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12.0 RADIATION PROTECTION

12.1 Assuring that Occupational Radiation Exposures Are as Low as Reasonably Achievable (ALARA)

12.1.1 Policy Considerations

TVA has established a formal program to ensure that occupational radiation exposures to employees are kept as low as reasonably achievable (ALARA). The program consists of: (1) full management commitment to the overall objectives of ALARA; (2) issuance of specific administrative documents and procedures to the TVA design and operating groups that emphasize the importance of ALARA through the design, testing, startup, operation, maintenance and decommissioning phases of TVA nuclear plants; and (3) continued appraisal of inplant radiation and contamination conditions by the onsite radiation protection staff.

12.1.2 Design Considerations

The facility and equipment design features for control of occupational radiation exposures are described in detail in Section 12.3. Although the original design of Watts Bar Nuclear Plant predated Regulatory Guide 8.8, the concept of keeping occupational exposures ALARA is an important consideration throughout new designs and modifications of the plant. In addition, the plant design effort routinely considers radiation protection experience at other nuclear plants.

New designs and modifications of the plant are performed and reviewed by engineers and health physicists with several years of experience in radiation protection design. In addition, the design of the plant is continually reviewed and modified as necessary when new ALARA concerns become known. Close communication among the design staff, equipment vendors, operating and maintenance personnel, and Radiological Control Personnel is maintained in order to design Watts Bar Nuclear Plant and its equipment with ALARA considerations as a primary concern.

Dose assessment based on operating experience is discussed in Section 12.4.

In general, piping which may contain significant concentrations of radioactive materials is not field-run. Some sample and radiation monitoring lines are field-run. While the exact location is set in the field, the general location is determined by the designer to minimize radiation exposure.

12.1.3 ALARA Operational Considerations

Consistent with TVA's overall commitment to keep occupational radiation exposures as low as reasonably achievable, specific plans and procedures are followed by operating and maintenance staff to assure that ALARA goals are achieved in the operation of the plant. Operational ALARA policy and procedures are formulated at the Corporate level in Nuclear Power and are implemented at each nuclear plant through the issuance of division procedures and plant instructions for the purpose of maintaining Total Effective Dose Equivalent (TEDE) ALARA. These procedures and instructions are consistent with the intent of Section C.1 of Regulatory Guide 8.8 and

Regulatory Guide 8.10. Included in these operating procedures and plant instructions are the provision that employee radiation exposure trends are reviewed periodically by management staff at the plant and in the central office. Summary reports are prepared that describe: (a) major problem areas where high radiation exposures are encountered; (b) which worker group is accumulating the highest exposures; and (c) recommendations for changes in operating, maintenance, and inspection procedures or modifications to the plant as appropriate to reduce exposures.

Maintenance activities that could involve significant radiation exposure of employees are carefully planned and carried out using well-trained personnel and proper equipment. Where applicable, specific radiation exposure reduction techniques, such as those set out in Section C.3 of Regulatory Guide 8.8, are used. Careful personnel radiation and contamination monitoring are integral parts of such maintenance activities. Upon completion of major maintenance jobs, personnel radiation exposures are evaluated and assessed relative to predicted man-rem exposures so that appropriate changes can be made in techniques or procedures for future jobs.

Additionally at the plant level, the Plant Operations Review Committee reviews operating and maintenance activities involving the major systems of the plant (i.e., radwaste, NSSS, etc.) to further assure that occupational exposures are kept as low as reasonably achievable.

An ALARA committee composed primarily of supervisory personnel is established to review periodically the effectiveness of implementation of the ALARA Program. Reviews include the site performance against ALARA goals, employee ALARA suggestions, ALARA planning documents, and trends. The Plant Manager or Assistant Plant Manager will normally serve as chairman of the site ALARA committee.

REFERENCES

None

12.2 RADIATION SOURCES

12.2.1 Contained Sources

With the exception of airborne radioactive sources discussed in Section 12.2.2, the source terms and associated bases are those employed in the initial design of the plant shielding. Source terms are presented in accordance with the following organization:

(1) Primary system sources

(2) Auxiliary systems sources

(3) Sources during refueling

(4) Maximum hypothetical accident (MHA) sources

Systems for which source terms are presented in this section are essentially the same in the Watts Bar plant as they are in the Sequoyah plant. Therefore, the shielding design source terms developed for the Sequoyah FSAR are applicable and are incorporated into the Watts Bar FSAR. For completeness, the bases for development of these source terms are given in the following sections. These bases were used in calculations related to shielding design source terms but do not describe plant operation constraints. The specific activities, flows, volumes, system mass, etc. represent typical or expected plant data, but do not constrain plant alignment.

12.2.1.1 Primary System Sources

12.2.1.1.1 Sources Shielded by Primary Shield Concrete

The major sources shielded by the primary concrete shield are the neutron and gamma sources inside the reactor pressure vessel, the gamma sources in the pressure vessel wall itself, and the coolant activities inside those parts of the coolant pipes from the reactor vessel to approximately the outer edges of the weld inspection openings.

Core center plane neutron fluxes across the primary shield concrete are shown on Figure 12.2-1. Fluxes across the primary concrete for distances (H) above and below the core center plane are determined by using the following axial peaking factors (F_p):

H (ft.)	0	1	2	3	4	5	6
F_p	1.0	1.0	0.99	0.93	0.76	0.51	0.30

The neutron fluxes along the vessel z axis (vertically) are essentially zero at the top and bottom outer surfaces of the reactor vessel.

Principal gamma sources during power operation are the prompt fission gammas, the delayed fission product gammas, prompt capture gammas, and the activation (Co-60, Fe-59, etc.) gammas. Core center plane gamma fluxes across the primary shield concrete during full power operation are shown on Figure 12.2-2. Fluxes across the primary concrete for other axial distances are found by using the same axial peaking factors as given above for neutron fluxes. The principal gamma sources during shutdown are the fission product delayed gammas and the activation gammas.

The principal gamma sources in the reactor coolant are the fission product, reactor coolant activation product, and noncoolant activation (corrosion) product activities. The fission product and noncoolant activation product activities are given in Table 11.1-2. The only coolant activation product significant for shielding purposes is N-16. Values of the N-16 activity are shown in Figure 12.2-3.

12.2.1.1.2 Sources Shielded by Secondary Shield Concrete

The secondary concrete shielding provides shielding for that portion of the primary coolant system which lies outside the primary concrete shield. Major elements are the coolant recirculation pumps, the steam generators, the pressurizer and all connecting piping. In addition, the secondary shielding attenuates radiation that penetrates the primary shield concrete. Activities in the primary coolant system have been identified above. Specified activities in the vapor and liquid spaces of the pressurizer are given in Table 11.1-4.

12.2.1.2 Auxiliary Systems Sources

The auxiliary systems equipment for which a determination of shielding sources is required consists of pumps, heat exchangers, demineralizers, filters, units of evaporator packages, liquid tanks, gas tanks, and pipes. Inventories (or specific activity and volume or mass from which inventory can be obtained) used in the determination of required shielding for auxiliary systems components are given in Tables 12.2-1 through 12.2-11. Development of these source terms is described in the following subsections.

12.2.1.2.1 Chemical and Volume Control System - Mixed-Bed Demineralizers

The reactor coolant with the specific activities (μCi/gm) given in Table 11.1-2 is assumed to flow at 75 gal/min (284,000 gm/min) through a mixed-bed demineralizer for 365 days. Consistent with the basis for generating the specific activities in Table 11.1-1, a mixed-bed removal efficiency of 0.09 for cesium, molybdenum, and yttrium and 0.90 for all other nongaseous elements is assumed.

In the case of the isotopes with half-lives that are not long in comparison with the rate at which reactor coolant is purified in the chemical and volume control system, demineralizer inventories can be higher than those obtained using these assumptions. This effect is taken into account by increasing the inventories of the following isotopes by the factor 1.11: Mn-56, Br-84, Rb-89, Sr-91, Y-91m, Y-91, Te-132, Te-134, I-132, I-133, I-134, and I-135. Similarly, the inventories of Mo-99, Tc-99m, and Cs-138 are increased by the factor 11.1. Source term inventories are given in Table 12.2-1.

12.2.1.2.2 Chemical and Volume Control System - Cation Bed Demineralizer

Reactor coolant is assumed to flow at an average 7.5 gal/min (28,400 gm/min) through the cation bed demineralizer for 365 days. A cation removal efficiency of 90% for cesium, molybdenum and yttrium is assumed. (For this calculation a removal efficiency of 0.0 for these three elements is assumed for the upstream mixed-bed demineralizer.) In the case of isotopes with half-lives that are not long in comparison with the rate at which reactor coolant is processed through the cation demineralizer, demineralizer inventories can be higher than those obtained using these assumptions. This effect is taken into account by increasing, by the factor 1.11, the inventories of the following isotopes: Mo-99, Tc-99m, Cs-136, Cs-138, and Y-90.

Inventories of other cations on the cation bed demineralizer are not included in Table 12.2-2. In general, the inventories of these other cation isotopes will be a maximum of about 1.0% of the inventories shown in Table 12.2-2, for the mixed-bed demineralizer. These maximum values, as translated into gamma source strengths, are small in comparison with the cation bed shielding source term represented by the gamma radiation from the molybdenum, yttrium, and cesium isotopes.

12.2.1.2.3 Chemical and Volume Control System - Volume Control Tanks

Isotopic inventories in the vapor and liquid spaces of the volume control tank are given in Table 11.1-3.

12.2.1.2.4 Chemical and Volume Control System - Reactor Coolant Filter

The source for which reactor coolant filter shielding is designed consists of activity from the upstream demineralizers. For shielding calculations, 1.0% of the mixed-bed demineralizer inventory as given in Table 12.2-1 is distributed uniformly throughout the reactor coolant filter cartridge. In the generation of the mixed-bed demineralizer inventory, it was assumed that the mixed-bed demineralizer removes all nongaseous activity including cesium, yttrium, and molybdenum. Thus, the mixed-bed demineralizer inventory as calculated contains the maximum inventory that could be collected on the mixed-bed demineralizers and cation demineralizer combined. It follows that the reactor coolant filter inventory determined with the stated prescription includes 1.0% of this maximum combined inventory. The filter cartridge volume over which the isotopic inventories are uniformly distributed is 1.17 cubic feet (diameter equals 8-15/16 inches, length equals 32-5/16 inches).

12.2.1.2.5 Chemical and Volume Control System - Seal Water Return Filter

The only significant radioactive source on the seal water return filter is accumulated during operation with the excess letdown heat exchanger. The excess letdown heat exchanger, can be employed when normal letdown is temporarily out of service or it can be used to supplement maximum letdown during final states of heatup. The seal water return filter will thus collect activity during relatively short and infrequent time periods. A maximum inventory on the filter is generated by allowing undemineralized and unfiltered reactor coolant to flow continuously for 30 days through the filter at the design flow rate of the excess letdown heat exchanger. The design flow rate is 24.7 gal/min or 12,380 lb/hr. The filter is assumed to remove all corrosion product activities and 0.4% of the cesium activity in the reactor coolant. The maximum inventory on a filter is given in Table 12.2-3.

12.2.1.2.6 Chemical and Volume Control System - Seal Water Injection Filters

The only significant radioactive source on the seal water injection filters will consist of activity of particulates that are passed by the seal water return filter (98% retention of 25 micron size) but are collected by the online injection filter (98% retention of 5 micron size). The maximum inventory on each filter assumed for shielding calculations is taken to be the same as that given in Table 12.2-3 for the seal water return filter. The inventory is distributed over a filter volume of 1.25 cubic foot (diameter equals 8.625 inches, length equals 37.0 inches).

12.2.1.2.7 Chemical and Volume Control System - Holdup Tanks

Holdup tank inventory development begins with the assumption that reactor coolant is discharged from both units simultaneously into a single holdup tank. Flow into the tank is 120 gal/min/unit. Flow from one unit is assumed to be undemineralized and unfiltered. For the flow from the other unit a decontamination factor (D.F.) of 10 across a mixed-bed demineralizer is taken for all nongaseous isotopes except those of cesium, yttrium and molybdenum. Filling continues until liquid occupies one-fourth of the tank volume. The liquid volume is then 32,000 gallons and the vapor space volume is 96,000 gallons. Under equilibrium conditions, 68% of the gases will be retained in the liquid. It is assumed that the other 32% is retained in the tank vapor space although physically, some of the vaporized gas would have been forced out of the tank to the vent header. Particulate daughters of noble gases entering the tank are retained in the tank liquid space. Inventories in the tank liquid and vapor spaces are given in Table 12.2-4 and 12.2-5, respectively.

12.2.1.2.8 Chemical and Volume Control System - Evaporator Feed Mixed-Bed Ion Exchangers

A constant flow at 30 gpm (boric acid-evaporator capacity) through an evaporator feed mixed-bed ion exchanger for 60 days is assumed. This procedure is a conservative mock-up of actual ion exchanger operation for shorter time periods during a year of plant operation. It is further assumed that all flow has been processed through one of the chemical and volume control system mixed-bed demineralizers and that 10% of the volume has been processed through a chemical and volume control system cation demineralizer. Thus, a minimum of 9.0% of the cesium, yttrium, and molybdenum isotopic activities and 90% of the other nongaseous activities have been removed from the coolant prior to its processing through the evaporator feed mixed-bed ion exchanger. The evaporator feed ion exchanger is assumed to remove all of the remaining nongaseous activity. Since coolant activities, as given in Table 11.1-2, do not take into account isotopic removal by the evaporator feed ion exchangers, the ion exchange inventories developed with this method are too high. Corrections to the inventories of isotopes with long half-lives are readily obtained. In the case of the Cs-134, Cs-137, and Y-91 (excluding that part of the Y-91 inventory that results from decay of Sr-91 on the ion exchanger), the inventory is reduced by the factor, (0.09 x 75 gpm)/(0.09 x 75 gpm + 0.91 x 30 gpm) = 0.198. The similarly obtained correction factor for other long-lived activity, (0.90 x 75 gpm)/(0.90 x 75 gpm + 0.10 x 30 gpm) = 0.96 is not applied. The derived shielding source team inventories are given in Table 12.2-6.

The CVCS boric acid evaporator package is not required for the operation of Unit 2; however, the package is installed and connected to the waste disposal system. Liquid waste will be processed through the mobile demineralizer until such time that the boric acid evaporator becomes an available option for processing liquid radioactive waste.

