube of the transfer-tube assembly. The source is centered vertically and horizontally within the transfer tube.

The gamma dose rate from the two, in-transit, spent fuel assemblies was calculated at six locations. The locations and the calculated dose rates at those locations are described in the following paragraphs.

The dose rate at the outer surface of the transfer tube assembly, at its vertical mid-height and in the annulus between the steel and concrete of the reactor building, is $86.4 \times 10^6$ mR/hr.

→(DRN 03-2066, R14)

At a point whose elevation is that of the vertical mid-height of the transfer tube, on the east face of the special shielding to the east of the transfer tube, the dose rate is calculated to be $1.60 \times 10^5$ mR/hr in the annulus between the steel and concrete of the reactor building. This high dose rate is very conservative, for the calculation did not consider the shielding benefit of the lead-loaded silicon foam (silfoam) material that fills the interstice between the reactor building steel and the north-south lead shielding.

←(DRN 03-2066, R14)

For a dose rate point 9'-9" west of the transfer tube centerline, and at the elevation of the tube's vertical mid-height, the dose rate (through the 4-foot concrete shield wall) is 36.4 mR/hr in the annulus between the steel and concrete of the reactor building.

At a point between the refueling canal wall and the reactor building steel at elevation 21.00' (on a concrete floor which has a thickness of 4 feet), above the transfer tube, the dose rate is 0.18 mR/hr.

For a location between the refueling canal wall and the reactor building steel, at elevation 21.00' and at the top of a crack that is actually filled with silfoam, but is assumed to be air, the dose rate is $1.79 \times 10^5$ mR/hr. Again, this high dose rate results from the assumption the crack is empty and the dose rate point looks directly at the unshielded transfer tube. The conservatism introduced by this assumption can be seen by comparing the high dose rate with that for a point shifted slightly so that there is no unshielded view of the transfer tube; in the latter case, the dose rate drops to less than 0.20 mR/hr.

In a related calculation, the dose rate at the bottom of the silfoam-filled crack of the preceding calculation, which is at elevation 17.00', the dose rate is $2.73 \times 10^5$ mR/hr.

Most of the dose rates given above represent conservative maximums, and this was the intent in performing the calculations. The conservatism was introduced into the calculations a number of ways, including the following specified input assumptions.

Two fuel assemblies for the source, each assumed to have the maximum source term.

A fuel exposure of 70,000 Mwd/mtU, with 5.5% initial enrichment (Note: enrichment used for conservative calculation purposes only; licensee can not posess fuel of this enrichment level).

A conservative peaking factor of 1.8 for both fuel assemblies. A cooling time of 3 days.

The use of AP (anterior-posterior) gamma-ray fluence-to-dose conversion factors (Reference 16).

The dose rates actually experienced by personnel should be significantly lower than the hypothetical dose rates shown above. In regions between the fuel transfer tube and the special shielding (regions not accessible to personnel), the dose rates may approach the values given above during periods when fuel assemblies are in transit.

← (DRN 99-1098, R11)

The fuel transfer tube is shielded in the space between the Reactor Building and the Fuel Handling Building (Part C) by a crescent shaped lead shield.

12.3.1.5 Shield Building

The Shield Building is a reinforced concrete structure with cylindrical wall three ft. thick and a 2.5 ft. thick dome. In conjunction with the primary and secondary shields, it limits the radiation level outside the Shield Building from all sources inside the containment to no more than 0.25 mrem/hr at full power operation.

The Shield Building provides protection to plant personnel from radiation sources inside the containment following a design basis accident. These radiation sources are discussed in Subsection 12.2.1.9. The Shield Building walls will act to greatly attenuate the direct offsite gamma dose following a design basis accident.

12.3.1.6 Neutron Streaming Shield

The potential for radiation streaming (neutron and gamma) through the annulus around the reactor vessel has been analyzed to determine the radiation fields that could occur in areas of containment which may require occupancy.

Because operating experience indicates that streaming gamma dose rates during operation are a relatively small fraction of the corresponding neutron dose rates, the analyses of the streaming dose rates in containment has been limited to the determination of neutron dose rates. The angular neutron flux as a function of energy which emerges from the surface of the reactor vessel at selected locations has been derived using the DOT 3(5) computer program utilizing an S4 angular quadrature and a P3 Legendre expansion coefficient for anisotropic scattering. Refer to Subsection 12.2.1.2 for a description of these sources.

These vessel emergent angular fluxes have been used as input to a Monte Carlo analysis of the streaming problem.

Morse-CG(6) a general purpose Monte Carlo multigroup neutron (and gamma ray) transport code with combinational geometry, has been used to compute the neutron streaming and the resultant dose rates at various locations inside containment.

The cross section library used in the calculations is based on DLC-23 or CASK library(7). This library is a coupled neutron and gamma ray library and the data in the library are obtained by collapsing cross sections over a PWR core spectrum.

The geometry model used for the containment dose rate calculation includes the containment vessel steel shell and Shield Building concrete, the major features of the containment internal structures such as the refueling cavity, the shield walls around the steam generators and pumps, and a detailed description of the reactor vessel cavity, the reactor vessel, the primary piping, the missile shield, the primary shield, and the ring girder support.

A check on the accuracy of the modelled geometrical representation of the reactor cavity and containment has been obtained by verification by means of computer generated pictures of the model taken at different elevations and sections.

The ring girder support of the reactor vessel has been designed in such manner as to also provide shadow shielding against neutrons and gamma rays streaming up the annular gap between the vessel

and the cavity walls. The ring girder design also reduces the neutron and gamma ray streaming through the primary penetrations. Vessel cooling and insulation requirements impose a limitation on the minimum ring girder to vessel gap that can be achieved.

The neutron histories start on the surface of the reactor pressure vessel with an energy and a direction determined by processing the annular fluxes at the outermost mesh of the vessel determined by an R-Z DOT 3 calculation, with the DOMINO code[8] , which is explicity set up to provide the proper source information for the MORSE program. Variation in source strength along the circumference has been neglected for conservatism.

The MORSE calculation has been performed in two stages by a MORSE to MORSE coupling technique. The first stage stops the random walk at the vessel flange level. Particle escapes at the flange as computed in the first stage are written to collision tapes which are used as inputs to subsequent MORSE runs utilizing the entire containment model.

No biasing has been used for either stage of the calculation. The Monte Carlo calculation of the dose rates in containment, however, has been limited to the energy group of neutrons emerging from the vessel with energies above 0.11 MeV in the interest of saving computer time and cost, as it is known from experience at operating plants that the dose rates in containment are due primarily to fast neutrons. The contribution to the containment dose rates from the lower energy groups has been estimated to be approximately three percent. The response to the energy groups considered is followed down to thermal energies.

Table 12.3-4 lists the estimated neutron dose rates at selected points within containment. These values are corrected for a factor of two conservatism in the calculated vessel emergent fluxes. The uncertainty in the Monte Carlo calculation is in the neighborhood of 30 percent.

While the computed neutron dose rates on the containment operating floor are relatively high, the dose rates in the general areas of the lower floor where personnel may require access are expected to be much lower. Computed values of the floor dose rates are less reliable due to the difficulty in modelling the problem and the large uncertainties in any ensuing results; however a reasonable estimate of the expected levels can be made by comparison of the dose rates measured at several operating plants on the lower floors with the corresponding dose rates on the operating floors of the same plants, and scaling of these lower floor dose rates by the ratio of the predicted Waterford 3 operating floor dose rates to the measured operating dose rates.

Measurements conducted at Calvert Cliffs [9], St. Lucie 1 [10], and Millstone 2 [11], scaled for full power, indicate that neutron dose rates at locations where the Waterford 3 dose rates are approximately 20 rem/hr, range from 60-65 rem/hr.

Middle floor dose rates in these plants range from 75-900 mrem/hr, while bottom floors dose rates are in the range of 15-250 mrem/hr.

Expected lower floors neutron dose rates for Waterford 3 should therefore be one third of those of the referenced operating plants, and can thus range from 25-300 mrem in the middle floor, and from five up to 83 mrem in the bottom floors.

Streaming gamma dose rates at the same plants were measured to be one fourth or less of the neutron dose rates for general containment areas. A similar ratio is expected for this plant.

Information was obtained relative to the estimated frequency, length of visit and number of persons to enter the containment during full power operation. Experience has indicated that most of the visits are related to instrument failures. The visits anticipated for Waterford 3 are listed below.

For the series of cabinets 1A, 1B, 1C, 1D, 2A, 2B, 2C and 2D at elevation +21.0 ft. MSL, the I&C personnel expect a visit once every two months for a time of one hour and with three persons. Other visits by I&C personnel at middle and lower levels are anticipated to be once every six months for two hours each visit and with three persons.

The operations personnel expect to visit the containment during full power operation at the middle and lower levels once every two months for 30 minutes per visit and with two persons. It is also anticipated by the operating personnel that visits to the operating floor will be made once every two months for five minutes each visit, and by two persons. These visits would be limited to the outer periphery of the operating floor where the dose rates are estimated to range from 700 to 3000 mrem/hr.

The annual man-rem contribution from these containment visits are conservatively estimated to range from 1.6 to 13.8 man rem using the dose rate estimates for the middle floor levels (25 to 300 mrem/hr) and 700 to 3000 mrem/hr for the operating floor.

#### 12.3.1.7 Reactor Auxiliary Building Shielding

Reactor Auxiliary Building shielding includes concrete walls, covers, and removable blocks which will shield the sources of radiation originating from the Chemical and Volume Control System, Boron Management System, Safety Injection System, the Waste Management System, and portions of the Fuel Pool System. Typical components which require shielding include holdup tanks, decay tanks, demineralizers, fitters, heat exchangers, and associated piping. Activities in these systems are based upon normal system operation with clad defects in fuel rods generating one percent of rated core thermal power and are specified in Tables 12.2-7 through 12.2-11.