12.2.1.2.9 Chemical and Volume Control System - Evaporator Feed Cation Bed Ion Exchanger

The shielding source term inventory on the evaporator feed cation bed ion exchanger is taken to be the same as the cesium, molybdenum, and yttrium inventories on the evaporator feed mixed bed ion exchanger except that yttrium activities on the mixed-bed that result from decay of strontium on the mixed-bed are excluded. The inventories of other cations on the cation demineralizer are not included since they would be very small in comparison with the inventories obtained from cesium, yttrium and molybdenum. (See the note in Section 12.2.1.2.8 relative to the Unit 2 operation without the boric acid evaporator package.)

The derived shielding source term inventories are given in Table 12.2-7.

12.2.1.2.10 Chemical and Volume Control System - Ion Exchange Filters

The maximum source on either of the ion exchange filters is assumed to be 1.0% of the maximum inventory that could exist on an evaporator cation bed ion exchanger. The cation exchanger is downstream of the mixed-bed ion exchanger and thus any detached resin fines from the mixed-bed ion exchanger would become lost in the cation bed resin. For shielding calculations the filter inventory is uniformly distributed over a filter cartridge volume of 0.44 cubic feet (diameter equals 7 inches, length equals 19-7/8 inches). (See the note in Section 12.2.1.2.8 relative to the Unit 2 operation without the boric acid evaporator package.)

12.2.1.2.11 Gas Stripper and Boric Acid Evaporator Package

The sources associated with the recycle evaporator are specified in this section. The gaseous activity is concentrated in the vent condenser portion, while particulate activity is concentrated in the evaporator section.

The maximum gaseous activity concentrations occur when the unit is processing reactor coolant at the maximum rate. No credit is taken for radioactive decay before processing in the gas stripper. Gas concentration in the reactor coolant is assumed to be 35 standard cubic centimeters per kilogram. The activity concentrations in the vent condenser in microcuries per cubic centimeter are thus obtained by increasing reactor coolant concentrations, in microcuries per gram, by the factor, 1000 gm/35 cc = 28.6 gm/cc. Concentrations are given in Table 12.2-8 (these concentrations are not affected by considerations of hydrogen removal via venting from the holdup tank if the fractional release of nitrogen and hydrogen from the coolant is assumed to be the same as that of the radioactive gases). (See the note in Section 12.2.1.2.8 relative to the Unit 2 operation without the boric acid evaporator package.)

For shielding calculations, the vent condenser is considered to be a cylindrical source with a diameter of 8 inches and a length of 20 inches. Although the composition is approximately 30% stainless steel, 22% water and 48% vapor (volume percents) the shielding calculations do not take credit for any attenuation in the steel.

To generate the shielding source inventories in the evaporator bottoms, it is assumed that reactor coolant is processed through the evaporator at 30 gpm (1.14×10^5 gm/min). It is assumed that only 0.10 of all nongaseous isotopes are removed in the evaporator feed mixed-bed and cation bed ion exchangers. Processing continues until isotopic concentrations in the evaporator bottoms is about 40 µCi/gm of liquid. The inventory is given in Table 12.2-9. This inventory is obtained in 417 minutes of processing. At this point the activity in the liquid has been concentrated by about a factor of 23. (See the note in Section 12.2.1.2.8 relative to the Unit 2 operation without the boric acid evaporator package.)

For shielding calculations, the source geometry is considered to be a cylinder with a diameter of 3.5 feet, and a length of 9.9 feet. The composition in volume percents is 10% stainless steel, 77% water, and 13% air. The source inventory is uniformly distributed throughout the 712 gallons homogenized volume.

12.2.1.2.12 Spent Fuel Pool Cooling System Demineralizer and Filters

The principal isotopes that will be collected on the demineralizer and filters of the spent fuel pool cooling system are I-131, Cs-134, Cs-137, and the corrosion product activities, Mn-54, Co-58, Co-60, Fe-59 and Cr-51. In the development of source terms for these components, fission product release from the fuel rods subsequent to the initiation of shutdown is neglected. Minimum time between beginning of cooldown and filling of the refueling canal is at least three days. Therefore, continuous purification at 75 gal/min (284,000 gm/min) through the chemical and volume control system mixed-bed and cation bed demineralizers is assumed to continue for three days before the reactor vessel head is removed and primary coolant mixes with water in the refueling canal (flow through the mixed-bed demineralizer could be 120 gpm but the design flow rate through the cation bed demineralizer is only 75 gpm). A D.F. of 10 between a point upstream of the mixed-bed demineralizer and a point downstream of cation bed demineralizer is taken for the iodine and cesium isotopes. Credit for radioactive decay during the three days of purification is taken for I-131.

Although the demineralizer D.F. of 10 is also applicable to the corrosion product activities, removal of corrosion product activities in the demineralizers could be balanced by further releases into the coolant. In fact, the evidence suggests that corrosion product activity levels in the coolant could actually increase during such temperature changes. Also, large increases in Co-58 activity have been observed after removal of the reactor vessel head and dilution of the coolant of the refueling canal[3]. This effect has been taken into account by increasing the corrosion product inventory in the reactor coolant at beginning of shutdown by a factor of 5. The radioactive inventories available for demineralization with the spent fuel pool demineralizer and for filtration with the refueling water purification filter are thus determined as follows:

(1) I-131 inventory is decreased exponentially with the removal constant, 1.60 day^{-1},

(2) Cs-134 and Cs-137 inventories are decreased exponentially with the removal constant, 1.51 day^{-1}, and

(3) Corrosion product inventories are increased by a factor of 5. No credit is taken for radioactive decay during accumulation on the demineralizer. Demineralizer inventories are given in Table 12.2-10.

The inventory on a refueling water purification filter will be accumulated in one of two ways. If water entering the filter has first passed through the spent fuel pool demineralizer, the filter source would consist of activity on resin dislodged from the demineralizer. One percent of the demineralizer inventories would be used as a filter source. If water is purified through the refueling water purification filter only, the filter inventory would be identical to the spent fuel pool demineralizer inventory in the case of the corrosion product activities.

Iodine would not be collected in significant quantity. As in the case of the seal water return and seal water injection filters, a filtration efficiency of 0.4% would be assumed for cesium. This latter mode of operation results in the worst case inventory for shielding calculations and thus is the basis for the refueling water purification filter inventories given in Table 12.2-11.

No large quantities of radioactivity are expected to be collected on the spent fuel pool skimmer filter and the spent fuel pool filter. These filters filter the water in the transfer canal and in the spent fuel pool only. Negligible activity from the refueling canal inside the containment will enter the transfer canal through the fuel transfer tube. Identifiable sources of spent fuel pool and skimmer filter activity are the following:

(1) Particulate corrosion and fission product activities that become dislodged from fuel assembly surfaces primarily during movement of a fuel assembly.

(2) Activity from fuel assemblies with defective cladding.

(3) In the case of the spent fuel pool filter only, activity on resin fines dislodged from the upstream spent fuel pool demineralizer. An upper limit for this source is considered to be 1.0% of the demineralizer activity inventory.

(4) Activity introduced from the refueling water storage tanks when the spent fuel pool is periodically emptied and refilled.

Of these, the largest potential contribution to filter activity is expected to be the corrosion and filter product activities dislodged from the fuel assembly. An extreme upper limit for this source should be the inventory assigned to a refueling water purification filter. Use of this inventory as the source term for the spent fuel pool filter and the spent fuel pool skimmer filter more than adequately takes into account activity contributions from the other three activity origins identified above. Therefore, the inventory on a refueling water purification filter, given in Table 12.2-11, is also used as the source term in shielding calculations for the spent fuel pool filter and the spent fuel pool skimmer filter.

12.2.1.3 Sources During Refueling

The principal radioactive source in the proximity of the fuel assembly transfer path is the spent fuel assembly fission product inventory discussed below. Activity in the refueling canal water will normally be reduced to 0.01 microcurie per cubic centimeter before fuel transfer begins.

The maximum fuel assembly fission product concentrations are calculated by TID-14844 methodology[5], and they are included in Table 12.2-12. The total power distribution peaking factor is 2.40. The axial peaking factor is 1.37, and the radial peaking factor is 1.75. The average inventory in a fuel assembly is 1/2.40 of the inventory in Table 12.2-12. Since there are 193 fuel assemblies in the core, the total core inventory is 193 times the average inventory in a fuel assembly.

Other sources encountered during refueling operations are the irradiated incore instrumentation thimble assemblies and the control rods. When removed from the vessel, the irradiated incoreinstrumentation thimble assembliesand the irradiated Ag-In-Cd control rods are also important sources. During refueling, the irradiated incore instrumentation thimble assemblies are stored in the BMI guide tubes.   . The shielding provided for spent fuel assemblies also shields spent irradiated control rod sources either in place in an assembly or when being transferred from one assembly to another.

The absorber materials used in the hybrid control rods are boron carbide (B_4C) and silver-indium-cadmium (Ag-In-Cd). The gamma ray source strengths associated with the Ag-In-Cd absorber are listed in Table 12.2-14 for various times after shutdown. The values are per cubic centimeter of absorber for an irradiation period of 4 years. There are no significant gamma ray sources associated with the B_4C absorber.

The material used for the control rod cladding, secondary source rod cladding, and burnable absorber rod cladding and inner sheath is type-304 stainless steel with a maximum cobalt content of 0.12 weight percent. The gamma ray source strengths associated with the stainless steel are also listed in Table 12.2-14 for various times after shutdown. The values are per cubic centimeter of stainless steel for an irradiation time of 15 years.

12.2.1.4 Maximum Hypothetical Accident (MHA) Sources

For MHA (maximum hypothetical accident) exposure calculations, fractions of the equilibrium core fission product inventory (see Table 12.2-12) that are assumed to be released from the fuel rods are the following:

(1) 100% of the noble gases released to containment

(2) 50% of the core iodines released to the sump and 50% of pure iodines released to containment

(3) 1% of core particulates released to the sump

Refer to Chapter 15 for details of accident analysis.

12.2.1.5 Condensate Demineralizer Waste Evaporator

The Condensate Demineralizer Waste Evaporator (CDWE) package is not required for the operation of Unit 2, however the package is installed and connected to the waste disposal system. System description, function, and interfaces are contained in Section 11.2.2. Liquid waste will be processed through the mobile demineralizer until such time that the CDWE becomes an available option for processing liquid radioactive wastes.

12.2.2 Airborne Radioactive Material Sources

Expected airborne activity levels and the corresponding derived air concentration (DAC) fractions, for the Containment, Turbine Building, and Auxiliary Building and the instrument room are presented in Tables 12.2-19 through 12.2-22, respectively. The DAC fractions are determined by dividing the estimated radioactive air concentrations by the corresponding DAC values. For the determination of these concentrations, the assumptions and models are those employed in Section 11.3.7 for the determination of gaseous effluent releases except for the following assumptions: (1) lower containment, upper containment, and the instrument room are assumed to be purged after each 60 days of operation, (2) purge flows are 14,958 cfm (upper containment), 7,500 cfm (lower containment), 540 cfm (instrument room), and (3) purge lasts for 6 hours.

REFERENCES

(1) Deleted by Amendment 84.

(2) Deleted by Amendment 84.

(3) "Coolant Activity Experiences at Connecticut Yankee," by R. H. Graves, Nuclear News, November 1970, page 66. (See 12.2.1.2.12)

(4) ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors," December 31, 1984.

(5) TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962.

Table 12.2-1 Chemical And Volume Control System Mixed Bed Demineralizer

Isotope	Inventory (Microcuries)	Isotope	Inventory (Microcuries)
Cr 51	.137(+8)	I 131	.107(+11)
Mn 54	.700(+8)	I 132	.543(+9)
Mn 56	.184(+7)	I 133	.204(+10)
Co 58	.917(+9)	I 134	.126(+8)
Co 60	.932(+8)	I 135	.365(+9)
Fe 59	.239(+8)	Ba 137m	.124(+11)
Br 84	.548(+6)	Ba 140	.286(+8)
Rb 88	.270(+8)	La 140	.299(+8)
Rb 89	.631(+6)	Ce 144	.241(+8)
Sr 89	.102(+9)	Pr 144	.241(+8)
Sr 90	.146(+8)	Mo 99	.865(+10)
Sr 91	.454(+6)	Tc 99m	.758(+10)
Y 90	.145(+8)	Cs 134	.240(+10)
Y 91m	.277(+6)	Cs 136	.968(+8)
Y 91	.172(+8)	Cs 137	.133(+11)
Zr 95	.228(+8)	Cs 138	.125(+8)
Nb 95m	.455(+6)		
Nb 95	.341(+8)		
Te 132	.498(+9)		
Te 134	.522(+6)		

a. Inventory is distributed over a resin volume of 30 cubic feet.

b. Numbers in parentheses denote exponents to the base 10.

Table 12.2-2 Chemical And Volume Control System Cation Bed Demineralizer

Isotope	Inventory (Microcuries)
Mo 99	.865(+9)
Tc 99m	.758(+9)
Cs 134	.240(+10)
Cs 136	.107(+9)
Cs 137	.133(+11)
Cs 138	.125(+7)
Ba 137m	.124(+11)
Y 90	.204(+5)
Y 91	.168(+8)

a. Inventory is distributed over a resin volume of 20 cubic feet.

b. Numbers in parentheses denote exponents to the base 10.

Table 12.2-3 Chemical and Volume Control System Seal Water Return Filter

Isotope	Inventory (Microcuries)
Cr 51	.265(+7)
Mn 54	.301(+7)
Mn 56	.607(+6)
Co 58	.877(+8)
Co 60	.298(+7)
Fe 59	.324(+7)
Cs 134	.335(+7)
Cs 136	.113(+7)
Cs 137	.162(+8)
Cs 138	.165(+5)
Ba 137m	.151(+8)

a. Inventory is distributed over a filter cartridge volume of 3.72 ft3, (Diameter = 13.75", Length 43.25”)

b. Numbers in parentheses denote exponents to the base 10.