#### 12.3.1.8 Main Control Room

For purposes of designing main control room shielding, the radioactivity releases from the maximum loss of coolant accident are controlling. The two sources considered in designing the shielding are:

a) Direct gamma radiation from the containment atmosphere and emergency filters.

→(DRN 03-2066, R14)

Alternative Source Terms based on Regulatory Guide 1.183, release fractions and plate-out fractions are considered. A uniform distribution of radioactivity within the containment is assumed. Credit for post-accident decay is considered.

←(DRN 03-2066, R14)

→(DRN 03-2066, R14)

Doses to Control Room personnel following a LOCA are presented in Section 15.6. Credit for seven ft. of concrete (three ft. for the containment shield wall and four ft. for the main control room shielding) was taken. A minimum shield thickness of four ft. separates the emergency filters from the main control room.

b) Direct Gamma Radiation from Radiation Leakage External to Containment (Cloud)

Thirty day post-LOCA Control Room doses are reported in FSAR Section 15.6.3.3.5 and FSAR Table 15.6-18.

←(DRN 03-2066, R14)

In addition to shielding from external exposures, the main control room is designed to be pressurized under accident conditions with filtered makeup and recirculation in order to minimize the quantity of airborne radioactivity which enters the main control room and thereby ensure compliance with GDC 19 of 10CFR50. A detailed description of the system design is provided in Subsection 9.4.1. Chapter 15 includes an evaluation of the exposures to main control room personnel following the design basis accident.

12.3.1.9 Fuel Handling Building

→(EC-14275, R306)

Shielding is provided for protection during all phases of spent fuel removal and storage. Operations requiring shielding of personnel are spent fuel removal from the reactor, spent fuel transfer through the refueling canal and transfer tube, spent fuel storage, spent fuel transfer cask loading, spent fuel canister seal welding, vacuum drying, helium backfilling, movement into a storage cask, and unloading from a storage cask into the spent fuel pool; and maintenance and inspection of the spent fuel pool purification loop of the Fuel Pool System.

←(EC-14275, R306)

→(DRN 99-1098, R11)

Since all spent fuel removal and transfer operations are carried out under borated water, a minimum water depth above the top of the fuel assemblies is established to provide radiation shielding protection. The dose rate at the water surface is less than 15 mrem/hr. The concrete walls of the fuel transfer canal and spent fuel pool supplement the water shielding and limit the maximum continuous radiation dose levels in working areas to less than 15 mrem/hr from spent fuel sources.

←(DRN 99-1098, R11)

→(LBDCR 16-016, R309)

The refueling water and concrete walls also provide shielding from activated control element assemblies (CEAS), reactor internals removed at refueling times, excores, filters, canisters, and other radiological waste less active than spent fuel. Although dose rates will generally be less than 2.5 mrem/hr in working areas, certain manipulation of fuel assemblies, CEAs, or reactor internals may produce areas where dose rates exceed 2.5 mrem/hr for short periods. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the dose limits specified in 10CFR20.

←(LBDCR 16-016, R309)

The spent fuel pool shielding is based upon the following considerations:

The controlling factor in the design of the spent fuel pool and fuel transfer canal walls are the irradiated fuel assemblies.

→(DRN 99-1098, R11)

The shielding design of the fuel transfer canal is based upon consideration of a spent fuel assembly with a source strength 1.8 times greater than that derived from Table 12.2-17.

For the spent fuel pool (from Reference 13):

a) In the room below the spent fuel pool, the dose rate from the full pool of spent fuel. The fuel in the pool will consist of 48 hot assemblies (3 day cooled, 1.8 peaking factors) and the remainder of the pool will be filled with assemblies which are 1 year cooled with peaking factors of 1.00. The dose rate is 4.17 mR/hr at an elevation of -30.00' (five feet above the floor).

b) In the room below the spent fuel pool, the dose rate from the full pool of spent fuel. The fuel in the pool will consist of 116 hot assemblies (5 day cooled, 1.8 peaking factors), 101 assemblies (5 day cooled, 1.4 peaking factors) and the remainder of the pool will be filled with assemblies which are 1 year cooled with peaking factors of 1.00. The dose rate is 11.7 mR/hr at an elevation of -30.00' (five feet above the floor).

c) At the pipe chase at the wall north of the fuel stored in the spent fuel pool racks. The northern-most row (outer row of assemblies next to the wall) will be 1 year cooled, 1.00 peaking factor assemblies. The remaining assemblies which form the source (interior rows) will be 3 day cooled, 1.80 peaking factor assemblies. The dose rate in the pipe chase is .17 mR/hr.

d) At the pipe chase at the wall north of the fuel stored in the spent fuel pool racks. The northern-most row (outer row of assemblies next to the wall) will be assumed to be water. The remaining assemblies which form the source (interior rows) will be 3 day cooled, 1.80 peaking factor assemblies. The dose rate in the pipe chase is 2.15 mR/hr.

e) In the fuel pool cooling pump room with the refueling canal racks fully filled with fuel (all assemblies will be 1 year cooled with 1.00 peaking factors). The dose rate is 0.05 mR/hr.

f) At the pipe chase at the wall north of fuel stored in the refueling canal racks (all assemblies will be 1 year cooled with 1.00 peaking factors). The dose rate is 0.05 mR/hr.

g) At the north face of Gate #3B, in the Cask Decontamination Pit, from a single assembly moved (suspended approximately 7' underwater from the Spent Fuel Handling Machine) in the Cask Storage Pit. The dose rate from the fuel assembly (5.5% initial enrichment, 70,000 Mwd/mtU bumup, 3-day cooling, 1.8 peaking factor) is 52.2 mRem/hr. The fuel assembly is no closer than above the eighth row of cells from the gate (Gate #3A). If, for any reason, the need arises to store irradiated fuel closer (than seven spaces) to Gate #3A during normal SFP operation the appropriate shielding calculations will be performed, prior to placing fuel into these cells, to ensure that dose rates, in this area, will remain acceptable.

→(EC-14275, R306)

h) In the Rail Bay Area from a single assembly moved (suspended approximately 7' underwater from the Spent Fuel Handling Machine) in the Cask Storage Pit. The dose rate point in this calculation is farther away from the source than the point in "g" above and receives additional shielding from Gate #4 (the gate north of the Cask Decontamination Area). The separation distance of the fuel assembly from Gate #3A (the gate leading to the Cask Decontamination Area) will be the same as in "g" above. The dose rate from the fuel assembly (5.5% initial enrichment, 70,000 Mwd/mtU burnup, 3-day cooling, 1.8 peaking factor) is 0.75 mRem/hr. If, for any reason,

←(DRN 99-1098, R11; EC-14275, R306)

→(DRN 99-1098, R11)

the need arises to store irradiated fuel closer (than seven spaces) to Gate #3A during normal SFP operation the appropriate shielding calculations will be performed, prior to placing fuel into these cells, to ensure that dose rates, in this area, will remain acceptable.

i) At the surface of the pool from an assembly suspended 7’4” below the surface of the water (3 day cooled, 1.80 peaking factor assembly). The dose rate directly overhead is 9.07 mr/hr. The maximum dose rate, slightly away from the vertical position, is 10.8 mR/hr.

j) At the surface of the pool from typical, measured, radionuclides in the pool water. The dose rate is 0.21 mR/hr.

k) At the surface of the pool from the expected (FSAR Table 11.1-17) radionuclides in the pool water. The dose rate is 7.29 mR/hr.

l) At the surface of the pool from the maximum (FSAR Table 11.1-17) radionuclides in the pool water. The dose rate is 72.7 mR/hr.

←(DRN 99-1098, R11)

Shielding for the Fuel Pool System is based upon source terms derived from normal system operation as specified in Table 12.2-9.

## 12.3.2 SHIELDING

### 12.3.2.1 Design Objectives

The primary design objective of the plant radiation shielding is to protect plant operating personnel and the general public against radiation exposure from the reactor, power conversion, auxiliary, and waste processing systems during normal operation, including anticipated operational occurrences, postulated accident conditions, and maintenance.

This objective is accomplished by designing the shielding to perform the following functions:

a) Limit inplant exposure to radiation of plant personnel, contractors, and authorized site visitors to as far below the limits set forth in 10CFR20 as reasonably achievable for normal operation, including anticipated operational occurrences and maintenance, in conformance with Regulatory Guide 8.8 (March 1977).

b) Limit radiation exposure of main control room personnel, in the unlikely event of an accident, to allow habitability of the main control room as specified in 10CFR50, Appendix A, Criterion 19, by limiting the total integrated dose over 90 days following the accident to three rem.

→(DRN 05-144, R14)

c) Limit exposures to the general public offsite from direct and air scattered radiation to a small fraction of the limits set forth in 10CFR20 during normal operation and anticipated operational occurrences, and to within the limits specified in 10CFR50.67 for postulated accident conditions.

←(DRN 05-144, R14)

d) Provide barriers for restricting personnel access to high radiation areas and for controlling the spread of contaminants. The plant radiation shielding is also designed to protect certain plant components from excessive radiation damage or activation.

To accomplish this objective, the plant shielding functions to:

1) Reduce neutron activation of equipment, piping, supports and other materials by the use of suitable shielding around the reactor vessel, designed to minimize neutron streaming into the reactor cavity upper reaches, steam generator subcompartments, and general containment spaces.

2) Limit radiation damage to equipment and materials to below the specific integrated life dose limits.

To comply with the above objectives, the plant shielding is designed to attenuate radiation levels throughout the plant, from direct and scattered neutron and gamma radiation to the dose limits specified in Table 12.3-1.

In part, the criteria which is used in determining shielding requirements for pumps and valve galleries are the following:

a) The dose rate in the near vicinity (few ft.) from the equipment in question.

b) The annual exposure time anticipated for personnel with respect to the specific equipment. Exposure times are classified according to the modes of activity of the exposed individual relative to the equipment during the exposure period. These are:

   1) function or operation of equipment,

   2) control or surveillance of equipment, and/or

   3) maintenance of equipment.

c) Background radiation dose due to adjacent potentially radioactive equipment.

   Average exposure distances are determined for each of the exposure time modes and the resultant dose rate calculated for each distance using the average dose rates prevalent at that distance for that particular exposure mode. Typical exposure times for the various modes of exposures, i.e., operation, maintenance and repair, have been obtained from data on similar operations at other nuclear plants.

The following guidelines are applicable with respect to shielding requirements of pumps and valve galleries:

a) All pumps and valve galleries involved in the transmission of fluids (liquid and gases) of potential reactor coolant nuclide concentrations have been shielded.

b) All pumps and valves involved in the transmission of secondary system fluids, excluding resins and concentrates, have generally not been shielded.

12.3.2.2 Design Description

For shielding design purposes, the plant has been divided into radiation access zones, based on the maximum zone dose rate levels listed in Table 12.3-1.

These zone designations were used for initial shielding purposes only. Design criteria was based on regulatory dose limits applicable at the time. Specific regulations cited in zone descriptions are not applicable after January 1, 1994 due to changes to 10CFR20. Compliance with new dose limits effective January 1, 1994 will be achieved through appropriate administrative controls established in plant operating procedures.

→(LBDCR 13-010, R307)
A description of each radiation zone chosen for design purposes is given below and shown on Figures 12.3-1 through 8a.
←(LBDCR 13-010, R307)

a) Zone I
→(DRN 99-2362, R11; LBDCR 13-010, R307)
This zone has no restriction on occupancy. Such a zone would represent areas in the plant where radiation due to occupancy on a 40 hr/wk, 50 wk/yr basis, will not exceed the whole body dose of 500 mrem/yr, as specified in paragraph 20.105 of 10CFR20. Most non-employees and visitors to the site will receive considerably less than 500 mrem/yr because of the relatively short time interval during which they are onsite.
←(DRN 99-2362, R11; LBDCR 13-010, R307)

b) Zone II

This zone is a restricted radiation area which can be occupied by plant personnel and authorized visitors on a 40 hr/wk, 50 wk/yr basis without exceeding the allowable total effective dose equivalent of 5000 mrem/calendar year.

→(DRN 99-2362, R11)
c) Zone III

This is a restricted radiation area that plant personnel can occupy on a periodic basis. The average radiation level in this zone may vary from 2.5 to 15.0 mrem/hr.

d) Zone IV

This zone represents a restricted radiation area. The average radiation level may vary from 15.0 mrem/hr to 100 mrem/hr. Occupancy will be limited. However, qualified personnel who have been issued a Radiation Work Permit can enter these areas for brief periods of time to operate and inspect components. The length of stay in these areas will be determined by the Radiation Protection Staff based on the
←(DRN 99-2362, R11)

actual radiation level in the area, the past radiation history of the person entering, and the nature of the radiation.

e) Zone V

This zone represents areas with high potential for becoming high radiation areas. Areas exceeding 100 mrem/hr @ 30 $cm^2$ from the source will be posted as “CAUTION” HIGH RADIATION AREA.

12.3.2.3 Methods of Shielding Design

Shield wall thicknesses are determined by using basic shielding data and equations. Data is taken from the Table of Isotopes, "Reactor Physics Constants, ANL-5800," XDC-59-8-179, and other pertinent texts. Radiation sources are determined as indicated in Section 12.2. The method of calculation normally employed is that of the point kernel integration, outlined in Reference 1 hereafter referred to as Rockwell's method. A computer code, ISOSHLD(2), has been used for some of the actual calculations. This program calculates the decay gamma ray and Bremstrahlung dose rate at the exterior of a shielded radiation source for a number of common geometric arrangements of sources and shields such as encountered in nuclear power plants. Source geometries that can be used include: point, linear, spherical, truncated conical, disc, cylindrical, and parallel piped sources. Slab shields are used for all cases. Spherical shields can only be used for spherical sources. In addition, special computer codes such as SPAN-4(3) and MORSE-CG(4) have been employed.

The correct combination of source and shield is used to approximate the actual configuration in the plant. Tanks and large pipes containing liquid are approximated by cylindrical sources. Gas filled tanks and pipes are simulated by line sources. Small liquid carrying pipes are also approximated by line sources. Where the source shield dose point of geometry was sufficiently complex to preclude use of the Rockwell method or the ISOSHLD program, a point kernel integration, SPAN-4, was utilized. This program calculates the dose rate at a point from any number of sources having complex geometry and complex shield configurations. The geometry of the sources and shields are described by suitable intersection of quadratic surfaces. ISOSHLD, SPAN-4, and Rockwell's method account for scattering effects in the shields by using appropriate build-up factors.

Other computer codes have been utilized for updated calculations of dose rates and shielding at various points of interest. These additional computer codes utilize similar methods and geometries as the above mentioned codes. They have been verified and validated throughout industry and at W3SES to calculate both accurate and conservative radiation dose rate and shielding results.

None of the methods considers the energy degradation (softening) of the radiation spectrum as it emerges from the shield, and thus each predicts conservative values of the dose rate at the point of interest.

Whenever scattering effects were expected to be important or dominant such as in the calculation of the neutron and gamma ray streaming inside containment, the MORSE-CG, Monte Carlo code has been used. This code solves the neutron or gamma ray transport problem in arbitrary geometry by following a sufficient number of random "flight paths" of individual particles or rays through the physical system. Importance sampling was used to reduce the number of "histories" which has to be followed, by arbitrarily terminating the "history" of certain rays in regions which are not considered important. Volumetric sources were simulated by a number of point sources.

Comparison of the measured dose rates at operating plants, both BWR and PWR type with corresponding theoretically calculated dose rates, indicate that the models and method of

calculation used predict higher dose rates than actually observed. Therefore, shielding calculations based on such models and methods are conservative.

To ensure that occupational exposures would be kept as low as reasonably achievable, the shielding design has been constantly reviewed, updated, and modified as necessary during all the phases of the plant design and construction.

12.3.2.4 Compliance with Regulatory Guide 1.69

Regulatory Guide 1.69, Concrete Radiation Shields for Nuclear Power Plants, December, 1973 generally invokes ANSI Standard N101.6-1972, Concrete Radiation Shields, as an acceptable method to the NRC for the design of concrete radiation shields for nuclear power plants. Waterford 3 complies with the intent of this guide with the following exceptions and clarifications:

| ANSI N101.6-72 Section | Exceptions and Clarifications |
|---|---|
| 4.3.1 | Concrete shielding at Waterford 3 pertains to gamma and/or neutron shielding only. There are no significant sources of alpha or beta radiation within the plant which could affect concrete shield design. The maximum temperature of the primary shield wall will be 150°F. This wall will be designed to afford required shielding at this temperature. |
| 4.3.4 | "The possibility of an explosion in the cell" is not applicable to Waterford 3, since there are no explosive materials contained within shielded compartments. |
| 4.3.5 | Assumptions and methods used for accident analyses are those given in Chapter 15. These assumptions and methods result in an acceptably conservative design. |
| 4.3.6 | Regulatory Guide 8.8 is used as guidance in limiting personnel exposure and determining shielding practices. |
| 4.7 | No design drawings will be prepared specifically for formwork. Concrete design drawings are provided in sufficient detail to allow proper design of formwork according to good construction practice. Formwork specifications are provided which require conformance to ACI-347-1968. |
| 4.8 | Not applicable to Waterford 3. No heavy aggregates are used. |
| 5.1.2 | Not applicable to Waterford 3. No high density concrete is used. |
| 5.1.3 | Not applicable to Waterford 3. No hydrous aggregate is used. |
| 5.1.4 | Not applicable to Waterford 3. No boron containing aggregates are used. |

| | |
|---|---|
| 5.1.6 | Coatings of clay, silt, gypsum, calcite or caliche on coarse aggregate will total no more than three Percent of the total weight of the aggregate. Radiation attenuation calculations take this into account. |
| 5.3.4 | Not applicable to Waterford 3. No pozzolans are used. |
| 5.3.5 | Not applicable to Waterford 3. No grout fluidifiers are used. |
| 5.4 | For Waterford 3, a maximum slump of four in. is permitted for certain application: where less slump is impracticable. |
| 5.4.2 | Not applicable to Waterford 3. The preplaned-aggregate (PA) method is not used. |
| 5.4.4 | Not applicable to Waterford 3. Heavy aggregates are not used. |
| 6.1 | The use of noncombustible or fire retardant formwork for shielding; is impractical. Formwork for shielding is consistent with good construction practice and as required by ACI-347-1968. |
| 6.2.1 | ACI-347-1968 is used for the design of formwork. |
| 6.2.2 | Approval of concrete forms prior to construction is per ACI-347-1968. |