Table 12.2-4 Chemical And Volume Control System-Hcldup Tank Liquid Space

Isotope	Inventory (Microcuries)	Isotope	Inventory (Microcuries)
Cr 51	.620(+5)	Te 132	.172(+8)
Mn 54	.514(+5)	Te 134	.808(+6)
Mn 56	.146(+7)	Cs 134	.255(+8)
Co 58	.167(+7)	Cs 136	.170(+8)
Co 60	.494(+5)	Cs 137	.121(+9)
Fe 59	.667(+5)	Cs 138	.492(+8)
Br 84	.912(+6)	Ba 137m	.110(+9)
Rb 88	.337(+9)	Ba 140	.280(+6)
Rb 89	.111(+7)	La 140	.104(+6)
Sr 89	.255(+6)	Ce 144	.180(+5)
Sr 90	.734(+u)	Pr 144	.180(+5)
Sr 91	.117(+6)	Kr 85m	.153(+9)
Zr 95	.447(+5)	Kr 85	.338(+9)
Nb 95	.427(+5)	Kr 87	.573(+8)
I 131	.166(+9)	Kr 88	.235(+9)
I 132	.484(+8)	Xe 131m	.156(+9)
I 133	.257(+9)	Xe 133m	.260(+9)
I 134	.181(+8)	Xe 133	.236(+11)
I 135	.131(+9)	Xe 135m	.921(+7)
		Xe 135	.486(+9)
		Xe 138	.849(+7)

a. Inventory is distributed over a volume, of 32,000 gallons (the bottom 1/4 of the tank volume).

b. Numbers in parentheses denote exponents to the base 10.0.

Table 12.2-5 Chemical And Volume Control System-Hcidup Tank Vapor Space

Isotope	Inventory (Microcuries)
Kr 85m	.721(+8)
Kr 85	.183(+9)
Kr 87	.270(+8)
Kr 88	.111(+9)
Xe131m	.736(+8)
Xe 133m	.123(+9)
Xe 133	.112(+11)
Xe 135m	.433(+7)
Xe 135	.229(+9)
Xe 138	.400(+7)

a.Inventory is distributed over a volume of 96,000 gallons (the top 3/4 of the tank volume).

b.Numbers in parentheses denote exponents to the base 10.

Table 12.2-6 Chemical And Volume Control System - Evaporator Feed Mixed Bed Ion Exchanger

Isotope	Inventory (Microcuries)	Isotope	Inventory (Microcuries)
Cr 51	.475(+6)	Te 132	.200(+8)
Mn 54	.707(+6)	Te 134	.209(+5)
Mn 56	.738(+5)	I 131	.474(+9)
Co 58	.186(+8)	I 132	.220(+8)
Co 60	.720(+6)	I 133	.821(+8)
Fe 59	.643(+6)	I 134	.505(+6)
Br 84	.219(+5)	I 135	.147(+8)
Br 88	.108(+7)	Ba 137m	.165(+10)
Rb 89	.253(+5)	Ba 140	.122(+7)
Rb 89	.256(+7)	La 140	.127(+7)
Sr 90	.108(+6)	Ce 144	.247(+6)
Sr 91	.182(+5)	Pr 144	.247(+6)
Y 90	.180(+6)	Mo 99	.316(+10)
Y 91M	.111(+5)	Tc 99m	.277(+10)
Y 91	.697(+7)	Cs 134	.362(+9)
Zr 95	.488(+6)	Cs 136	.376(+9)
Nb 95m	.908(+4)	Cs 137	.177(+I0)
Nb 95	.586(+6)	Cs 138	.457(+7)

a. Inventory is distributed over a resin volume of 27 ft^3.

b. Numbers in parentheses denote exponents to the base 10.

Table 12.2-7 Chemical And Volume Control System-Evaporator Feed Cation Bed Ion Exchanger

Isotope	Inventory (Microcuries)
Mo 99	.316(+10)
Tc 99m	.277(+10)
Cs 134	.376(+9)
Cs 136	.177(+10)
Cs 137	.177(+10)
Cs 138	.457(+7)
Ba 137m	.165(+10)
Y 90	.746(+5)
Y 91	.697(+7)

a.Inventory is distributed over a resin volume of 20 ft^3.

b.Numbers in parentheses denote exponents to the base 10.

Table 12.2-8 Gas Stripper And Boric Acid Evaporator Package-Vent Condenser

Isotope	Concentration (Microcuries/Cubic Centimeter)
Kr 85m	63.0
Kr 85	134
Kr 87	34.3
Kr 88	106
Xe 131m	54.3
Xe 133m	91.5
Xe 133	8230
Xe 135m	5.43
Xe 135	180
Xe 138	19.0

a. Activity concentration exists in 48% of the total condenser volume. Shielding calculations are performed by assuring concentrations in the entire condenser volume, 1.648×10^4 cm^3, are 48% of the concentrations in this table.

Table 12.2-9 Gas Stripper And Boric Acid Evaporator Package-Evaporator Bottoms

Isotope	Inventory (Microcuries)	Isotope	Inventory (Microcuries)
Cr 51	.439(+4)	Zr 95	.317(+4)
Mn 54	.365(+4)	Nb 95m	.167(+1)
Mn 56	.623(+5)	Nb 95	.303(+4)
Co 58	.118(+6)	Mo 99	.242(+8)
Co 60	.351(+4)	Tc 99m	.671(+7)
Fe 59	.473(+4)	I 131	.117(+8)
Br 84	.219(+5)	I 132	.248(+7)
Br 88	.108(+7)	I 133	.169(+8)
Rb 89	.253(+5)	I 134	.501(+6)
Sr 89	.181(+5)	I 135	.746(+7)
Sr 90	.521(+3)	Te 132	.120(+7)
Sr 91	.710(+4)	Te 134	.209(+5)
Y 90	.613(+3)	Ba 137m	.439(+7)
Y 91m	.369(+4)	Ba 140	.198(+5)
Y 91	.260(+5)	La 140	.784(+4)
Y 92	.189(+4)	Ce 144	.128(+4)
		Pr 144	.128(+4)
		Cs 134	.995(+6)
		Cs 136	.659(+6)
		Cs 137	.474(+7)
		Cs 138	.502(+6)

a.Inventory is distributed over a homogenized volume of 712 gallons.

b.Numbers in parentheses denote exponents to the base 10.

Table 12.2-10 Spent Fuel Pool Cooling System-Spent Fuel Pool Demineralizer

Isotope	Inventory (Microcuries)
I 131	.500(+7)
Cs 137	.261(+7)
Cs 134	.547(+6)
Mn 54	.940(+6)
Co 58	.305(+8)
Co 60	.905(+6)
Fe 59	.122(+7)
Cr 51	.114(+7)

a.Inventory is distributed uniformly over a resin volume of 30 cubic feet.

b.Numbers in parentheses denote exponents to the base 10.

Table 12.2-11 Spent Fuel Pool Cooling System-refueling Water Purification Filter

Isotope	Inventory (Microcuries)
Cs 137	.104(+5)
Cs 134	.219(+4)
Mn 54	.940(+6)
Co 58	.305(+8)
Co 60	.905(+6)
Fe 59	.122(+7)
Cr 51	.114(+7)

a.Numbers in parentheses denote exponents to the base 10.

Table 12.2-12 Core Inventory
(Page 1 of 3)

	Isotope	Half-Life (Sec.)	Fisson Yield	Inventory (ci)
1	Kr 83m	6.6960E+03	5.3069E-03	1.64E+07
2	Kr 85m	1.6128E+04	1.3017E-02	3.95E+07
3	Kr 85	3.3862E+08	2.8825E-03	9.99E+05
4	Kr 87	4.5780E+03	2.5421E-02	7.59E+07
5	Kr 88	1.0080E+04	3.5840E-02	1.08E+08
6	Kr 89	1.9080E+02	4.6812E-02	1.40E-08
7	Kr 90	3.2320E+01	4.6891E-02	1.45E-08
8	Xe 131m	1.0282E+06	3.9694E-04	6.68E+05
9	Xe 133m	1.9440E+05	1.9140E-03	5.16E+06
10	Xe 133	4.5706E+05	6.7705E-02	2.03E+08
11	Xe 135m	9.3900E+02	1.0564E-02	5.46E+07
12	Xe 135	3.2940E+04	6.6334E-02	5.55E+07
13	Xe 137	2.2980E+02	6.1325E-02	1.89E+08
14	Xe 138	8.5020E+02	6.2836E-02	1.79E+08
15	Xe 139	4.0000E+01	5.1578E-02	1.59E+08
16	Xe 140	1.3600E+01	3.7182E-02	1.15E+08
17	I 130	4.4496E+04	2.4100E-06	7.43E+03
18	I 131	6.9638E+05	2.8352E-02	8.80E+07
19	I 132	8.2080E+03	4.2083E-02	1.34E+08
20	I 133	7.4880E+04	6.7653E-02	1.97E+08
21	I 134	3.1560E+03	7.6117E-02	2.31E+08
22	I 135	2.3756E+04	6.4065E-02	1.79E+08
23	I 136m	4.6000E+01	2.1095E-02	6.51E+07
24	Br 83	8.6400E+03	5.3069E-03	1.64E+07
25	Br 84m	3.6000E+02	1.9217E-04	5.93E+05
26	Br 84	1.9080E+03	9.6650E-03	2.98E+07
27	Br 85	1.8000E+02	1.2953E-02	3.99E+07
28	Br 87	5.5700E+01	2.2016E-02	6.79E+07
29	Cs 134	6.5070E+07	4.5000E-07	6.25E+02
30	Cs 135	7.2580E+13	6.6348E-02	1.10E+02
31	Cs 136	1.1215E+06	5.2710E-05	1.63E+05
32	Cs 137	9.4671E+08	6.2626E-02	7.78E+06
33	Cs 138	1.9320E+03	6.7178E-02	2.07E+08
34	Cs 139	5.6400E+02	6.4137E-02	1.98E+08
35	Cs 140	6.3700E+01	5.9022E-02	1.82E+08
36	Cs 141	2.4900E+01	4.4186E-02	1.36E+08
37	Rb 88	1.0680E+03	3.6243E-02	1.12E+08
38	Rb 89	9.1200E+02	4.8470E-02	1.49E+08
39	Rb 90m	2.5600E+02	1.0034E-02	3.09E+07
40	Rb 90	1.5400E+02	4.9249E-02	1.52E+08
41	Rb 91	5.8000E+01	5.7450E-02	1.77E+08

Table 12.2-12 Core Inventory
(Page 2 of 3)

	Isotope	Half-Life (Sec.)	Fisson Yield	Inventory (ci)
42	Se 84	1.9800E+02	9.4511E-0	32.91E+07
43	Sr 89	4.3632E+06	4.8501E-02	1.50E+08
44	Sr 90	8.9937E+08	5.9155E-02	7.73E+06
45	Sr 91	3.4200E+04	5.9163E-02	1.82E+08
46	Sr 92	9.7560E+03	5.9482E-02	1.83E+08
47	Sr 93	4.5600E+02	6.2663E-02	1.93E+08
48	Sr 94	7.8000E+01	6.0148E-02	1.85E+08
49	Y 90	2.3040E+05	5.9157E-02	1.82E+08
50	Y 91m	2.9826E+03	3.6685E-02	1.13E+08
51	Y 91	5.0553E+06	5.9171E-02	1.82E+08
52	Y 92	1.2744E+04	5.9560E-02	1.84E+08
53	Y 93	3.6360E+04	6.3667E-02	1.96E+08
54	Y 94	1.1460E+03	6.4079E-02	1.98E+08
55	Y 95	6.4200E+02	6.4304E-02	1.98E+08
56	Y 96	1.3800E+02	5.9745E-02	1.84E+08
57	Zr 95	5.5279E+06	6.4593E-02	I.99E+08
58	Zr 97	6.1200E+04	5.9446E-02	1.83E+08
59	Nb 95	3.0370E+06	6.4594E-02	I.99E+08
60	Nb 97m	6.0000E+01	5.5931E-02	1.72E+08
61	Nb 97	4.3260E+03	5.9603E-02	1.84E+08
62	Mo 99	2.3832E+05	6.1327E-02	1.89E+08
63	Tc 99m	2.1672E+04	5.3968E-02	1.66E+08
64	Tc 99	6.7210E+12	6.1327E-02	1.10E+03
65	Tc 101	7.5200E+02	5.0440E-02	1.56E+08
66	Ru 103	3.4214E+06	3.1351E-02	9.67E+07
67	Ru 105	1.5984E+04	9.8670E-03	3.04E+07
68	Ru 106	3.1882E+07	3.9171E-03	8.52E+06
69	Ru 107	2.5200E+02	1.7339E-03	5.35E+06
70	Rh 103m	3.4200E+03	3.1038E-02	9.57E+07
71	Rh 105m	4.5000E+01	2.6641E-03	8.22E+06
72	Rh 105	1.2730E+05	9.8670E-03	3.04E+07
73	Rh 106	3.0400E+01	3.9171E-03	1.21E+07
74	Rh 107	1.3020E+03	1.7340E-03	5.35E+06
75	Sn 130	2.2320E+02	9.0010E-03	2.78E+07
76	Sb 127	3.0931E+05	1.2844E-03	3.96E+06
77	Sb 129	1.5525E+04	6.3718E-03	1.96E+07
78	Sb 130m	3.7800E+02	1.1157E-02	3.44E+07
79	Sb 130	2.4000E+03	2.9877E-03	9.21E+06
80	Sb 133	1.6200E+02	2.2346E-02	6.89E+07

Table 12.2-12 Core Inventory
(Page 3 of 3)