| ANSI N101.6-72 <br> Section Cont'd. | Exceptions and Clarifications |
|---|---|
| 6.4 | No substitute for detailed thermal stress analysis is made. |
| 6.5 | See position for Section 4.7 with regard to shop drawings. |
| 7.2 | See position on Section 4.7 regarding shop drawings. Any changes in specifications must be reviewed and approved prior to construction activity. The effects of supplemental tracings on shield adequacy is evaluated at that time. |
| 8.1.3 | Not applicable to Waterford 3. No high-density concrete is used. |
| 8.1.8 | Aggregate is from one source and is continually sampled throughout the construction phase for conformance to project specifications. Considering these controls, bagging and retention of samples is not necessary. |
| 8.2.6 | Vibrators having a speed of 6000 cpm are used. This speed is adequate for producing satisfactory consolidation. Sufficient spare vibrators are maintained but not necessarily one for every two being used. |
| 8.4 | Not applicable to Waterford 3. The puddling method is not used. |
| 8.6.1 | The composition and fluidity of the mortar or grout when used in pressure grouting, is specified in project specifications. |
| 8.6.2 | Filling of forms is done in accordance with good construction practice. Specifications require that no voids will be left in the concrete. |

| | |
|---|---|
| 8.7.1 | The only construction joints shown on drawings are those essential to the design of the structure. Therefore, construction joints at other locations do not require approval of the engineer responsible for the design.<br><br>Construction joints are not stepped (i.e., not provided with offsets to prevent radiation streaming). Streaming between joints is not considered to be a problem since sufficient amplitude between joints is provided. |
| 8.7.2 | Concrete is cured for the specified times. The requirements of ACI-347-1968 are not met regarding time limits for removing forms. |
| 8.7.5 | Patching and finishing is performed, as soon as practicable to ensure a quality product; however, not necessarily within the specified times. |
| 8.7.6 | Traffic or other operations is restricted after curing and finishing to prevent damage to the concrete but not necessarily for the time specified. |
| 9.1 | Only certain areas subject to contamination by radioactive substances have a protective coating. ANSI N101.4-72 and ANSI N512-72 are utilized for such applications within Waterford 3. |
| 10.1.2 | Dimensional tolerances for hatches and openings as specified in ACI-347-1968 are used rather than those given in Table 1. Minimum practicable joint clearances are specified. |
| 10-1.3 | Not applicable to Waterford 3. Service trenches are not used. |
| 10.2.2 | The weight of each block is clearly marked on the block; however, not necessarily by stenciling. |
| 10.2.3 | Blocks are cured according to good construction practice but not necessarily in the absence of direct sunlight or heat. This sunlight or heat, however, does not result in the loss of shielding efficiency. |
| 10.3.1 | There are no present plans for penetrations through shielding plugs. However, if they are required, streaming is prevented by proper design of the penetration. |
| 10.6 | All precast shielding components are fabricated at the site. If, however, offsite fabrication is used, precast shielding components are not necessarily protected from direct sunlight or high temperatures during transit or storage. This exposure is not expected to result in loss or shielding efficiency. |
| 11.5.1 | Preoperational tests of shielding are not performed. Normal post-operational tests identify any areas necessary. |

→(DRN 03-2066, R14)

11.5.2 Not applicable to Waterford 3. Containment leak testing is performed in accordance with 10CFR50 Appendix J.

←(DRN 03-2066, R14)

## 12.3.3 VENTILATION

The plant ventilation systems are designed to provide a suitable environment for personnel and equipment during normal plant operation and to provide a safe environment for operating personnel and the public during design basis accident conditions when controlling the plant to a safe shutdown conditio n.

### 12.3.3.1 Design Objectives

→(DRN 03-2066, R14)

The plant ventilation system for normal plant operation and design basis accident conditions is designed to meet the requirements of 10CFR20, 10CFR50, and 10CFR50.67.

←(DRN 03-2066, R14)

Design criteria for the plant ventilation systems include the following:

a) During normal operation and design basis accident conditions, the maximum airborne radioactive material concentrations in air breathed by personnel in restricted areas of the plant must be as low as is reasonably achievable and within the limits specified in Appendix B of 10CFR20.

   The maximum airborne radioactive material concentrations in unrestricted areas of the plant must be within the limits specified in Appendix B of 10CFR20.

b) During normal operation and design basis accident conditions, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary are as low as is reasonably achievable and within the limits specified in 10CFR20 and 10CFR50.

→(DRN 03-2066, R14)

c) The dose guidelines of 10CFR50.67 must be satisfied following postulated design basis accidents.

d) The dose to main control room personnel shall not exceed the limits specified in General Design Criterion 19 of Appendix A to 10CFR50 and 10CFR50.67.

←(DRN 03-2066, R14)

e) Airborne radioactivity monitoring is provided in compliance with General Design Criteria 63 and 64 of Appendix A to 10CFR50.

In the design of all ventilation systems, the following guidelines are used:

a) The airflow is directed from areas of low potential airborne contamination to areas of higher potential airborne contamination.

b) Airborne radiation monitoring is provided (see Subsection 12.3.4).

c) Consideration is given to the disruption of normal airflow patterns by maintenance operations. Ventilation systems are provided with back draft dampers and/or adjustable isolating dampers to allow servicing of redundant equipment without discontinuing system operation.

d) Ventilation fans and filters are provided with adequate space around the units to allow servicing and replacement of sections.

e) Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. In addition, the following concepts are used to minimize the spread of contamination:

   1) Equipment vents and drains are piped directly to the collection device connected to the collection system thus preventing spread of contamination.

   2) All welded piping systems and ductwork are employed on contaminated systems to the extent possible to reduce system leakage to a minimum acceptable level.

f) Filters containing radioactivity can be easily maintained and will not create additional radiation hazard to personnel in normally occupied areas.

12.3.3.2 Design Description

The air conditioning, heating, and ventilation systems are described for all plant buildings in Section 9.4 and Subsections 6.2.5 and 6.5.1. The aspects of the design that relate to removal of airborne radioactivity from equipment rooms, corridors and normally occupied areas are discussed under Subsection 11.3.2.

12.3.3.3 Air Cleaning System Design

Air cleaning systems are either safety-related fission product removal systems which operate following a design basis accident or non-safety related systems which control airborne radioactivity in normally occupied areas during normal operation. The central exhaust system of the Reactor Auxiliary Building Normal Ventilation System is an illustrative example of a non-safety related air cleaning system which functions during normal operation (refer to Subsection 9.4.3).

An example layout of the RAB central exhaust system housing showing filter mountings, access doors, aisle space, service galleries and provision for testing, isolation and decontamination is provided on Figures 12.3-9 and 12.3-10.

Periodic testing for filters and adsorbers will be performed after initial operation. The frequency of changeout of adsorbers will be determined from periodic testing. The frequency of changeout of filters will be determined by monitoring the filters loading levels.

### 12.3.3.4 Ventilation Systems Compliance to Regulatory Guides

The ventilation systems with atmosphere cleanup features meet the intent of Regulatory Guide 1.52, Design, Testing and Maintenance Criteria for Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, June 1973. Waterford 3 compliance to Regulatory Guide 1.52 is described in Subsection 6.5.1.

## 12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

The radiation monitoring system provided for Waterford 3 consists of the following:

a) Area Radiation Monitoring System,

b) Airborne Radiation Monitoring System,

c) Process and Effluent Radiological Monitoring and Sampling System, and

d) High Range Area Radiation Monitoring System.

The radiation monitoring systems for in-plant personnel radiation exposure determinations consist of (1) Area Radiation Monitoring System, and (2) In-Plant Airborne Radiation Monitoring System which is part of the Airborne Radiation Monitoring System.

The Process and Effluent Radiological Monitoring and Sampling System is discussed in Section 11.5. The radiation Monitoring system block diagram is shown on Figure 12.3-11.

The Area Radiation Monitoring System informs operations personnel, both locally and in the main control room, of radiation levels in areas where Area Radiation Monitoring System detectors are located, provides warning when abnormal radiation levels occur in specific plant areas, and warns of possible equipment malfunctions. Some channels of the Area Radiation Monitoring System are designed to Class 1E requirements and can withstand loss-of-coolant accident environmental conditions. Some of these Class 1E channels provide a containment purge isolation signal (refer to Subsection 7.6.1.5) in the event of abnormally high radiation inside the containment and enable main control room operators to monitor radioactivity levels inside the containment. The post-LOCA instrumentation provides information on the general direction of the accident. The High Range Area Monitors indicate radiation levels in areas requiring post-accident access or in areas with containment penetrations or hatches. In the event of a fuel handling accident, the Area Radiation Monitoring System provides a signal to isolate the Fuel Handling Building and start the emergency ventilation system.

The Airborne Radiation Monitoring System provides information, both locally and in the main control room, for the purpose of maintaining low in-plant personnel radiation exposure in accordance with 1OCFR20 and Regulatory Guide 8.8 (March 1977). It provides information on the airborne activity levels inside the control room outside air intakes and in the event of detection of high airborne activity generates a signal to isolate the normal outside air intakes and start the emergency ventilation system. The detectors assist operators in picking the one of two emergency intakes with the lowest airborne activity levels, thereby

➔ (DRN 99-2362)
minimizing the amount of noble gases entering the control room environment and also minimizing the amount of emergency ventilation system filter loading. The In-Plant Airborne Radiation Monitoring System consists of three channel, movable, radiation monitors. The monitors consist of a particulate channel, iodine channel, and noble gas channel. The primary purpose of these monitors is: (a) To assure that the in-plant personnel are not over-exposed to airborne radionuclides during maintenance of routine operations and that no localized high activity concentrations exist in any of the plant areas, and (b) to assist personnel in deciding whether or not breathing apparatus is necessary prior to entering high or airborne radiation areas. Monitors comprising the in-plant Airborne Radiation Monitoring and the ranges are given in Table 12.3-3..
← (DRN 99-2362)
Regulatory Guide 8.10 (September 1975) is discussed in Section 12-1.