	Isotope	Half-Life (Sec.)	Fisson Yield	Inventory (ci)
81	Te 125m	5.0110E+06	6.7480E-05	2.08E+05
82	Te 127m	9.4176E+06	2.2418E-04	6.80E+05
83	Te 127	3.3660E+04	1.2799E-03	3.95E+06
84	Te 129m	2.9030E+06	1.9080E-03	5.88E+06
85	Te 129	4.1760E+03	6.2156E-03	1.92E+07
86	Te 131m	1.0800E+05	3.5440E-03	1.09E+07
87	Te 131	1.5000E+03	2.5405E-02	7.83E+07
88	Te 132	2.8080E+05	4.1877E-02	1.29E+08
89	Te 133m	3.3240E+03	3.9298E-02	1.21E+08
90	Te 133	7.4700E+02	3.0283E-02	9.34E+07
91	Te 134	2.5080E+03	6.7648E-02	2.09E+08
92	Ba 137m	1.5312E+02	5.9248E-02	1.83E+08
93	Ba 139	5.0940E+03	6.4816E-02	2.00E+08
94	Ba 140	1.1059E+06	6.3164E-02	1.95E+08
95	Ba 141	1.0962E+03	5.8670E-02	1.81E+08
96	Ba 142	6.4200E+02	5.8292E-02	1.80E+08
97	La 140	1.4497E+05	6.3221E-02	1.95E+08
98	La 141	1.4148E+04	5.8868E-02	1.82E+08
99	La 142	5.5620E+03	5.9304E-02	1.83E+08
100	La 143	8.4000E+02	5.9369E-02	1.83E+08
101	Ce 141	2.8080E+06	5.8868E-02	1.82E+08
102	Ce 143	1.1880E+05	5.9687E-02	1.84E+08
103	Ce 144	2.4538E+07	5.4554E-02	1.34E+08
104	Ce 145	1.8000E+02	3.9396E-02	1.21E+08
105	Pr 143	1.1733E+06	5.9687E-02	1.84E+08
106	Pr 144	1.0368E+03	5.4555E-02	1.68E+08
107	Pr 145	2.1528E+04	3.9408E-02	1.22E+08
108	Np 239	2.0300E+05	0.0000E+00	0.00E+00

Table 12.2-13 Deleted By Amendment 95I-

Table 12.2-14 Irradiated Ag-In-Cd Control Rod Sources

Control Rod, Ag-In-Cd Tip

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm^3-sec)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20-0.40	2.3×10^8	2.3×10^8	2.2×10^8	1.4×10^8	8.5×10^7	1.5×10^6
0.40 - 0.90	1.1×10^{12}	1.1×10^{12}	1.0×10^{12}	6.6×10^{11}	4.0×10^{11}	7.1×10^9
0.90 - 1.35	2.0×10^{11}	1.9×10^{11}	1.8×10^{11}	1.2×10^{11}	7.2×10^{10}	1.3×10^9
1.35 - 1.80	3.7×10^{11}	3.7×10^{11}	3.4×10^{11}	2.3×10^{11}	1.4×10^{11}	2.5×10^9

The absorber cross-sectional area is 0.589 square centimeters per rod.
The absorber material density is 10.17 grams per cubic centimeter.

Control Rod Cladding (Type 304 Stainless Steel)

Energy Group (Mev/gamma)	Source Strength at Time After Shutdown (Mev/cm^3-sec)					
	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20-0.40	7.1×10^9	6.1×10^9	3.4×10^9	8.3×10^7	9.9×10^5	0
0.40 - 0.90	3.1×10^{10}	2.9×10^{10}	2.6×10^{10}	1.2×10^{10}	6.4×10^9	2.3×10^8
0.90 - 1.35	1.4×10^{11}	2.3×10^{11}	2.3×10^{11}	2.1×10^{11}	2.0×10^{11}	1.2×10^{11}
1.35 - 1.80	1.9×10^8	1.8×10^8	1.4×10^8	3.3×10^7	5.4×10^6	0

The cladding cross-sectional area per rod is 0.136 square centimeter.
Maximum cobalt is 0.12 weight percent.

Table 12.2-15 Deleted By Amendment 84

Table 12.2-16 Deleted By Amendment 84

Table 12.2-17 Deleted By Amendment 84

Table 12.2-18 Deleted By Amendment 84

Table 12.2-19 Estimated Average Airborne Radioactivity Concentrations In The Containment Building (Page 1 of 3)

	LOWER COMPARTMENT		UPPER COMPARTMENT	
Isotope	Concentration 6 Hour Purge (μCi/cc)	10CFR20 DAC Fraction	Concentration 6 Hour Purge (μCi/cc)	10CFR20 DAC Fraction
Kr-85m	2.835E-06	1.418E-01	2.77E-07	1.388E-02
Kr-85	6.933E-06	6.933E-02	1.826E-06	1.826E-02
Kr-87	2.135E-06	4.270E-01	1.695E-07	3.390E-02
Kr-88	4.692E-06	2.346E+00	4.347E-07	2.174E-01
Xe-131m	1.339E-05	3.348E-02	2.165E-06	5.413E-03
Xe-133m	1.334E-06	1.334E-02	1.565E-07	1.565E-03
Xe-133	4.879E-05	4.879E-01	6.502E-06	6.502E-02
Xe-135m	8.897E-07	9.886E-02	3.521E-08	3.912E-03
Xe-135	1.586E-05	1.586E+00	1.644E-06	1.644E-01
Xe-137	7.7875E-08	7.787E-01	1.045E-09	1.045E-02
Xe-138	7.731E-07	1.933E-01	2.870E-08	7.175E-03
Br-84	4.381E-11	2.191E-06	2.551E-12	1.276E-07
I-131	2.519E-10	1.260E-02	3.677E-11	1.839E-03
I-132	9.077E-10	3.026E-04	8.155E-11	2.718E-05
I-133	7.190E-10	7.190E-03	7.822E-11	7.822E-04
I-134	1.144E-09	5.720E-05	8.129E-11	4.065E-06
I-135	1.271E-09	1.816E-03	1.287E-10	1.839E-04
Rb-88	2.98E-06	9.727E-02	3.864E-07	1.288E-02
Cs-134	9.917E-12	2.479E-04	2.587E-12	6.468E-05
Cs-136	9.713E-13	3.238E-06	1.609E-13	5.363E-07
Cs-137	1.345E-11	2.242E-04	3.551E-12	5.918E-05
H-3	2.607E-06	1.304E-01	6.867E-07	3.434E-02
Na-24	4.627E-11	2.314E-05	4.937E-12	2.469E-06
Cr-51	3.776E-12	4.720E-07	7.603E-13	9.504-08
Mn-54	2.230E-12	7.433E-06	5.724E-13	1.908E-06
Fe-55	1.694E-12	2.118E-06	4.432E-13	5.540E-07
Fe-59	3.829E-13	3.829E-06	8.444E-14	8.444E-07
Co-58	6.075E-12	2.025E-05	1.427E-12	4.757E-06

Table 12.2-19 Estimated Average Airborne Radioactivity Concentrations In The Containment Building (Page 2 of 3)

	LOWER COMPARTMENT		UPPER COMPARTMENT	
Isotope	Concentration 6 Hour Purge (µCi/cc)	10CFR20 DAC Fraction	Concentration 6 Hour Purge (µCi/cc)	10CFR20 DAC Fraction
Co-60	7.525E-13	7.525E-05	1.977E-13	1.977E-05
Zn-65	7.090E-13	7.090E-06	1.806E-13	1.806E-06
Sr-89	1.800E-13	3.000E-06	4.045E-14	6.742E-07
Sr-90	1.704E-14	8.520E-06	4.492E-15	2.246E-06
Sr-91	9.265E-13	9.265E-07	9.619E-14	9.619E-08
Y-90	1.704E-14	5.680E-08	4.493E-15	1.498E-08
Y-91m	4.96-13	7.066E-09	5.442E-14	7.774E-10
Y-91	9.133E-15	1.827E-07	2.923E-15	5.846E-08
Y-93	4.064E-12	4.064E-06	4.236E-13	4.236E-07
Zr-95	5.109E-13	1.022E-05	1.185E-13	2.370E-06
Nb-95	4.043E-13	8.086E-07	1.086E-13	2.172E-07
Mo-99	6.554E-12	1.092E-05	7.870E-13	1.312E-06
Tc-99m	4.998E-12	8.330E-08	6.260E-13	1.043E-08
Ru-103	9.461E-12	3.154E-05	2.045E-12	6.817E-06
Ru-106	1.260E-10	2.520E-02	3.249E-11	6.498E-03
Rh-103m	9.465E-12	1.893E-08	2.047E-12	4.094E-09
Rh-106	1.260E-10	1.260E-03	3.249E-11	3.249E-04
Te-129m	2.361E-13	2.361E-06	4.951E-14	4.951E-07
Te-129	1.726E-11	5.753E-07	1.357E-12	4.523E-08
Te-131m	1.507E-12	7.535E-06	1.680E-13	8.400E-07
Te-131	3.767E-12	1.884E-06	2.115E-13	1.058E-07
Te-132	1.748E-12	1.942E-05	2.145E-13	2.383E-06
Ba-137m	1.269E-11	1.269E-04	3.342E-12	3.342E-05
Ba-140	1.464E-11	2.440E-05	2.415E-12	4.025E-06
La-140	2.701E-11	5.402E-05	3.939E-12	7.878E-06
Ce-141	1.859E-13	9.295E-07	3.873E-14	1.937E-07
Ce-143	2.814E-12	4.020E-06	3.159E-13	4.513E-07
Ce-144	5.577E-12	9.295E-04	1.428E-12	2.380E-04

Table 12.2-19 Estimated Average Airborne Radioactivity Concentrations In The Containment Building (Page 3 of 3)

	LOWER COMPARTMENT		UPPER COMPARTMENT	
Isotope	Concentration 6 Hour Purge (µCi/cc)	10CFR20 DAC Fraction	Concentration 6 Hour Purge (µCi/cc)	10CFR20 DAC Fraction
Pr-143	3.222E-12	1.074E-05	5.580E-13	1.860E-06
Pr-144	5.577E-12	1.115E-07	1.428E-12	2.856E-08
Np-239	2.241E-12	2.490E-06	2.641E-13	2.934E-07
Total	8.738E-05	6.454E+00	1.416E-05	5.987E-01

Table 12.2-20 Estimated Average Airborne Equilibrium Radioactivity Concentrations In The Turbine Building
(Page 1 of 3)

Isotope	Turb. Bldg Equilibrium Conc. (µCi/cc)	10CFR20 DAC Fraction
Kr-85m	3.29E-14	1.64E-09
Kr-85	5.09E-14	5.09E-10
Kr-87	2.77E-14	5.54E-09
Kr-88	5.64E-14	2.82E-08
Xe-131m	1.24E-13	3.09E-10
Xe-133m	1.42E-14	1.42E-10
Xe-133	4.85E-13	4.85E-09
Xe-135m	2.22E-14	2.46E-09
Xe-135	1.76E-13	1.75E-08
Xe-137	2.85E-15	2.85E-08
Xe-138	1.77E-14	4.43E-09
Br-84	7.50E-16	3.75E-11
I-131	1.30E-14	6.52E-07
I-132	2.99E-14	9.98E-09
I-133	3.71E-14	3.71E-07
I-134	2.45E-14	1.22E-09
I-135	5.64E-14	8.06E-08
Rb-88	1.61E-14	5.37E-10
Cs-134	2.18E-15	5.45E-08
Cs-136	2.57E-16	8.56E-10
Cs-137	2.82E-15	4.70E-08
H-3	9.24E-10	4.62E-05
Na-24	8.54E-15	4.27E-09
Cr-51	6.99E-16	8.73E-11
Mn-54	3.66E-16	1.22E-09
Fe-55	2.77E-16	3.47E-10
Fe-59	6.77E-17	6.77E-10
Co-58	1.04E-15	3.48E-09
Co-60	1.22E-16	1.22E-08

Table 12.2-20 Estimated Average Airborne Equilibrium Radioactivity Concentrations In The Turbine Building
(Page 2 of 3)

Isotope	Turb. Bldg Equilibrium Conc. (µCi/cc)	10CFR20 DAC Fraction
Zn-65	1.11E-16	1.11E-09
Sr-89	3.22E-17	5.36E-10
Sr-90	2.77E-18	1.39E-09
Sr-91	1.61E-16	1.61E-10
Y-90	2.77E-15	9.23E-09
Y-91m	2.75E-17	3.93E-13
Y-91	1.23E-18	2.45E-11
Y-93	7.01E-16	7.01E-10
Zr-95	8.76E-17	1.75E-09
Nb-95	6.32E-17	1.26E-10
Mo-99	1.34E-15	2.23E-09
Tc-99m	6.80E-16	1.13E-11
Ru-103	1.78E-15	5.92E-09
Ru-106	2.00E-14	3.99E-06
Rh-103m	1.78E-15	3.55E-12
Rh-106	1.85E-14	1.85E-07
Te-129m	4.33E-17	4.33E-10
Te-129	1.27E-15	4.22E-11
Te-131m	3.04E-16	1.52E-09
Te-131	1.65E-16	8.24E-11
Te-132	3.68E-16	4.09E-09
Ba-137m	2.70E-15	2.71E-08
Ba-140	2.88E-15	4.81E-09
La-140	5.17E-15	1.03E-08
Ce-141	3.44E-17	1.72E-10
Ce-143	5.74E-16	8.20E-10
Ce-144	9.09E-16	1.51E-07
Pr-143	5.76E-16	1.92E-09
Pr-144	9.09E-16	1.82E-11

Table 12.2-20 Estimated Average Airborne Equilibrium Radioactivity Concentrations In The Turbine Building
(Page 3 of 3)

Isotope	Turb. Bldg Equilibrium Conc. (µCi/cc)	10CFR20 DAC Fraction
Np-239	4.70E-16	5.22E-10
		Total 5.20E-05

Table 12.2-21 Estimated Average Airborne Equilibrium Radioactivity Concentrations In The Auxiliary Building

Isotope	Aux. Bldg Equilibrium Conc. (µCi/cc)	10CFR20 DAC Fraction
Kr-85m	2.30E-09	1.15E-04
Kr-85	3.70E-09	3.70E-05
Kr-87	1.99E-09	3.98E-04
Kr-88	3.95E-09	1.97E-03
Xe-131m	9.09E-09	2.27E-05
Xe-133m	9.94E-10	9.94E-06
Xe-133	3.51E-08	3.51E-04
Xe-135m	1.21E-09	1.34E-04
Xe-135	1.24E-08	1.24E-03
Xe-137	1.46E-10	1.46E-03
Xe-138	1.07E-09	2.69E-04
Br-84	4.72E-12	2.36E-07
I-131	1.70E-11	8.49E-04
I-132	7.50E-11	2.50E-05
I-133	5.27E-11	5.27E-04
I-134	1.10E-10	5.49E-06
I-135	9.68E-11	1.38E-04
Rb-88	1.37E-09	4.58E-05
Cs-134	4.21E-14	1.05E-06
Cs-136	5.17E-15	1.72E-08
Cs-137	5.59E-14	9.32E-07
H-3	1.39E-09	6.95E-05
Na-24	2.82E-13	1.41E-07
Cr-51	1.86E-14	2.32E-09
Mn-54	9.58E-15	3.19E-08
Fe-55	7.18E-15	8.98E-09
Fe-59	1.80E-15	1.80E-08
Co-58	2.76E-14	9.20E-08
Co-60	3.18E-15	3.18E-07

Table 12.2-21 Estimated Average Airborne Equilibrium Radioactivity Concentrations In The Auxiliary Building

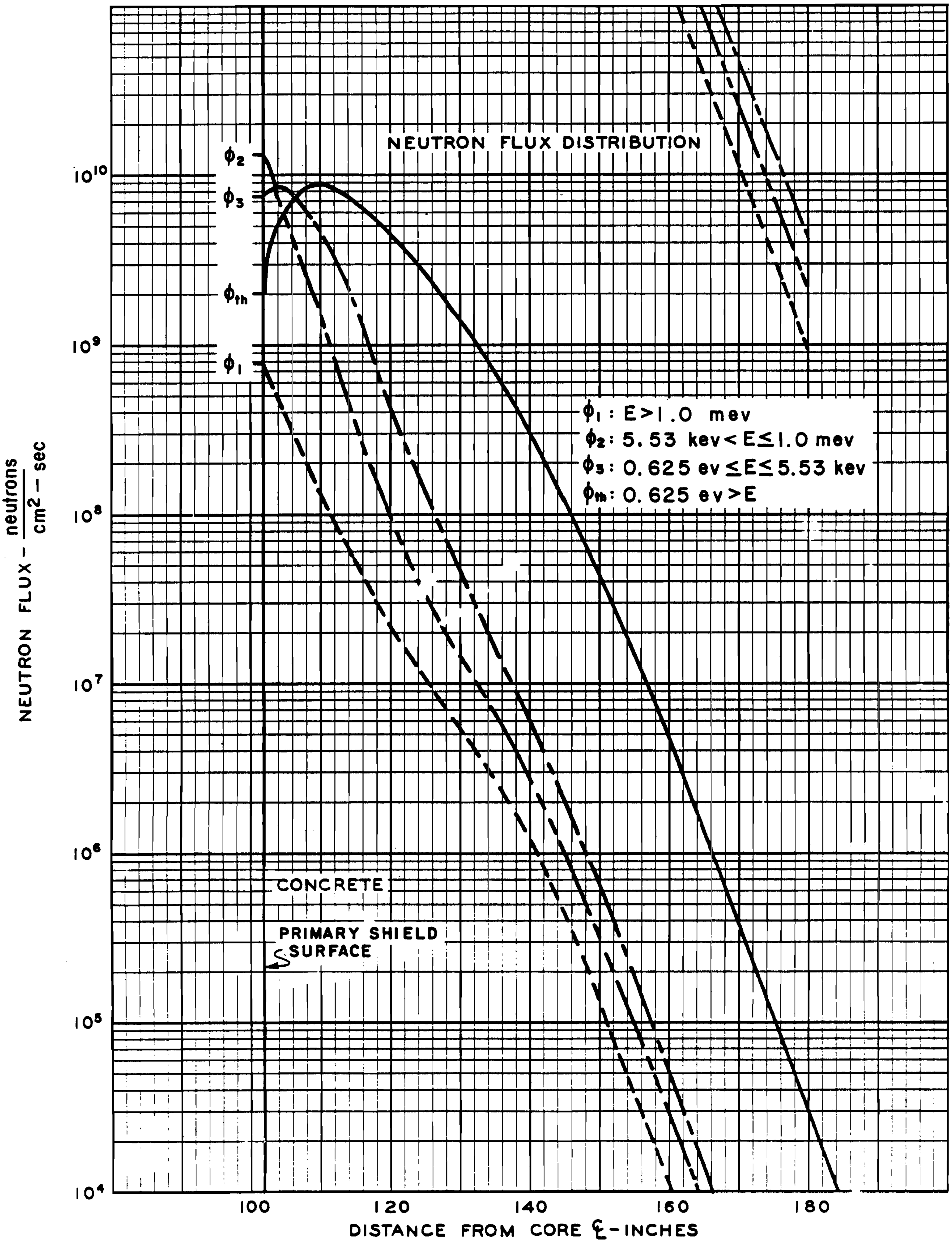
Zn-65	3.06E-15	3.06E-08
Sr-89	8.38E-16	1.40E-08
Sr-90	7.18E-17	3.59E-08
Sr-91	5.72E-15	5.72E-09
Y-90	7.18E-17	2.39E-10
Y-91m	2.89E-15	4.13E-11
Y-91	3.18E-17	6.36E-10
Y-93	2.50E-14	2.50E-08
Zr-95	2.34E-15	4.67E-08
Nb-95	1.68E-15	3.36E-09
Mo-99	3.84E-14	6.40E-08
Tc-99m	2.87E-14	4.78E-10
Ru-103	4.50E-14	1.50E-07
Ru-106	5.40E-13	1.08E-04
Rh-103m	4.50E-14	9.00E-11
Rh-106	5.40E-13	5.40E-06
Te-129m	1.14E-15	1.14E-08
Te-129	1.29E-13	4.30E-09
Te-131m	9.02E-15	4.51E-08
Te-131	3.46E-14	1.73E-08
Te-132	1.02E-14	1.13E-07
Ba-137m	5.33E-14	5.33E-07
Ba-140	7.81E-14	1.30E-07
La-140	1.50E-13	3.00E-07
Ce-141	9.01E-16	4.50E-09
Ce-143	1.68E-14	2.40E-08
Ce-144	2.40E-14	4.00E-06
Pr-143	1.69E-14	5.63E-08
Pr-144	2.40E-14	4.80E-10
Np-239	1.32E-14	1.47E-08
Total		7.79E-03

Table 12.2-22 Estimated Average Airborne Radioactivity Concentrations In The Instrument Room

Isotope	Concentration 6 Hour Purge (µCi/cc)	10CFR20 DAC Fraction	Isotope	Concentration 6 Hour Purge (µCi/cc)	10CFR20 DAC Fraction
Kr-85m	8.116E-08	4.058E-03	Zn-65	3.420E-14	3.420E-07
Kr-85	3.415E-07	3.415E-03	Sr-89	8.041E-15	1.340E-07
Kr-87	5.169E-08	1.034E-02	Sr-90	8.398E-16	4.199E-07
Kr-88	1.287E-07	6.435E-02	Sr-91	2.767E-14	2.767E-08
Xe-131m	4.980E-07	1.245E-03	Y-90	8.400E-16	2.800E-09
Xe-133m	4.253E-08	4.253E-04	Y-91m	1.550E-14	2.214E-10
Xe-133	11.649-06	1.649E-02	Y-91	5.141E-16	1.028E-08
Xe-135m	1.179E-08	1.310E-03	Y-93	1.217E-13	1.217E-07
Xe-135	4.733E-07	4.733E-02	Zr-95	2.326E-14	4.652E-07
Xe-137	3.740E-10	3.740E-03	Nb-95	2.016E-14	4.032E-08
Xe-138	9.666E-09	2.417E-03	Mo-99	2.111E-13	3.518E-07
Br-84	8.167E-13	4.084E-08	Tc-99m	1.650E-13	2.750E-09
I-131	8.898E-12	4.449E-04	Ru-103	4.131E-13	1.377E-06
I-132	2.429E-11	8.097E-06	Ru-106	6.125E-12	1.225E-03
I-133	2.209E-11	2.209E-04	Rh-103m	4.134E-13	8.268E-10
I-134	2.526E-11	1.263E-06	Rh-106	6.125E-12	6.125E-05
I-135	3.729E-11	5.327E-05	Te-129m	1.013E-14	1.013E-07
Rb-88	1.108E-07	3.693E-03	Te-129	4.125E-13	1.375E-08
Cs-134	4.856E-13	1.214E-05	Te-131m	4.690E-14	2.345E-07
Cs-136	3.659E-14	1.220E-07	Te-131	6.733E-14	3.367E-08
Cs-137	6.639E-13	1.107E-05	Te-132	5.687E-14	6.319E-07
H-3	1.284E-07	6.420E-03	Ba-137m	6.248E-13	6.248E-06
Na-24	1.406E-12	7.030E-07	Ba-140	5.501E-13	9.168E-07
Cr-51	1.582E-13	1.978E-08	La-140	9.529E-13	1.906E-06
Mn-54	1.081E-13	3.603E-07	Ce-141	7.944-15	3.972E-08
Fe-55	8.309E-14	1.039E-07	Ce-143	8.784E-14	1.255E-07
Fe-59	1.692E-14	1.692E-07	Ce-144	2.698E-13	4.497E-05
Co-58	2.787E-13	9.290E-07	Pr-143	1.243E-13	4.143E-07
Co-60	3.701E-14	3.701E-06	Pr-144	2.699E-13	5.398E-09

Table 12.2-22 Estimated Average Airborne Radioactivity Concentrations In The Instrument Room

	Concentration			Concentration	
Isotope	6 Hour Purge (μCi/cc)	10CFR20 DAC Fraction	Isotope	6 Hour Purge (μCi/cc)	10CFR20 DAC Fraction
			Np-239	7.156E-14	7.951E-08
			Total	3.527E-06	1.673E-01



NEUTRON FLUX DISTRIBUTION

Figure 12.2-1

Figure 12.2-1 Neutron Flux Distribution

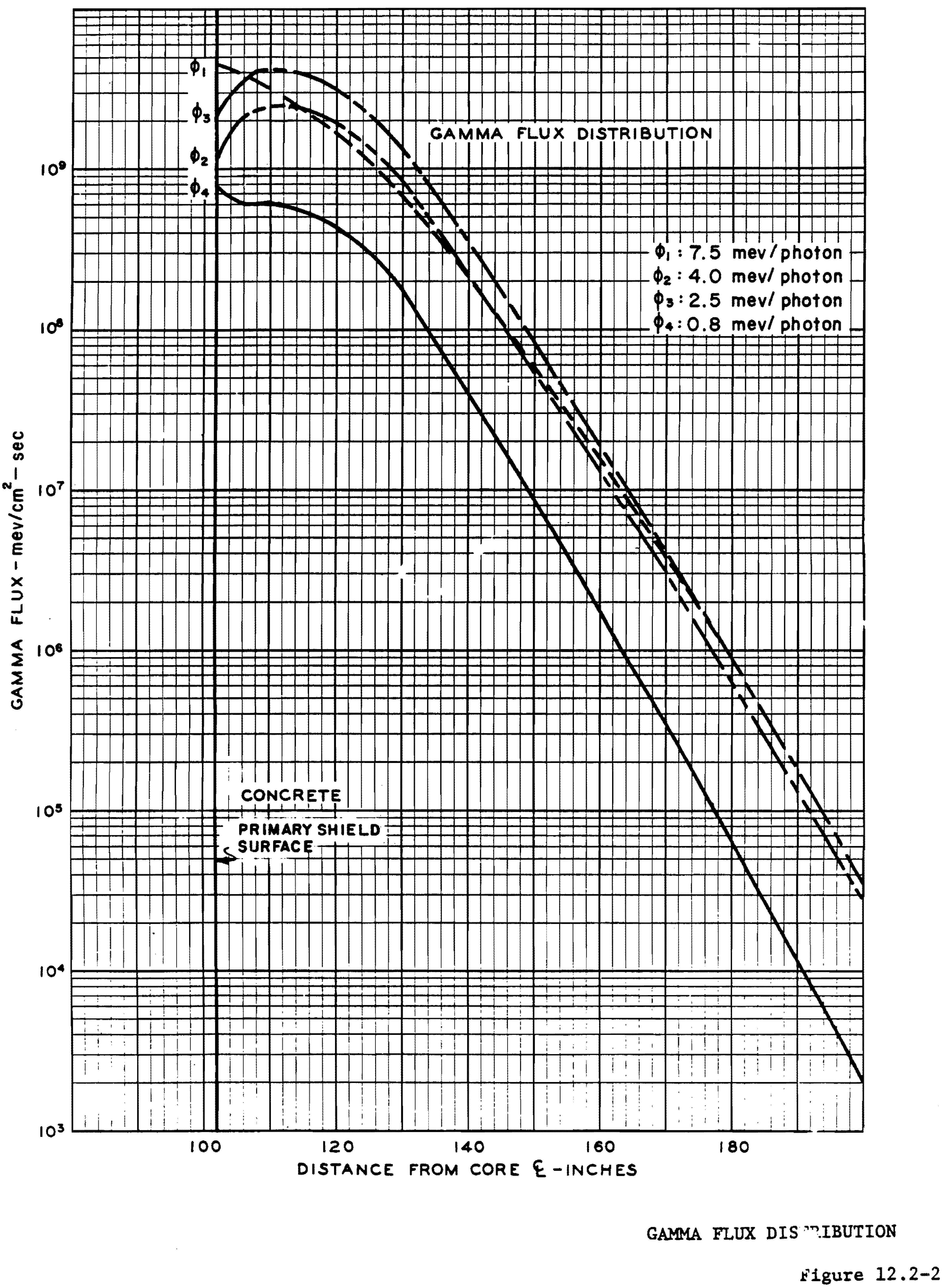
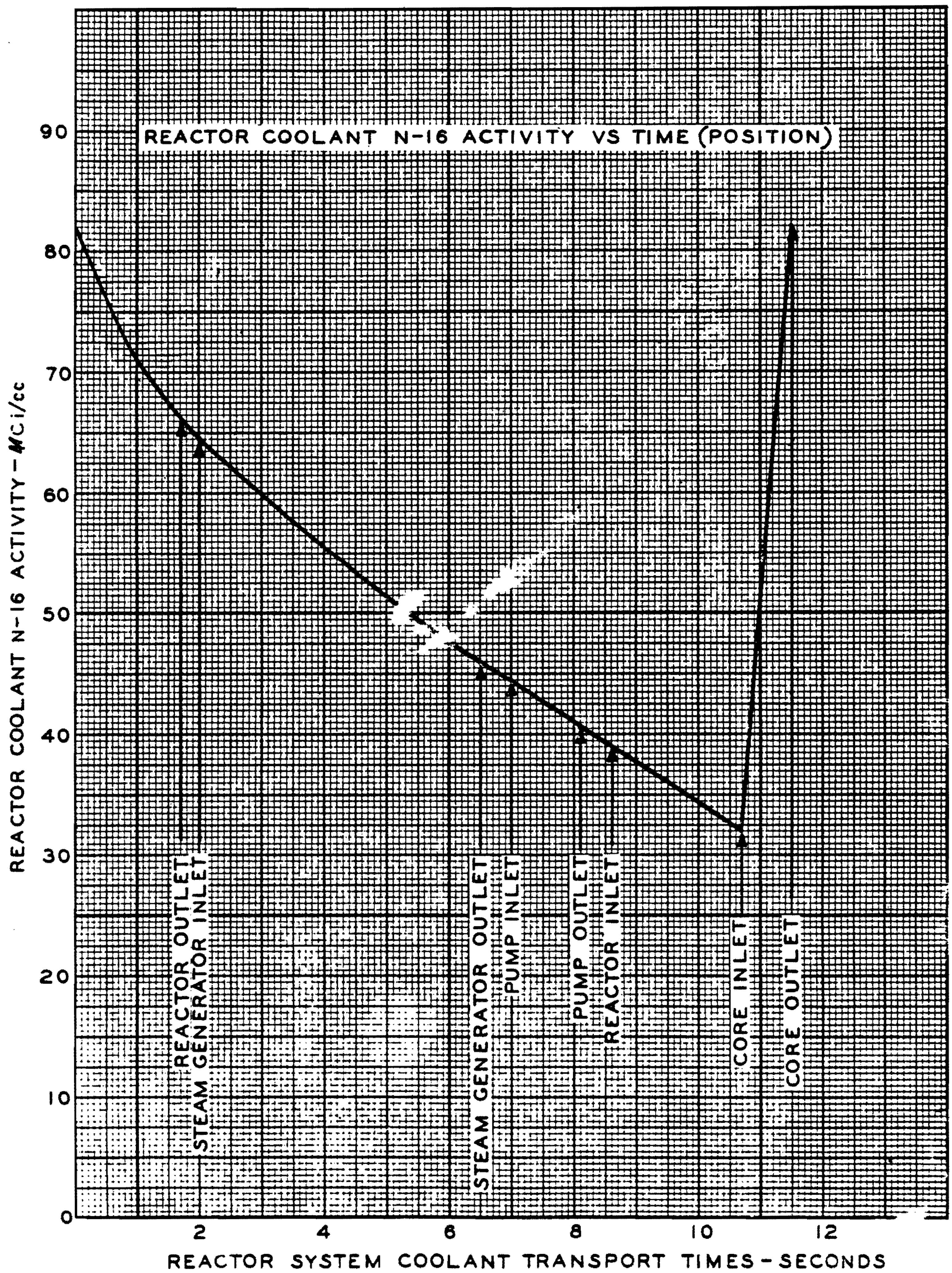