12.3.4.1 <u>Area Radiation Monitoring System</u>

12.3.4.1.1 Design Objectives

The objectives of the Area Radiation Monitoring System during normal operating plant conditions and anticipated operational occurrences are:

a) to measure ambient gamma radiation and to indicate to operations personnel the ambient gamma radiation in specific areas of the plant,

b) to annunciate and warn of abnormal radiation levels in specific areas of the plant,

c) to furnish records of radiation levels in specific areas of the plant,

d) to provide base data in controlling access of personnel to radiation areas,

e) to warn of uncontrolled or inadvertent movement of radioactive material in the plant,

f) to provide local indication and alarms at key points where a substantial change in radiation levels might be of immediate importance to personnel frequenting the area,
➔ (DRN 99-2362)
g) to assist operations and plant personnel in decisions on deployment of personnel in the event of an accident or equipment malfunction resulting in a release of radioactive material in the plant,
← (DRN 99-2362)
h) to annunciate and warn of possible equipment malfunctions in specific areas of the plant,

i) to furnish information for making radiation surveys, and

j) to assist the supervisors in planning work schedules.

The objectives of the Area Radiation Monitoring System during postulated accidents are:

→ (DRN 99-2362)

a) provide the capability to alarm and initiate a containment purge isolation signal in the unlikely event of a loss-of-coolant accident, fuel handling accident, or abnormally high radiation inside the containment, (Note, CIAS picks up LOCA indications before CPIS does)

← (DRN 99-2362)

b) provide long term post-accident monitoring of conditions inside the containment, and

c) provide a signal to isolate the Fuel Handling Building and start the emergency ventilation system in the event of a fuel handling accident.

12.3.4.1.2 Criteria for Location of Monitors

Considerations for area monitor locations are based on the following:

a) frequency and length of personnel occupancy of a specific area,

b) potential for personnel unknowingly to receive high radiation doses,

c) potential for equipment malfunction,

d) areas where during normal plant operation including refueling, radiation exposures could exceed the radiation limits due to system failure or personnel error,

e) areas where new and spent fuel is received and stored,

f) containment area for indicating the level of radioactivity and detecting the presence of fission products due to a reactor coolant pressure boundary leak, and

→ (DRN 99-2362)

g) normally or potentially radioactive release areas.

← (DRN 99-2362)

12.3.4.1.3 System Description

The Area Radiation Monitoring channels are located at selected places inside the plant to detect and store information on the radiation levels and, if necessary, annunciate abnormal radiation conditions. The areas where the gamma monitors are located are shown in Table 12.3-2. Indication, annunciation and storage is provided for all 39 channels in the main control room and other cathode ray tubes (CRTS) and computer magnetic storage tape. The instrument locations, ranges, sensitivities, accuracies and alarm set points are shown in Table 12.3-2. Environmental design conditions are discussed in Subsection 11.5.2.3.

A typical channel consists of a gamma sensitive Geiger-Muller (GM) or ion chamber detector, a microprocessor, power supply, a local indicator and audio-visual alarm and a check source or test current. The system utilizes local microprocessors with inputs to the radiation monitoring computers for purposes of data logging, processing, editing and displaying of information obtained from the radiation sensors. This microprocessor approach provides considerable flexibility in the means of collecting data and manners of displaying and utilizing such data.

All channel information is processed through a dedicated microprocessor which is then interrogated by the radiation monitoring computer for processing, indications on CRTS, storage, alarming and hard copy production if so desired

→(EC-12329, R306)

Those channels identified on Table 12.3-2 as safety related are first indicated and recorded on digital ratemeters and recorders housed on the radiation monitoring panels in the main control room as shown on Figure 12.3-11. Through a properly qualified isolation buffer the signal is then transmitted to the Radiation Monitoring System computers for the kind of processing mentioned above. Upon a seismic event where the Radiation Monitoring System computers and peripherals are presumed to fail, the safety related channels maintain their function ability.

←(EC-12329, R306)

The detectors are wall-mounted gamma sensitive Geiger-Muller tubes or ion chambers. Their energy dependence is typically flat (within ± 20 percent) from 80 KeV to 1.5 MeV and each GM detector is provided with an integral check source, operated from the local microprocessor or the CRT in the main control room. The ion chamber area monitors have a test current to test circuit integrity.

Each monitor channel provides three alarms. One alarm level is set high enough above the normal measured radiation levels (background) in the area to prevent spurious alarms, yet low enough to indicate transient radiation level increases. A second alarm is set at a higher level. A third alarm acts as a tube failure, circuit failure or cable disconnect alarm. On occasion, other alarm points may be selected depending either upon work in progress in the area or operations that will vary the normal (background) measured radiation levels in the area. Some monitors have different alarm points when the reactor is critical than when the reactor is shut down. Plant chemistry personnel approve the alarm points consistent with radiological safety controls for the area.

→(DRN 99-2362, R11, 00-801

The instruments are calibrated and maintained on a routine schedule, as discussed in the Off Site Dose Calculation Manual, Technical Requirements Manual, the Technical Specifications, or plant procedures. All alarms initiate continuous audible and visual alarms at the detector. The tone and volume of the local audible alarm varies in intensity to be easily heard in operating areas. The radiation monitoring computer through the PMC provides alarms for any channel detecting high radiation levels. Identification of the channel alarmed is done at the CRT in the main control room and –4 Access Control point office.

←(DRN 99-2362, R11, 00-801)

Area radiation monitoring channels 24 through 31 have a 1,500 volt isolation buffer between the microprocessor and the radiation monitoring computer. Upon a postulated seismic event, all non-seismic channels, including the micro processor, and Radiation Monitoring Computers are presumed to fail. The buffer, designed in accordance with Reactor Research and Development Standards C163T-1971, Plant Protection System Buffers, and Sections 4.3.2 and 4.3.3 of C16-1T-1969, Supplementary Criteria and Requirements for RDT Reactor Plant Protection System, isolates the essential channels and maintains the signal flow to the ratemeters in the main control room. The main control room panel housing all safety-related ratemeters is qualified to seismic Category I requirements.

Area radiation monitoring channels 24 through 27 (ion chambers) (Containment Purge Isolation Detectors) are designed to Class 1E requirements and can withstand a LOCA environment for a period of at least 10 min. after the accident. Channels 24 through 27 are powered from two 120 V ac nuclear instrumentation buses SA and SB and are arranged in two groups with two monitors in each group. The output signals from these monitors make up

part of the containment purge isolation signal. Channels 28 through 31 are designed to Class 1E requirements. These are powered by two 120 V AC vital buses (SA and SB). All other monitors' power supplies feed from an interruptible source and become inoperative during a loss of offsite power.

In addition to qualification to Class 1E requirements, channels 24 through 31 are physically and electrically separated from each other in accordance with the criteria set forth in IEEE 279-1971 and IEEE 308-1971.

A retractible solenoid operated radioactive check source provides a means of checking the integrity of GM tube detector area radiation monitors. A test current provides a means of checking the integrity of ion chamber detector area radiation monitors.

The microprocessor and computer receives, processes and displays information on request. Three alarms are provided: one for alert radiation, the second for high radiation and a third for when a channel becomes inoperative. The microprocessor has the ability to activate the check source into position or activate the test current for operational check purposes.

→ (DRN 99-2362)

Four redundant fuel pool monitors are provided to detect radioactivity in the event of a fuel handling accident in the Fuel Handling Building. In the event of a refueling accident, four GM tubes positioned above the fuel pool area will sense the radioactivity released and will supply a signal for the startup of the Fuel Handling Building Ventilation System (only emergency portion, refer to Subsection 9.4.2) as well as the closure of isolation dampers in the normal ventilation system. These monitors are designed to Class IE requirements and in accordance with IEEE-279-1971, IEEE-308-1971, and IEEE-344-1971. The calibration performed is in accordance with vendor supplied transfer calibration procedures. Recalibration will be performed at periodic intervals as set by plant procedures.

← (DRN 99-2362)

12.3.4.2 Airborne Radiation Monitoring System

12.3.4.2.1 Design Objectives

The objectives of the Airborne Radiation Monitoring System during normal operating plant conditions and anticipated operational occurrences are:

a) to inform operations personnel of airborne particulate, gaseous and iodine ($\beta$-$\gamma$, gross $\beta$, and $\gamma$, respectively) activity trends in the various buildings and structures of the plant,

→ (DRN 99-2362)

b) to alarm in case of abnormal increase in the airborne activity levels,

← (DRN 99-2362)

c) to furnish records of gross airborne trends in the various plant areas and of the amount of radioactive releases to the environment through the plant buildings or structures during normal, or abnormal operational occurrences,

d) to help detect identified or unidentified leaks inside the reactor coolant pressure boundary (as recommended in Regulatory Guide 1.45, May 1973) and other areas of the plant,

→ (DRN 99-2362)

e) to assist personnel in deciding whether or not breathing apparatus is necessary when entering a high airborne radiation area, and

← (DRN 99-2362)

f) to provide information for evaluation of the performance of all plant systems that function to minimize the release of radioactivity to accessible areas of the plant and to the environment.

The objective of the main control room airborne radiation monitoring system during postulated accidents is to provide the capability to alarm and initiate isolation of the main control room normal ventilation system and actuate the emergency ventilation system in the unlikely event that radioactivity is introduced into the main control room intake ductwork.

12.3.4.2.2 Criteria for Location of Monitors

Considerations for locating the Airborne Radiation Monitoring System monitors are based on the following:

a) paths that normally, or potentially, may release airborne radioactivity to the environment,

b) areas where the airborne radioactivity can abruptly increase and where personnel normally have access to the areas,

c) in ventilation ducts where the monitors can survey, at large, the airborne radioactivity level, and

d) inside the containment for the purpose of monitoring unidentified leaks.

12.3.4.2.3 System Description

Airborne radioactivity detection devices are provided in the plant to monitor normal radiation levels and to detect and annunciate any abnormal radiation conditions. The Airborne Radiation Monitoring System consists of monitors for the containment, the main control room, as well as certain areas in the Reactor Auxiliary Building (RAB). Table 12.3-3 lists the Airborne Radiation Monitoring System monitors including the locations, number of channels, type, range, sensitivity, accuracy, and alarm set points. Environmental design conditions are discussed in Subsection 11.5.2.3.

→ (DRN 99-2362)

The RAB is an area which has the possibility of containing airborne contamination when personnel are present. Calculations were performed to determine the capability of the airborne radioactivity monitors in the RAB to detect 1 DAC in air of particulate or iodine radioactivity in one hour in any compartment sampled by these monitors. Based on the calculations, monitor PRM-IR-6710D cannot satisfy the subject detection requirement for those compartments with flowrates less than 198 cfm for iodine and 88 cfm for particulates. In those compartments, during extended maintenance where the 1 DAC in one hour requirement cannot be met, administrative procedures will provide guidance.

← (DRN 99-2362)

The guidance of regulatory position paragraph C.2.g of Regulatory Guide 8.8 (March 1977) has been factored into the design of the Airborne Radiation Monitoring System.

This system also utilizes the microprocessor approach as described in Subsection 12.3.4.1.3. Additional alarms are provided for malfunction of the particulate and iodine filters and high pressure and low pressure across the filters.

12.3.4.2.3.1 Containment Atmosphere Radiation Monitor

The containment atmosphere radiation monitor is designed to provide a indication in the main control room of the particulate, iodine and gaseous radioactivity levels inside the containment. Radioactivity in the containment atmosphere indicates the presence of fission products due to a reactor coolant pressure boundary leak.

The containment atmosphere sample is drawn into the monitoring assembly by a one in. stainless steel sampling line as shown in Figure 12.3-15. The flow sample is distributed to the iodine and the particulate samplers by a V pipe connection. The iodine sample is obtained when the sample passes through a fixed charcoal filter bed. The fixed charcoal filter bed is then monitored by a gamma sensitive scintillation detector. The particulate sampler collects particles greater than or equal to 0.3 microns on a moving paper filter. A beta sensitive scintillation detector aimed at the moving paper filter monitors for particulate radiation. After passing through particulate and iodine filters, the sample will be dried in a moisture control unit and then monitored for radioactive gas content in the gas sampler. The sample lines are kept as short as possible and horizontal runs are minimized to reduce plate-out and losses due to gravity deposition prior to iodine and particulate monitoring. The moisture control unit can be isolated from the radiation monitor via isolation valves installed between the two units.

→(DRN 06-1029, R15)

The pumping system consists of two sample pumps; one pump draws air through the iodine and gas monitors and the other draws air through the particulate monitor. Flowmeters are placed downstream of each pump, to indicate flow rates and to alarm abnormal flow rates. Mass flow probes control the flow of the monitor pathways via the monitor microprocessor. The mass flow probes also signal alarms in the microprocessor when a sudden change in flow takes place indicating abnormal filter function in the particulate and iodine monitors. The isolation of the noble gas monitor allows remote purging with clean air for background checkup and maintenance. The air stream bypasses the noble gas monitor whenever it is isolated by its isolation valves. The purpose of having the air stream bypass the noble gas monitor is to allow operation of the iodine and particulate monitors when the noble gas monitor is undergoing testing or maintenance, thus achieving additional system flexibility.

←(DRN 06-1029, R15)

A containment isolation actuation signal will isolate this monitor from the containment.

The iodine filter is periodically replaced and can be analyzed in the laboratory. Adequate shielding is arranged in a $4\pi$ geometry around the detectors to prevent interference from background radiation and electromagnetic fields.

A solenoid operated check source is provided for verifying detector operation.

Each channel provides three alarm modes; FAIL, ALERT and HIGH. ALERT and HIGH alarms are adjustable over the full span of the scale. One alarm is set to alarm for detector signal failure, power failure, or from a failure due to a disconnected cable. The second is set to alert that a specified radiation level has been exceeded. The third is set at a higher level to alarm at higher radiation levels.

The containment atmosphere radiation monitor is designed to Class 1E requirements and in accordance with IEEE-279-1971, IEEE-308-1971 and IEEE-344-1971.

Indication devices are located on the main control room radiation monitoring panel. All radiation channels are indicated, stored, as required, and annunciated on the main control room radiation monitoring panel. Abnormal radiation levels are indicated both visually and audibly, locally and in the main control room. Each detector is calibrated at the factory using two or more National Institute Standards Technology calibrated isotopes prepared in the proper form to simulate the effluent for which the system will be used.

→ (DRN 99-2362)

A recalibration of the detectors will be performed at periodic intervals as set by the Offsite Dose Calculation Manual, Technical Requirements Manual, the Technical Specifications, or plant procedures. Calibration is performed in accordance with vendor supplied transfer calibration procedures.

← (DRN 99-2362)

As described in Subsection 11.5.2.1, the following items are applicable to the containment atmosphere radiation monitor: monitor cabinet, sampler, check source, detector assembly, recorders and power supplies.

12.3.4.2.3.2 Plant Stack Radiation Monitor

The plant stack radiation monitor is designed to representatively sample, monitor, indicate and store the radioactivity levels in the plant effluent gases being discharged from the plant stack. It provides a continuous indication in the main control room of the activity levels of radioactive materials released to the environs so that determination of the total amount of activity release is possible. These monitors have been designed to the applicable requirements of Regulatory Guide 1.21 (June 1974), regulatory position paragraphs C.2 through C.11.

A schematic diagram of the plant stack radiation monitor is shown on Figure 12.3-13.

→ (DRN 99-2362)

The plant stack radiation monitor monitors the plant stack for particulates, iodine and noble gases at the point of release to the atmosphere. Its function is to confirm that releases of radioactivity do not exceed the predetermined limits set by the Offsite Dose Calculation Manual (ODCM) or Technical Requirement Manual (TRM).

← (DRN 99-2362)

The sample flow is withdrawn from the stack through an isokinetic nozzle located at a minimum of eight stack diameters from the last point of radioactivity entry. The nozzles are designed such that the sampling velocity is the same as that in the stack pipe so that preferential selection does not occur, i.e., so that the weights of the radioactive particles do not become a factor in obtaining a representative sample. The isokinetic sampling system is designed in accordance with ANSI N13.1-1969.

The particulate iodine and gaseous detectors used for each plant stack monitor have the same technical description as those for the containment atmosphere radiation monitor described above (with the exception of a moisture control unit). The calibration performed

is similar to that for the containment atmosphere radiation monitor, also discussed in that subsection. As described in Subsection 11.5.2.1, the following items are applicable to the containment atmosphere radiation monitor; monitor cabinet, sampler, check source, detectors assembly, recorders and power supplies.

12.3.4.2.4 Main Control Room Radiation Monitors

Two redundant pairs of radiation monitors, as shown on Figure 12.3-14, are provided for monitoring radioactive airborne concentration levels inside the two outside air intake plenums. Each of the four monitors consists of two separate plastic scintillator detectors; one detector with a beta shield and one without. The detector with a beta shield is not used. It is an installed spare for the beta-gamma detector. The detector with the beta shield does not have any alarm functions. All alarms are applicable for the beta-gamma detector only. Using the beta-gamma detector operators are able to pick the air intake with the lowest airborne concentration levels.

Under normal operation the control room ventilation system draws air from the north side intake plenums. Upon detection of high radiation in any one of the radiation detectors, a signal is developed to isolate the control room. The operator shall examine the radiation monitor outputs to determine which intake point has the lowest radiation level and choose the intake point with the lowest airborne contamination level to pressurize the control room.

These monitors are designed to Class 1E requirements and in accordance with IEEE 279-1971, IEEE 308-1971, and IEEE 344-1971.

The calibration performed is in accordance with vendor supplied transfer calibration procedures.

12.3.4.2.5 Other Airborne Radiation Monitors

→(DRN 02-407)

Other airborne radiation monitors exist in the plant as shown in Table 12.3-3. Of those listed, the hot machine shop and the decontamination facility monitors are movable carts.

←(DRN 02-407)

12.3.4.3 <u>High Range Area Radiation Monitoring System</u>

12.3.4.3.1 Design Objectives

The objectives of the High Range Area Radiation Monitoring System during and following postulated accidents are:

a) To indicate radiation levels in area with containment penetrations or hatches to provide indication of breach, and

b) To indicate radiation exposure rate in areas where access is required to service equipment post-accident.

12.3.4.3.2 System Description

The High Range Area Radiation Monitoring channels are located at selected places inside the plant to detect and store information on the radiation levels and, if necessary, annunciate abnormal radiation conditions. The area where the gamma monitors are located are shown in Table 12.3-2.

Indication, annunciation and storage is provided for all 11 channels in the main control room and other cathode ray tubes (CRTS) and computer magnetic storage tape. The instrument locations, ranges, sensitivities, accuracies and alarm setpoints are shown in Table 12.3-2.

For the High Range Area Radiation Monitoring System, a typical channel consists of a gamma sensitive ion chamber detector, a microprocessor, power supply, a local indicator and audio-visual alarm and a test current. The system utilizes local microprocessors with inputs to the radiation monitoring computers for purposes of data logging, processing, editing and displaying of information obtained from the radiation sensors. This microprocessor approach provides considerable flexibility in the means of collecting data and manners of displaying and utilizing such data.

All channel information is processed through a dedicated microprocessor which is then interrogated by the radiation monitoring computer for processing, indications on CRTS, storage, alarming and hard copy production if so desired.