Figure 12.2-2 Gamma Flux Distribution



REACTOR COOLANT N- : ACTIVITY

Fi :e 12.2-3

Figure 12.2-3 Reactor Coolant N-16 Activity

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12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

Some specific design features to limit inplant radiation exposures are provided in the following sections.

Where practicable, instruments and components which require frequent maintenance or calibration are located in the lowest practicable radiation fields. This practice serves the twofold purpose of reducing exposure to operations personnel as well as lessening radiation damage to this equipment.

Penetrations of shielding and containment walls are located when practicable, and designed so as to minimize exposures. Details of design considerations in the location of shield wall penetrations are provided in Section 12.3.2.2.

Radiation sources and routinely occupied areas are separated wherever possible. In general, pipes or ducts containing potentially highly radioactive materials do not pass through routinely occupied areas and long runs of radioactive piping are restricted to shielded pipe chases where practicable. This piping/ductwork was considered in the development of the radiation zone maps (Figs. 12.3-1 through 12.3-15). These areas will have limited access based on actual radiation levels. In addition, an effort is made to assure that piping entering an equipment cubicle serves only the equipment housed in that cubicle.

Design features are incorporated to minimize the spread of contamination and to facilitate decontamination in the event spillage occurs. Floor drains are provided in equipment cubicles to prevent the spread of radioactive liquids. Tight fitting doors with seals are utilized in access ways to cubicles housing radioactive gas handling equipment or equipment which processes high temperature fluids. If the doors are louvered, they are equipped with back draft dampers. With the exception of floors in pipe tunnels, walls and floors in these areas are coated with special materials which are easily cleaned in case excessive leakage of contaminants has occurred.

The layout of ducts and pipes is designed to minimize buildup of contamination. Lengthy runs of horizontal radioactive piping are avoided where possible. Vents and drains on piping are located so as to minimize potential crud traps.

The Ventilation System is designed to ensure control of airborne contaminants and for easy access and service to keep doses As Low As Reasonably Achievable (ALARA) during alterations, maintenance, decontamination, and filter changes. Typically, cleaner areas are exhausted to areas of higher potential airborne radioactivity which are then exhausted to the atmosphere through air cleanup units. Air cleanup units are designed for ease of maintenance and to facilitate removal of filters to minimize personnel exposure from contaminated filters. Ventilation system design in plant buildings is discussed in Section 9.4.

Radiation monitoring devices are located throughout the plant to assist in the control of personnel exposure. Area and airborne radioactivity monitoring equipment with local readout is provided in selected areas to which personnel normally have access. Portable continuous air monitors, area monitors, and hand-held survey instruments are available to provide surveillance of areas not covered by fixed monitors and/or supplement fixed monitor surveillance. The locations of radiation monitors are discussed in detail in Section 11.4 and Section 12.3.4. Section 11.4 provides information on process and effluent radiation monitors.

Where practicable, shielding is provided between radiation sources and areas to which personnel may have routine access, and shielding is designed for maintaining doses ALARA. Where practicable, the designs include component arrangement such that those components which may require periodic maintenance are shielded from the components most likely to be high radioactive sources. Temporary shielding and convenient means for its utilization are available for use where permanent shielding is needed but impractical. Details of permanent shielding design are discussed in Section 12.3.2.

Remote handling equipment is provided wherever it is needed and practicable. Valves with remote operators are used in plant areas when manual operation is frequently required. Valve and valve gallery locations are discussed in Section 12.3.2.2. Provisions are made to remotely remove certain cartridge filters which are potentially highly radioactive. Remote handling equipment is also used to transfer new fuel and spent fuel assemblies and to the extent practicable to package and transfer liquid and solid waste products.

Instrumentation and sampling is designed to ensure that exposures will be ALARA during such routine operations as sampling off-gas, primary coolant, liquid waste, and instrument calibration. Liquid sampling from highly radioactive components is remote to a hot sampling room. Radiation protection features are designed to allow normal plant operations to continue unimpeded when radioactivity inventories are at design levels. In the initial plant design, radiation levels were based on ANSI/ANS-18.1, 1984, Revision 0, which represents approximately 1/8% failed fuel. Radiation design levels for accident conditions were based on TID-14844 methodology. Current operational radiation levels are based on ANSI/ANS-18.1, 1984, Revision 1, and accident levels are based on the Oak Ridge National Laboratory ORIGEN computer code.

Layouts of the containment and surrounding Shield Buildings and of the Auxiliary, Control, and Turbine Buildings are provided in Figures 12.3-1 through 12.3-17. While generally to scale, these drawings cannot be scaled to determine accurately the thickness of concrete shield walls. Shield wall thicknesses are therefore tabulated in Table 12.3-6.

The layouts provide the radiation zone designations including zone boundaries and maximum expected radiation levels during all phases of normal plant operation. Refueling/shutdown radiation levels will be no higher and in most areas will be much lower than those shown. These layouts show elevations and room numbers for reference to Table 12.3-6, so that shield wall thickness may be readily determined for any location. The layouts also show controlled access areas, decontamination areas, the location of the onsite laboratory for analysis of chemical and radioactivity samples and the location of the counting room. FSAR Section 12.5.2 provides additional information on radiological control (RADCON) equipment, instrumentation and facilities. The locations of area radiation monitors are tabulated according to elevation and nearest position coordinates in Table 12.3-4. Airborne radioactivity monitors are tabulated in Table 12.3-5 and special radiation monitors are tabulated in Table 12.3-7. These elevations and position coordinates are readily locatable on the layout drawings, Figures 12.3-1 through 12.3-17.

12.3.2 Shielding

12.3.2.1 Design Objectives

The design objectives of the plant shielding are the following:

(1) During normal operation, including anticipated operational occurrences, to restrict occupational doses to the 10 CFR 20 limits and insure operational radiation dose to personnel is ALARA.

(2) To restrict off-site exposures in accordance with the As Low As Reasonably Achievable (ALARA) provisions in 10 CFR 50.

(3) To limit, under accident conditions, the off-site exposure from activity in the containment so that the total exposure from this source and from airborne radiation will not exceed the 10 CFR 100 exposure limits.

(4) To satisfy the requirements of 10 CFR 50, Appendix A, Criterion 19, sufficient radiation protection is provided to permit access and occupancy of the main control room under accident conditions without personnel receiving excessive radiation exposure. The design also provides limited access, defined in Section 12.3.2.2, to other plant areas during accident conditions. The dose an operator receives during any such extra-control room mission and the dose received while gaining access to and occupying the main control room will not exceed exposures of 5 rem deep dose equivalent (DDE).

12.3.2.2 Design Description

Plant Shielding

Expected frequency, duration of occupancy, and access controls determine what exposure rates will be allowed in all interior and other on-site areas in order to assure that shielding design objectives will be met. Each area is classified as one of seven types listed on Figure 12.3-1.

In numerous cases where access requirements are expected to range from almost continuous occupancy to a few hours per week (access Types II and III), shielding is required to achieve acceptable exposure rate levels. The description of the shielding applied to the plant design does not establish parameters under which personnel access to areas in the plant will be controlled. Control of personnel access will be in accordance with actual radiological conditions and will be controlled as described in Section 12.5, Radiological Control (RADCON) Program.

Shield Walls

Presented in this section are the criteria for the erection of the plant shield walls and for penetrations through these walls. The calculational methods used to determine the thickness and other dimensions of the shield walls are given at the end of this section. See Section 12.2.1 for the relationship between the Sequoyah and Watts Bar shielding design source terms.

Many structural walls also serve a shielding requirement which often sets the wall thickness. Some walls serve only a shielding function. Most of these shielding walls are cast in place up to within 2 inches of the ceiling above. When necessary, this gap between wall and ceiling is filled over part of the wall thickness with grout. Those shield walls or portions of shield walls that are subject to removal for equipment repair or replacement are constructed of solid concrete blocks.

Except for two applications, which are cited in later subsections, the poured concrete shield walls throughout the plant are ordinary concrete with a minimum density of 145.0 lb/ft^3.

Areas where design DDE exposure rates are greater than 5 mrem/hr incorporate permanent shielding, to the degree feasible, to prevent the need for temporary shielding.

Access to many equipment enclosures is provided through the sidewalls of the compartments. In these cases, the effectiveness of the shield walls in limiting exposure rates outside the equipment enclosures is maintained by providing labyrinth entrances. Access to some equipment enclosures, principally filter, and demineralizer is through the floor above. In these cases, the removable concrete floor slab that provides the entrance generally has the same thickness as the cubicle walls.

The design criterion for shield wall penetrations in the Auxiliary Building, such as those for piping and ventilation ducts, is to locate them whenever practical so that their effect on the DDE exposure rates in accessible areas outside the shielded enclosure is minimized. Often this criterion is satisfied by locating the penetrations as nearly as possible to the corners and to the ceiling of the shielded enclosure. In using this technique, however, consideration is given to the increased length of piping sources that may result. If direct or reflected radiation passing through the penetrations of a shield wall creates a radiation area outside the wall, the criteria given for the erection of shield walls are used to establish the necessity for a wall to shield this area.

The following general shielding considerations are employed in the arrangement of Shield Building penetrations:

(1) Where practical, most penetrations of the Shield Building except those that connect the Shield Building to a shielded enclosure in the Auxiliary Building, are opposite unpenetrated areas of the crane wall. This arrangement

adequately shields outside areas and areas inside the Auxiliary Building from sources inside the containment shell during normal operation. When this arrangement is not used, shadow shields or high-density silicon elastomer fill are provided to eliminate radiation streaming from major sources inside the containment to areas outside the Shield Building.

(2) Radiation sources in the annulus between the containment and the Shield Building are located behind unpenetrated portions of the Shield Building or behind the Shield Building penetrations that connect the Shield Building to shielded enclosures in the Auxiliary Building.

(3) Penetrations of crane wall sections that provide necessary shielding for containment areas accessible during power operations are avoided when possible.

(4) Shadow shields or high-density silicon elastomer fill are provided at Shield Building penetrations that connect the Shield Building to unshielded areas where access cannot be completely controlled during accident conditions.

Valve and Valve Operation Stations

The following arrangement guidelines were used in the initial plant layout for manually operated valves that control process equipment function. The radiation dose rates specified were estimated at the time the initial layout was performed. Actual radiation dose to personnel operating valves will be controlled by radiation work permit, which includes surveys to determine actual radiation dose at the work location and limitation of time spent at that location.

(1) Valves located and operated in the enclosure with the controlled equipment:

(a) Used when the anticipated whole body exposure rate at the valve would be no greater than 0.1 rem/hr based on design level activities in the equipment and piping and anticipated occupancy for valve operation.

(b) Used when sources and piping in the equipment enclosure could be sufficiently removed, without economic penalty, to allow valve maintenance.

(c) Used when design activities are low enough to limit personnel doses to no more than 0.006 rem/hr without source removal. (Source removal could involve pumping or draining a liquid, venting a gas, flushing demineralizer resin, or replacing a filter cartridge.)

(d) Used for manually operated valves which isolate, drain, or vent process equipment such as pumps that contain relatively low amounts of activity.

(2) Valves located and operated in a radiation area outside the equipment enclosure.

(a) Used when whole body exposure rates at the valve must be less than 0.1 rem/hr.

(3) Valves located in the equipment enclosure but are operated from behind a shield wall.

(a) Used when whole body exposure rates at the valve operating station would be less than 0.015 rem/hr. The anticipated exposure limitations during valve maintenance would be the same as those for the first arrangement.