The detectors are wall-mounted gamma sensitive ion chambers. Their energy dependence is typically flat (within ± 20 percent) from 60 keV to 3 MeV and each monitor is provided with a test current, operated from the local microprocessor or the CRT in the main control room.

Each monitor channel is provided with three alarms. One alarm level is set high enough above the normal measured radiation levels in the area to prevent spurious alarms, yet low enough to indicate transient radiation level increases. A second alarm is set at a higher level. A third alarm acts as a tube failure, circuit failure or cable disconnect alarm. On occasion, other alarm points may be selected depending upon work in progress in the area or operations that will vary the normal measured radiation levels in the area. Some monitors are expected to have different alarm points when the reactor is critical then they will have when the reactor is shut down. Plant chemistry personnel will specify the alarm points consistent with radiological safety controls for the area.

→ (DRN 99-2362, 00-801)

The instruments are calibrated and maintained on a routine schedule. All alarms initiate continuous audible and visual alarms at the detector. The tone and volume of the local audible alarm is of variable intensity to be easily heard in operating areas. The radiation monitoring computer through the PMC provides alarms for any channel detecting high radiation levels. Identification of the channel alarmed is done at the CRT in the main control room and the –4 Access Control Point Office.

← (DRN 99-2362, 00-801)

A means of checking the integrity of the High Range Area Radiation Monitoring System is accomplished through the use of a test current.

The microprocessor and computer receives, processes and displays information on request. Three alarms are provided: one for high radiation, the second for high-high radiation and a

third for when a channel becomes inoperative. The microprocessor has the ability to activate the test current for circuit verification purposes.

SECTION 12.3: REFERENCES


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10. Florida Power & Light. "St. Lucie Unit 1 Docket No. 50-355 License Condition D, Neutron Streaming Shield," April 1977.

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→ (DRN 99-1098)

13. Waterford 3 Rerack Radiological and Shielding Calculations, Holtec Report HI-971735, Calculation EC-M98-021.

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15. SAS2H-ORIGEN-S, in Scale 4.3-Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG-CR-0200, Rev. 5, September 1995 (Issued by Radiation Shielding Information Center).

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← (DRN 99-1098)

TABLE 12.3-1 Revision 308 (11/14)

ALLOWABLE DOSE RATES

| Location (Area) | Dose Rate mrem/hr (max) |
|---|---|
| Site Boundary<br>→(DRN 99-2362, R11) | 0.05 |
| ←(DRN 99-2362, R11)<br>Outside Tool Room | 0.05 |
| Maintenance Support Building | 0.05 |
| Service Building<br>→(DRN 99-2362, R11) | 0.05 |
| Main Control Room, Turbine Building, Westside Access Enclosure and Reactor Auxiliary Building areas outside controlled access areas<br>←(DRN 99-2362, R11) | 0.25 |
| Reactor Auxiliary Building and Fuel Handling Building areas inside controlled access areas: areas of 40 hr/wk occupancy | 2.5 |
| Reactor Auxiliary Building, Fuel Handling Building and containment: areas of six hr/wk occupancy | 15.0 |
| Reactor Auxiliary Building, Fuel Handling Building and containment: areas where occupancy is determined by health physics staff | 100 |
| High radiation areas | >100 |
| →(DRN 03-2066, R14)<br>Main control room following maximum hypothetical accident<br>← (DRN 03-2066, R14) | 5 rem TEDE over 30 days following the accident |
| →(LBDCR 13-009, R307)<br>Low Level Radwaste Storage Facility (LLRWSF) Restricted Area Boundary<br>←(LBDCR 13-009, R307) | 0.6 mrem/hr |
| →(LBDCR 13-010, R307)<br>Original Steam Generators Storage Facility At the fence boundary<br>←(LBDCR 13-010, R307) | 0.05 |
| →(LBDCR 14-007, R308)<br>Entrapment Area<br>←(LBDCR 14-007, R308) | 0.05 |

TABLE 12.3-2 (Sheet 1 of 4) Revision 11-A (02/02)

AREA RADIATION MONITORS

| Channel | | Location[3] | Safety Classification | Range (mR/hr) | Sensitivity (cpm/mR/hr) | Accuracy (%) | Maximum Alarm Setpoint | Local Alarm | Background Radiation (mR/hr) |
|---|---|---|---|---|---|---|---|---|---|
| | | Reactor Auxiliary Bldg. (RAB) | | | | | | | |
| ARM-IR-5001[4] | RE-1 | Main Control Room | Non-safety | 0.01-1000 | 104 | ± 20[1] | 1.0 | Yes | 0.25 |
| ARM-IR-5002 | RE-2 | EL. +46 ft. MSL, Column K-5A | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5003 | RE-3 | EL. +21 ft. MSL, Column H-4A | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5004 | RE-4 | EL. -4 ft. MSL, Column K-2A | Non-safety | 0.01-1000 | 104 | ± 20 | 7.5 | Yes | 2.5 |
| ARM-IR-5005 | RE-5 | EL. -4 ft. MSL, Wall H bet. Column 4A and 5A | Non-safety | 0.01-1000 | 104 | ± 20 | 30.0 | Yes | 15 |
| ARM-IR-5006 | RE-6 | Counting Room, EL. -4 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 1.0 | Yes | 0.25 |
| ARM-IR-5007 | RE-7 | Sampling Room, EL. -4 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 1.0 | Yes | 0.25 |
| ARM-IR-5008 | RE-8 | Boric Acid Precon, filters - EL. -35 ft. MSL (Column H-3A) | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5009 | RE-9 | Waste Filters Wall, EL. -35 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ➜ (DRN 99-2362; 00-1046) | | | | | | | | | |
| ARM-IR-5016 | RE-16 | Hot Tool Room, Column H-2A | Non-safety | 0.01-1000 | 36.7 | ± 20 | 5.0 | Yes | 2.5 |
| ← (DRN 99-2362; 00-1046) | | | | | | | | | |
| ARM-IR-5017A | RE-17 | Charging Pumps Wall, EL. -35 ft. MSL | Non-safety | $0.1\text{-}10^4$ | 36.7 | ± 20 | 30.0 | Yes | 15 |
| ARM-IR-5019 | RE-19 | Elevator Shaft, EL. +21 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5020 | RE-20 | Radio Chemistry Lab bet. Columns 10A and 11A, EL. -4 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5021 | RE-21 | Corridor near Valve Gallery (Column 4A) EL. -4 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5022 | RE-22 | Corridor near Filter flush Tank (Wall 6A-L) EL. -4 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |

TABLE 12.3-2 (Sheet 2 of 4) Revision 7 (10/94)

AREA RADIATION MONITORS

| Channel | | Location(3) | Safety Classification | Range (mR/hr) | Sensitivity (cpm/mR/hr) | Accuracy (%) | Maximum Alarm Setpoint | Local Alarm | Background Radiation (mR/hr) |
|---|---|---|---|---|---|---|---|---|---|
| ARM-IR-5022B | RE-22B | EL. -35ft. MSL, South of Column H-4A | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5022C | RE-22C | EL. -4 ft. MSL, West of Column H-3A | Non-safety | 0.01-1000 | 104 | ± 20 | 30.0 | Yes | 15 |
| ARM-IR-5023 | RE-23 | Corridor near Gas Decay Tanks (Column K-4A) EL. -35 ft. MSL | Non-safety | 0.01-1000 | 104 | ± 20 | 30.0 | Yes | 15 |
| ARM-IR-5023A | RE-23A | Decontamination Room EL. +21 ft. MSL, near Column 3A-H | Non-safety | 0.01-1000 | 104 | ± 20 | 5.0 | Yes | 2.5 |
| ARM-IR-5200 | | Recirc. Penetration Area EL. -4 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5201 | | Personnel Air Lock Area #1; EL. -4 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 2.5 |
| ARM-IR-5202 | | SIS Sump Penetration Area; EL. -4 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5204 | | Post Accident Sampling System Area; EL. -4 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5205 | | DG Room 3A-S EL. +21 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5206 | | DG Room 3B-S EL. +21 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5207 | | Electrical Equipment Area EL. +21 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5208 | | Component Cooling Water Area EL. -4 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 0.25 |
| ARM-IR-5209 | | Waste & Boron Control Panel Area EL -4 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 30.0 |
| ARM-IR-5210 | | Turbine Driven EFW Pump Area; EL. -35 ft. | Non-safety | $10^2$-$10^7$ | $1.2x10^{-10}$ A/R/hr | ± 20 | $1x10^3$ | Yes | 30.0 |

TABLE 12.3-2 (Sheet 3 of 4) Revision 10 (10/99)

AREA RADIATION MONITORS

| Channel | | Location(3) | Safety Classification | Range (mR/hr) | Sensitivity (cpm/mR/hr) | Accuracy (%) | Maximum Alarm Setpoint | Local Alarm | Background Radiation (mR/hr) |
|---|---|---|---|---|---|---|---|---|---|
| | | Reactor Bldg. (RB) | | | | | | | |
| *ARM-IR-5013 | RE-13 | Refueling Machine (during shutdown only) | Non-safety | $0.1\text{-}10^4$ | 36.7 | ± 20 | 2 x Bkg | Yes | 15,000n:1,000 (power operation), 2.5 (shutdown) |
| *ARM-IR-5014 | RE-14 | Southwest Staircase | Non-safety | $0.1\text{-}10^4$ | 36.7 | ± 20 | 2 x Bkg | Yes | 1,000n:1,000 |
| *ARM-IR-5015(4) | RE-15 | Northeast Staircase | Non-safety | $0.1\text{-}10^4$ | 36.7 | ± 20 | 2 x Bkg | Yes | 1,000n:1,000 |
| *ARM-IR-5018 | RE-18 | EL. -4 ft. MSL, Personnel Lock | Non-safety | $0.1\text{-}10^4$ | 36.7 | ± 20 | 2 x Bkg | Yes(2) | 15n:10 |
| ARM-IR-5024S | RE-24 | North side of SG Shield Wall, EL. +46 ft. MSL (containment purge isolation) | 1E | $20\text{-}5\times10^5$ | $7\times10^{-10}$ A/R/hr | ± 20 | 2 x Bkg | Yes | 2,000n:10,000 |
| ARM-IR-5025S | RE-25 | South side of SG Shield Wall, EL. +46 ft. MSL (containment purge isolation) | 1E | $20\text{-}5\times10^5$ | $7\times10^{-10}$ A/R/hr | ± 20 | 2 x Bkg | Yes | 2,000n:10,000 |
| ARM-IR-5026S | RE-26 | Southwest side of SG Shield Wall, EL. +21 ft. MSL (containment purge isolation) | 1E | $20\text{-}5\times10^5$ | $7\times10^{-10}$ A/R/hr | ± 20 | 2 x Bkg | Yes | 300n:2,000 |
| ARM-IR-5027S | RE-27 | North side of SG Shield Wall, EL. +21 ft. MSL | 1E | $20\text{-}5\times10^5$ | $7\times10^{-10}$ A/R/hr | ± 20 | 2 x Bkg | Yes | 300n:2,000 |
| ARM-IR-5028S | RE-28 | Outside RB Shield Wall (105°F), EL. +46 ft. MSL (Post-LOCA) | 1E | $10^1\text{-}10^5$ | 36.7 | ± 20 | None | Yes | 0.25 |
| ARM-IR-5029S | RE-29 | Outside RB Shield Wall (230°F), EL. +21 ft. MSL (Post-LOCA) | 1E | $10^1\text{-}10^5$ | 36.7 | ± 20 | None | Yes | 2.5 |
| ARM-IR-5030S | RE-30 | Outside RB Shield Wall (115°F), EL. +21 ft. MSL (Post-LOCA) | 1E | $10^1\text{-}10^5$ | 36.7 | ± 20 | None | Yes | 0.25 |

*Used by HP during shutdown for personnel protection. Not required during normal operation.

TABLE 12.3-2 (Sheet 4 of 4) Revision 10 (10/99)

AREA RADIATION MONITORS

| Channel | | Location(3) | Safety Classification | Range (mR/hr) | Sensitivity (cpm/mR/hr) | Accuracy (%) | Maximum Alarm Setpoint | Local Alarm | Background Radiation (mR/hr) |
|---|---|---|---|---|---|---|---|---|---|
| ARM-IR-5031S | RE-31 | Outside RB Shield Wall (240°F), EL. +46 ft. MSL (Post-LOCA) | 1E | $10^1$-$10^5$ | 36.7 | ± 20 | None | Yes | 0.25 |
| | | Fuel Handling Bldg. (FHB) | | | | | | | |
| ARM-IR-5010 | RE-10 | EL. +50 ft. MSL, Column U-7FH | Non-safety | 0.01-1000 | 104 | ± 20 | 20 | Yes | 2.5 |
| ARM-IR-5011 | RE-11 | EL. +46 ft. MSL, New Fuel Vault | Non-safety | 0.01-1000 | 104 | ± 20 | 20 | Yes | 2.5 |
| ARM-IR-5012 | RE-12 | Fuel Pool Pump's Hall Wall near Equipment Hatch | Non-safety | 0.01-1000 | 104 | ± 20 | 30 | Yes | 15 |
| ARM-IR-0300.1S | RE-32 | FHB Adjacent to Spent Fuel Pool | 1E | 0.1-$10^4$ | 36.7 | ± 20 | $10^3$ | Yes | 2.5 |
| ARM-IR-0300.2S | RE-33 | FHB Adjacent to Spent Fuel Pool | 1E | 0.1-$10^4$ | 36.7 | ± 20 | $10^3$ | Yes | 2.5 |
| ARM-IR-0300.3S | RE-34 | FHB Adjacent to Spent Fuel Pool | 1E | 0.1-$10^4$ | 36.7 | ± 20 | $10^3$ | Yes | 2.5 |
| ARM-IR-0300.4S | RE-35 | FHB Adjacent to Spent Fuel Pool | 1E | 0.1-$10^4$ | 36.7 | ± 20 | $10^3$ | Yes | 2.5 |
| ARM-IR-5203 | | Refueling Canal Area, EL. +46 ft. | Non-safety | $10^2$-$10^7$ | $1.2\times10^{-10}$ A/R/hr | ± 20 | $10^3$ | Yes | 2.5 |

(1) Of indicated field intensity due to a combined action of ± 20% voltage variation and environmental design conditions as discussed in Subsection 11.5.2.3.
(2) Outside containment on platform.
(3) Elevations are floor levels.
(4) LP&L tag numbers.

TABLE 12.3-3 Revision 14 (12/05)

AIRBORNE RADIATION MONITORS

| Numbers | Location | Type | Number of Channels | Range (μci/cc) | Sensitivity (μci/cc) | | Ratemeter Accuracy | Maximum Alarm Setpoint (μci/cc) | Background Radiation (mR/hr) |
|---|---|---|---|---|---|---|---|---|---|
| →(DRN 05-455, R14) | | | | | | | | | |
| 1. | Containment Atmosphere (PRM-IR-0100Y)*** | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.56X10^{-12}$<br>$1.75x10^{-10}$<br>$1.85x10^{-7}$ | (Cs-137)<br><br>(Xe-133) | ± 20% of scale | $3.01E^{-10}$<br>$3.39E^{-8}$<br>$1.33E^{-2}$ | 2.5 |
| ←(DRN 05-455, R14) | | | | | | | | | |
| 2. | Main Control Room, at outside air intakes (PRM-IR-0200.1, -0200.2, 0200.5, -0200.6 | β Scintillator | 4 | $10^{-8}$-$10^{-2}$ | $3.56X10^{-4}$ | (Kr-85) | ± 20% of scale | 5.45 $E^{-6}$ | 0.25 |
| 3. | Hot Machine Shop (PRM-IR-5132)*** | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.56X10^{-12}$<br>$1.75X10^{-10}$<br>$1.85X10^{-7}$ | (Cs-137) | ± 20% of scale | $6E^{-8}$<br>$2E^{-8}$<br>$1E^{-4}$ | 2.5 |
| 4. | Decontamination Facility (PRM-IR-5144)*** | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.56X10^{-12}$<br>$1.75X10^{-10}$<br>$1.85X10^{-7}$ | (Cs-137)<br><br>(Xe-133) | ± 20% of scale | $6E^{-8}$<br>$2E^{-8}$<br>$1E^{-4}$ | 2.5 |
| 5. | RAB Monitor A - (PRM-IR-6710A)*** located in ductwork on EL. -4 ft. MSL at Column J-4A | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.56X10^{-12}$<br>$1.75X10^{-10}$<br>$1.85X10^{-7}$ | (Cs-137)<br><br>(Xe-133) | ± 20% of scale | $6E^{-8}$<br>$2E^{-8}$<br>$1E^{-4}$ | 2.5 |
| 6. | RAB Monitor B - (PRM-IR-6710B)*** located in ductwork on EL. +21 ft. MSL west of Column M-5A | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.56X10^{-12}$<br>$1.75X10^{-10}$<br>$1.85X10^{-7}$ | (Cs-137)<br><br>(Xe-133) | ± 20% of scale | $6E^{-8}$<br>$2E^{-8}$<br>$1E^{-4}$ | 2.5 |
| 7. | RAB Monitor C - PRM-IR-6710C)*** located in ductwork on EL. -4 ft. MSL column J-6A | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.56X10^{-12}$<br>$1.75X10^{-10}$<br>$1.85X10^{-7}$ | (Cs-137)<br><br>(Xe-133) | ± 20% of scale | $6E^{-8}$<br>$2E^{-8}$<br>$1E^{-4}$ | 2.5 |
| 8. | RAB Monitor D - (PRM-IR-6710D)*** located in ductwork on EL. +46 ft. MSL Column K-4A | Particulate<br>Iodine<br>Gaseous | 3 | $10^{-11}$-$10^{-5}$<br>$10^{-9}$-$10^{-3}$<br>$10^{-7}$-$10^{-1}$ | $1.546X10^{-12}$<br>$1.75X10^{-10}$<br>$1.85X10^{-7}$ | (Cs-137)<br><br>(Xe-133) | ± 20% of scale | $6E^{-8}$<br>$2E^{-8}$<br>$1E^{-4}$ | 2.5 |

→ (DRN 99-2362, R11; 02-407, R12)

← (DRN 99-2362, R11; 02-407, R12)

** Setpoints determined in accordance with Offsite Dose Calculation Manual.

*** W-3 tag number.

TABLE 12.3-4

NEUTRON STREAMING DOSE RATES IN CONTAINMENT

| | Containment Location | Neutron Dose Rate (Rem/hr) |
|---|---|---|
| 1. | Edge of Refueling Cavity at Refueling Machine Normal Parking Location-Operating Floor | 16.5 |
| 2. | Extreme Edge of Refueling Cavity Near Steel Containment-Operating Floor | 12.0 |
| 3. | Edge of Refueling Cavity Near Steam Generator Shield Wall-Operating Floor | 22 |
| 4. | Corner of Refueling Cavity Below Missile Shield-Operating Floor | 25.6 |
| 5. | Near Access Lock-Mezzanine Floor | 0.25 |
| 6. | +21 EL General Mezzanine Floor Area | 0.025-0.3 |
| 7. | -4 EL General Lower Floor Area | 0.005-0.06 |