(4) Valves located in a valve gallery:

(a) Typically most of the valves that serve a few identical or similar plant components share a valve gallery.

(b) One side of the valve gallery forms a shield wall which separates the valves from the process equipment. The opposite side of the gallery forms a shield wall penetrated by either extension stem arrangements joining valves to hand-wheel operators or by flexible shaft controls.

(c) Extension stems are solid metal and the annular space between extension stem housing and shield wall sleeve was grout filled. With this arrangement, the effectiveness of the shield wall between valves and hand-wheel operators would be virtually undisturbed and the whole body exposure rates at the hand-wheel would be less than 0.001 rem/hr.

(d) A flexible shaft control was used in the case of a few filters. The shafts followed an oblique or curved path through the wall to prevent direct radiation streaming from high intensity sources. The design whole body exposure level outside the valve gallery for this arrangement was 0.0025 rem/hr.

(e) Some of the design guides used to provide for valve maintenance without first removing the sources from the process equipment include:

(1) Penetrations through the shield wall between the equipment enclosures and valve gallery were placed as near the ceiling and as close to the corner of the equipment enclosures as practical.

(2) Piping runs in the gallery, that will contain radioactive fluid when the control valve was isolated for maintenance, were kept as short as practical.

(3) Excessive annular spaces between pipe and pipe sleeve in the wall between equipment and valves were avoided.

With these precautions, the design objective of 0.006 rem/hr whole body exposure rate would be achieved when the process equipment contained up to a significant fraction of design level activity. As an outside limit, the design ensured an exposure rate of less than 0.1 rem/hr in the valve gallery during valve maintenance without removal of the process equipment sources. Even at exposure levels of 0.1 rem/hr some valve inspection and maintenance would be possible.

(f) Motor operated or pneumatic valves that isolated, drained, or vented process equipment if activity levels would prohibit emergency access to the valves.

(g) Advantages of using valve galleries include:

(1) In the unlikely event of valve operator failure, the arrangement allows limited direct operation at a valve location until maintenance could be performed.

(2) The arrangement provides a second shield between process equipment and general access areas.

(3) The arrangement eliminates the need for removal of the process source and decontamination of the valve enclosure before beginning valve maintenance.

Most of the advantages of locating hand-operated valves in valve galleries also apply to the location of remote-manual (motor operated or pneumatically-operated) valves in valve galleries.

Manually operated valves used to isolate, drain, or vent process equipment such as pumps that contain relatively small amounts of activity are generally located and operated in the enclosure with the equipment. As a rule, remote valve operation is a design consideration when the anticipated dose from the process equipment during valve operation is significant. Remote valve location is a design consideration when the anticipated dose from the equipment during valve maintenance is significant, meaning a large dose could be received if local valve station were used.

A valve is never used to isolate, drain, or vent process equipment located and operated in the enclosure with the equipment if the whole body exposure level is greater than 0.1 rem/hr. The limit selected for each valve depends on the expected occupancy time at the valve station and is generally much less than 0.1 rem/hr. If anticipated exposure rates are too high to allow location and manual operation of the valve in the enclosure with the equipment, one of the following procedures is used: (1) the operation of the valve is from behind a shield wall which limits the whole body exposure rate at the operating location to less than 0.015 rem/hr or (2) the valve is located in a valve gallery and operation of the valve is from behind the valve gallery wall which restricts the whole body exposure rate at the valve operating location to 0.0025 rem/hr. Typically, these valves share a valve gallery with the equipment control valves.

Primary and Secondary Shielding

The primary shield consists of the following parts:

(1) Shield elements inside the reactor pressure vessel. These elements, which are the core baffle, the core barrel, neutron pads, and water annuli, provide a water shield and a steel shield, each several inches thick.

(2) The reactor pressure vessel.

(3) A concrete structure surrounding the reactor vessel from the floor at the 702.78 foot elevation to the floor at the 725.12 foot elevation. The concrete thickness is 5 feet 9 inches on the radius through each of eight out-of-core neutron detector slots. On other radii, the concrete thickness opposite the active fuel is 8 feet 6 inches. There is an opening in the shield at each of the eight primary coolant pipes. Four of the openings start at the vessel flange surface elevation of 725.12 feet and go down to Elevation 712.71 feet. The other four openings extend from the vessel flange surface to Elevation 715.00 feet.

That part of the opening above each pipe (nozzle inspection ports) is filled during power operation with a removable plug. Removal of the plugs during shutdown allows inspection of the weld joints between the primary coolant pipes and the reactor vessel nozzles. Inspection time available will be very limited since exposure levels under pressure vessel equilibrium Co-60 and Fe-59 activity conditions will be on the order of 10 rem/hr at the bottom of the opening and 1 rem/hr at the top.

Except across the refueling canal, the primary concrete structure extends upward at reduced thickness (minimum is 2'6") from the 725.12 feet elevation to the operating floor (Elevation 756.63 feet). (The blowout panels in this upper structure are located just under the floor at Elevation 756.63 feet. The panels extend from Elevation 754.13 feet down to Elevation 749.63 feet. With this arrangement, radiation from the reactor vessel that penetrates the blowout panel area is attenuated by at least one reflection off concrete before it reaches accessible plant areas outside the primary concrete).

The upper part of the primary concrete shielding is completed by the walls of the refueling canal which extend upward from Elevation 709.23 feet, by the control rod drive missile shield and by a gate which spans the refueling canal from Elevation 756.63 feet down to Elevation 725.12 feet. The control rod drive missile shield and the gate are removed during refueling. The primary shielding makes possible necessary access inside the crane wall during shutdown.

The secondary shield consists principally of the crane wall, the Shield Building, the concrete operating floor at Elevation 756.63 feet, and the concrete structures which combine with the crane wall to enclose those sections of the steam generators and the portion of the pressurizer that extend above Elevation 756.63 feet.

In addition to their providing biological radiation protection, the primary and secondary shielding are arranged and structured to provide additional shielding functions such as:

(1) The primary shielding elements inside the vessel attenuate neutron flux sufficiently to prevent excessive radiation damage to the reactor vessel.

(2) The primary shielding prevents excessive radiation damage to plant components from neutron and gamma radiations, and the secondary shielding prevents excessive radiation damage to plant components from gamma radiation.

(3) The metal and water inside the pressure vessel and the pressure vessel itself serve to reduce the heat flux from neutron and gamma radiation at the vessel outer surface. This helps to avoid high temperatures and possible dehydration in the surrounding concrete.

(4) Parts of the primary and secondary shields serve as portions of the divider, necessary for the ice condenser containment, between lower and upper containment compartments.

(5) The Shield Building, which is part of the secondary shielding, is also part of the double containment.

Personnel enter and leave the containment vessel through either of two personnel air locks. To protect (from primary coolant system radiation) personnel entering the containment through the airlock from the platform at Elevation 716.00 feet, heavy concrete (218.0 LB/ft^3) is used in a section of the crane wall. With the reactor at significant power levels, personnel access to the lower compartment, which is access Type IVa, will be controlled based on radiological conditions. During full power operation, the upper compartment and the ice condenser upper plenum, access Type IVa areas, will be entered as appropriate based on radiological condtions for upper compartment inspection, ice bed and ice condenser inspection and maintenance. The seal table and instrument room, which are access Type IV areas, may be entered routinely during full power operation depending on radiological conditions. The accumulator rooms, ventilation equipment rooms, and tunnel area outside the crane wall may be entered from the seal table and instrument room only as needed and as radiation and airborne contamination permit. Some of these rooms contain access Type IVa areas. Access to the annulus between the containment vessel and the Shield Building is not normally required during power operation; however, access, if necessary, is through a hatch. Most annulus areas are access Type IV areas although some areas opposite crane wall penetrations are access Type IVa areas.

Auxiliary Building Shielding

Shielding in the fuel handling area of the Auxiliary Building is discussed in a following subsection. The balance of the shielding in the Auxiliary Building protects personnel, during normal operation including anticipated operational occurrences, from the components and piping of the following systems and facilities:

(1) Chemical and Volume Control System (CVCS)

(2) Waste Disposal Systems (WDS)

(3) Residual Heat Removal System (RHR)

(4) Spent Fuel Pool Cooling and Cleanup System (SFPCCS)

(5) Sampling System collection and analysis facilities

The hot instrument shop and decontamination area enclosures furnish some minimal shielding, but their main function is to minimize the spread of contamination.

The Auxiliary Building shielding is designed to limit DDE exposure levels in accessible corridors and open spaces in the building to 0.001 rem/hr (access Type II); however, exceptions occur at certain shield wall penetrations. If the exposure rate at a penetration exceeds 0.005 rem/hr, the procedures in Section 12.5.3 for designating radiation areas apply. Auxiliary Building shielding is also designed so that equipment areas may be entered for maintenance without shutdown of adjacent operating systems or system equipment. Satisfying this requirement results in a high degree of compartmentalization in the building.

Most piping carrying fluid of high specific activity is routed through shielded pipe chases. The pipe chase walls have a minimum thickness of 27 inches of concrete, which will significantly reduce the DDE exposure rate from a 14-inch Residual Heat Removal (RHR) System pipe carrying reactor coolant water. Pipe chases run along the A-5 and A-11 coordinate lines from Elevation 676.0 ft. to Elevation 757.0 ft (see Figures 12.3-4, 12.3-8, 12.3-10 and 12.3-12). The pipe chase areas are enlarged at one end between the floors at Elevation 713.0 feet and Elevation 737.0 feet to form Shield Building penetration areas. Most radioactive fluid carrying pipes running from the containments to the Auxiliary Building pass through these pipe chase sectors which extend from approximately Az 270° to approximately Az 300° (See Figure 12.3-3). Another pipe chase runs along the fuel transfer canal and adjoins the A-5 and A-11 line pipe chases between the floors at Elevation 713.0 feet and Elevation 737.0 feet. A concrete partition in this pipe chase along the A-8 line, between units, inhibits the spread of contamination from one unit to the other should a pipe rupture occur.

Fuel Transfer Shielding

During fuel transfer operations, the refueling canal and the region above the open reactor vessel are filled with borated water to Elevation 749.12 feet. The water level in the fuel transfer canal and spent fuel pit, which are in the Auxiliary Building, is also at Elevation 749.12 feet. The bottom of the refueling canal is at Elevation 709.23 feet in the fuel assembly tilting device area and at Elevation 713.87 feet elsewhere. A fuel assembly is transferred from the reactor vessel through the refueling canal toward the Auxiliary Building. It travels in a fuel transfer tube from the containment to the fuel transfer canal in the Auxiliary Building, and it is then moved into a storage location in the adjacent spent fuel pool. The reactor cavity filtration system assures water clarity in the reactor cavity during refueling.

After the fuel transfer has begun, the principal radioactive sources in the proximity of the fuel assembly transfer path are the following: (1) activity in the water which is a mixture of reactor coolant and water from the refueling water storage tank and (2) the fission product inventory in the fuel assembly being transferred. The activity in the water will not normally be above 0.01 μCi/cc of nontritium activity when a fuel assembly is moved from the vessel. If the activity is above this level, it will be reduced to this level with the Spent Fuel Pool Cooling System (SFPCS) equipment.

The minimum water shield above the active fuel region of a spent fuel assembly as it moves from the reactor vessel to the storage position in the spent fuel pool is 9.9 feet except when the assembly is in the fuel transfer tube. The design of the transfer equipment incorporates restraints to assure that this minimum water shield is maintained. The calculated DDE exposure rate to a person on the Spent Fuel Pit Bridge or the manipulator crane bridge resulting from a fuel assembly at its maximum elevation during fuel transfer activities is less than 2.5 mrem/hr. The transfer of spent fuel assemblies does not generate any high radiation areas in accessible plant areas. The minimum shielding between the fuel assembly and the emergency passageway under the fuel assembly tilting device in th refueling canal is 3 feet 0 inch of heavy concrete (218.0 LB/ft^3). The DDE dose rate in the passageway is less than 50 m rem/hr. The minimum shielding inside the primary containment between fuel assembly and personnel on the floor at Elevation 716.0 feet is 1 foot of water and over 5.5 feet of ordinary concrete. The corresponding maximum DDE exposure level is approximately 5 m rem/hr. During fuel assembly transfer, the region in the annulus between the steel containment and the Shield Building is protected from the fuel assembly by concrete and water equivalent to more than 6 feet of concrete. A small access opening is provided through the shielding in the annulus to allow for inspection of the fuel transfer tube. This opening is normally filled with solid concrete blocks which are removed only when access for inspection purposes is required. A radiation streaming gap between the steel containment and the concrete on each side of it in the vicinity of fuel transfer tube is avoided by offsetting the concrete and attaching to each side of the steel containment a steel ring. Similarly, offsets in the Shield Building concrete and in the Auxiliary Building wall in the area of the transfer tube are used to avoid a direct streaming path between these two structures.

When the spent fuel assembly is outside the Shield Building, during passage through the Auxiliary Building wall and fuel transfer canal to the spent fuel pool, it is shielded by a minimum of 6 feet of concrete or by a minimum of 9.9 feet of water. Spent fuel pool concrete walls which separate spent fuel assemblies in their storage locations from the Auxiliary Building access area at Elevation 692.0 feet are 7 feet thick.

Turbine Building and Service Building

Radioactivity in the Turbine Building occurs only in the event of steam generator primary-to-secondary leakage. Almost the entire Turbine Building is an access Type I area.

Located in the Turbine Building are condensate demineralizers and associated regeneration equipment. This equipment is adequately shielded to maintain maximum dose rates in controlled access areas to 0.001 rem/hr. Cubicles in which condensate demineralizers and associated equipment are located are generally designed for Type III or IV access (see Figures 12.3-11 and 12.3-13).

There are several areas that contain or store radioactive material in the Service Building, such as the protective clothing change room, health physics laboratory, and radiochemical laboratory filter room. These areas furnish necessary shielding, but their principal purpose is to minimize the spread of contamination. Refer to Section 12.5.2 for additional infomation cocerning radiological control (RADCON) equipment, instrumentation and facilities.

Outside Areas

Except for the following, all areas outside the plant buildings are either access Type I or II areas during normal operation including anticipated operational occurrences.

(1) As discussed in Sections 11.5.5.2 and 11.5.6, for short periods of time when solid waste shipping is imminent, the casks will be outside. The number of casks or containersallowed outside at any one time is controlled and depends on the exposure rates from each cask or container. The maximum exposure rate from each cask satisfies the provisions of 49 CFR 173.

Access to the outside region where these casks or containers are locatedfor the short period prior to shipment and during loading of the transportation vehicle is controlled. The type of control required depends on the designated access type, which in turn is established by the exposure rate.

(2) The area adjacent to the Condensate Demineralizer Waste Evaporator (CDWE) Building is used for temporary storage of dry active waste (DAW). During storage activites, controls are placed to limit dose rates from each individual container to 1 rem/hr at 30 cm.

(3) There are six outside tanks that contain radioactive liquids: Two refueling water storage tanks, two primary water storage tanks, and two condensate storage tanks. The activity in each is low level, and no shielding is required. Maximum DDE exposure rates at the site boundary from these tanks are 4.3E-7 mrem/hr, 2.1E-9 mrem/hr, and 1.3E-9 mrem/hr, respectively, for a refueling water storage tank, a primary water storage tank, and a condensate storage tank. These dose rates are calculated using computer code QAD-P5Z[4].

(4) The Old Steam Generator Storage Facility (OSGSF) is a non-safety related, non-seismic reinforced concrete structure that provides interim storage for the Old Steam Generators (OSGs) removed from the Reactor Building as a result of steam generator replacement during the Unit 1 Cycle 7 refueling outage. The OSGSF is located north of the plant, outside the protected area but within the exclusion area and site boundary. The general location of the OSGSF is shown on Figures 2.1-4b and 2.1-5.

The reinforced concrete walls and roof (minimum density of 145 lb/ft3) of the OSGSF and its access vestibule have been designed to ensure that the dose rates outside the facility are below station administrative and 10CFR20 and 40CFR190 limits. Radiation zones inside the OSGSF were calculated to range from a "radiation area" near the OSGs to "unlimited access" at the vestibule door and "unlimited access" outside the OSGSF as defined on UFSAR Figure 12.3-1. Based on measured radiation levels inside the OSGSF, the interior of the OSGSF is controlled as a radiologically controlled area.

The radiation dose assessment was accomplished using the MCNP[5] (Monte Carlo N-Particle) computer code. MCNP is a Monte Carlo program that calculates direct and skyshine doses.

Shielding For Accident Conditions

Some shielding provided for normal operation also has a function during accident conditions. However, other shielding has a function during accident conditions only. This accident shielding is required to serve two functions:

(1) It must restrict the exposure at the site boundary from activity in the containment to a small fraction of 10 CFR 100 limits and (2) it must attenuate exposure rates at interior and other onsite locations from activity in the containment to levels which will allow required access. Continuous occupancy of the main control room and the Technical Support Center is required during accident conditions. Infrequent access is required for the operational support center, radio-chem lab, and other plant areas identified below during accident conditions. Analyses have been done to ensure the following post accident activities can be accomplished with dose to workers remaining below 5 rem as required by NUREG-0737, II.B.2.

(1) Continuous main control room and Technical Support Center occupancy is required.

(2) Control or verification functions in the MG set room and/or 480 shutdown board room.

(3) Install CCS/ERCW spool piece.

(4) Refill RWST following a LOCA.

(5) Since a single crew cannot remain in the main control room for the duration of the accident, it must be possible to make the trip from the site boundary to the main control room sometime after 24 hours without receiving an excessive dose.

(6) Sampling of gaseous effluents (at the Shield Building vent monitor location) per the requirements of NUREG-0737.

(7) Sampling of the reactor coolant and the containment atmosphere per NUREG-0737 requirements.

(8) Realignment of component cooling water to the spent fuel pool cooling system.

(9) Mission to the Intake Pumping Station to place ERCW backwash screens in service.

(10) Survey of the Auxiliary Building for leaks.

(11) Survey of the main steam lines and the steam generator blowdown during a steam generator tube rupture accident.

(12) Control and verification functions in the switchyard, Diesel Generator Building, and Turbine Building.

The Shield Building is the principal structure that limits exposure at the site boundary and at site exterior locations from activity in the Containment. The Shield Building also, in concert with other shields, limits exposure levels at interior and other on-site locations. The accident shielding functions of the Shield Building are shared by the structures that shield its penetrations, such as the steam line penetrations, the personnel hatches, the equipment hatch, ventilation ducts, and the many smaller penetrations. Some of the structures that shield the Shield Building penetrations are Auxiliary Building external walls. These and other Auxiliary Building walls and the Auxiliary Building ceilings further attenuate radiation from sources within the Containment to improve accessibility during accident conditions.

The ESF equipment compartment shielding provides for emergency maintenance. (To make possible this maintenance, the equipment will be drained before the maintenance begins and the operator will wear anti-contamination clothing and have breathing protection). In the case of ESF equipment, such as the RHR pumps which also operate during normal operation, the shielding required for normal operation is controlling.

The main control room and the Technical Support Center (TSC) are shielded so that the integrated DDE dose from external sources (activity inside the primary containment, in the passing cloud and in surrounding rooms) obtained during occupancy following a loss-of-coolant accident would be a very small fraction of 5.0 rem. The major portion of the Total Effective Dose Equivalent (TEDE) dose can then come from the airborne activity within the main control room. (The dose from this airborne activity which is more difficult to limit than that from the external sources is discussed in Chapter 15 which considers integrated exposures in the main control room under accident conditions from all sources).

In the main control room's shielding design, sufficiently thick walls, ceiling, and floor are provided. In addition, special attention is given to the doorways. Shield doors are provided at the entrances from the Turbine Building to attenuate radiation from the radioactive cloud which is assumed to occupy the Turbine Building.

Analysis shows that shield doors at the small entrances from the main control room to the Auxiliary Building are not necessary.

Shielding Calculations

Shielding required to reduce the exposure rates based on conservative source strengths in known source geometries to design objective values were determined with hand calculations and/or with the SDC computer code. A computer program is used to solve the equations for the whole body beta and gamma dose rates from airborne activity. The program also provides the whole body gamma exposure rate after attenuation by a shield. Both the hand calculations and the computer codes employ the point-to-point kernel integration method. The SDC code[1] integrates the basic exponential attenuation point kernel over the various geometries to provide the uncollided gamma-ray flux. Many of the integrations found in the Reactor Shielding Design Manual[2] are utilized. Exposure rates are obtained by multiplying the uncollided flux by the product of a flux-weighted buildup factor and a dose-conversion factor. The hand calculations generally employ the more conservative procedure of multiplication of the buildup factors for the different materials between source point and exposure point.

When reflected gamma rays are important contributors to exposure rates, as in the case of labyrinth design and in the case of some shield penetrations, the angularly and energy distributed source strengths at the reflection surface are calculated using albedo techniques.

Condensate Demineralizer Waste Evaporator Building

Components of the Condensate Demineralizer Waste Evaporator are contained in a specially designed building adjacent to the Auxiliary Building near the on-site packaging area. Each component of the processing package is located in separately shielded compartments with the potentially more highly radioactive equipment further separated from equipment with less potential for radioactive contamination. Access to the building is designed for an Access Type III (Radiation Area) with radioactive components located in areas designed for Access Types IVa and IVb (High Radiation Area). The design DDE rate outside equipment cubicles is 0.001 rem/hr, for areas generally accessible on a routine basis. Layout of the building showing radiation protection design features is provided on Figure 12.3-7. See the note in Section 12.2.1.5 relative to unit operation without the Condensate Demineralizer Waste Evaporator package.

12.3.3 Ventilation

The plant ventilation systems are designed to assure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity. The ventilation systems will maintain concentrations of airborne radioactive material below 10% of the respective derived air concentrations (DAC) and annual limits on intake (ALI) in areas routinely occupied. Additionally, (in accordance with the provisions of 10 CFR 20 Paragraph 20.1701) the systems reduce concentrations of airborne radioactivity in areas not normally occupied, but where maintenance or in-service inspection has to be performed.

12.3.3.1 Airflow Control

The Watts Bar Nuclear Plant ventilation systems are designed to supply air to the relatively cleaner plant areas and to exhaust air from areas of potentially higher airborne radioactivity levels. Major plant areas that could be subjected to radiation contamination, and their associated air exhaust flow rates, are shown in Table 12.3-3. Air that is removed from these areas following an accident is passed through air cleanup units equipped with HEPA filters and charcoal adsorbers before being exhausted to the environment.

12.3.3.2 Typical System

The following is an illustrative example of the air cleanup system design. This typical system is designed to provide fresh, clean air inflow to and removal of potentially contaminated air from the Auxiliary Building to assure personnel comfort and safety during normal plant operations.

The Auxiliary Building general ventilation supply subsystem has a major impact on the personnel protection features incorporated in the design of the ventilation system. To control airborne activity, the Auxiliary Building ventilation supply air is delivered to clean areas and areas of general personnel occupancy. This air is then routed to areas of progressively greater contamination potential by natural pressure gradients induced by the exhaust system. Air is supplied as follows:

Area	
El. 782	Control Rod Drive Equipment Rooms*
El. 757	Fuel Handling Area
	EGTS Room
	Blowdown Treatment Room
	Waste Packaging Area
	CDWE Building*
El. 737	Penetration Rooms
	General Areas*
	Hot Instrument Shop
El. 730	CDWE Building*
El. 713	General Areas*
	Cask Loading Area
	Nitrogen Storage Area
	Post Accident Sampling Facility*
	Penetration Rooms

Area	
El. 692	General Areas*
	Penetration Rooms
	Spent Resin Tank Room
	Cask Decon Tank Room
El. 676	General Areas*

Air is exhausted from the Auxiliary Building rooms/areas as shown in Table 12.3-3 and in Section 9.4. Air supplied to the relatively clean areas, marked by an asterisk (*), is allowed to follow natural flow paths to air exhausts in areas of potentially greater contamination. Because of its potentially higher levels of radioactivity and requirements for personnel access, the remaining areas listed above are provided with both air supply and air exhaust. The exhaust from the Hot Instrument Shop is from a hooded area over the potentially higher radioactivity areas. Clean outside air is supplied to the Auxiliary Building air supply system through medium efficiency filter cells at each inlet plenum. The rated efficiency of each cell is 85%, based on the National Bureau of Standards atmospheric dust spot test. Each filter bank is provided with a static pressure differential indicating gauge. See Figure 12.3-16 for the general layout plan of the air intakes for this system. The exhaust air during normal operation is released to the outside through the Auxiliary Building vent.

12.3.3.3 Additional Radiation Controls

The ventilation system is designed so that filters containing radioactivity will not create a radiation exposure hazard to personnel in normally occupied areas. Normally, waste filters containing radioactive contaminants will be removed from the filter housings, transported to the waste packaging area, and stored in appropriate shipping containers to await shipment to a disposal/storage site. Filters with especially high levels of contamination may be transported to and temporarily stored in the shielded filter storage area. Although the basic design of the air cleanup units was completed prior to the publication of Regulatory Guide 1.52, good general compliance with the requirements of Section 4 of that document has been accomplished. See Section 6.5.1 for specific compliance with these requirements.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

12.3.4.1.1 Objectives and Design Basis

The area monitoring system assists in compliance with 10 CFR 50, Appendix A, General Design Criteria 19, 63, and 64.

Monitors are provided throughout the plant to monitor exposure rates and to warn personnel if the measured dose rate exceeds selected levels. Monitors are placed as follows:

(1) In areas where personnel routinely work without continuous health physics surveillance if the area is or could become a radiation area during normal operation.

(2) In a few selected locations in the Auxiliary Building to provide knowledge of any increasing trends in general plant exposure rate levels. These monitors also provide warning of hazardous airborne noble gas concentrations.

(3) In specific areas where exposure rates are normally low but in which high exposure rates could occur under postulated anticipated operational occurrences or accident conditions.

(4) At locations outside the Shield Building at which detected exposure rates can provide a measure of airborne concentrations in the containment under postulated accident conditions.

(5) In the control room to indicate exposure rates during accident conditions.

(6) Portable continuous air monitors, area monitors, and hand-held survey instruments are available to provide surveillance of areas not covered by fixed monitors, or to supplement fixed monitor surveillance.

12.3.4.1.2 Operation Characteristics

Table 12.3-4 lists the physical location (by building elevation and coordinates), and detector range of each area monitor. The specific location may be found on Figures 12.3-4 through 12.3-12 using the coordinates given in Table 12.3-4. Other characteristics of the area radiation monitoring system are given in the following sections.

12.3.4.1.2.1 Area Monitor Detector

The detectors for the Reactor Building upper and lower compartment post accident monitors and upper compartment personnel lock monitor are ion chambers. The other area monitors employ Geiger-Mueller type gamma detectors. Each detector has its own independent high-voltage power supply located on panel 0-M-12 or 1-,2-M-30 in the main control room.

The Reactor Building upper and lower compartment post accident monitors are redundant high range monitors which are required to meet the requirements of RG 1.97 and NUREG-0737.

12.3.4.1.2.2 Main Control Room Ratemeter (0-M-12, 1-,2-M-30)

Ratemeters are of solid-state construction containing a solid-state, high-voltage power supply. Alarms are provided on the ratemeter chassis forhigh radiation and instrument malfunction. Visual and audible alarms are provided for high radiation and instrument malfunction in the main control room.

12.3.4.1.2.3 Local Indicator-Alarm Panel

With the exception of the main control room and Reactor Building upper and lower compartment post accident monitors, each monitor has a locally mounted panel which contains an indicator, a visual and audible high radiation alarm, and a power-on light.

12.3.4.1.2.4 Multipoint Recorders (Main Control Room 0-M-12, 1-,2-M-31)

The area monitors, with the exception of the Reactor Building upper and lower compartment post accident monitors, are recorded on multipoint recorders on panel 0-M-12, which is in the MCR. The upper and lower compartment post accident monitors are input to the Integrated Computer System (ICS) for recording purposes.

12.3.4.1.2.5 Monitor Sensitivity and Range

The ranges of the instrumentation provided are given in Table 12.3-4. The area monitors set points, adjustable over the entire range, are determined by the radiation control group based on operating background levels. The setpoints for the Reactor Building upper and lower compartment post accident monitors are determined by engineering analysis.

12.3.4.1.3 Area Monitor Calibration and Maintenance

With the exception of the Reactor Building upper and lower compartment post accident monitors, periodic testing of each area monitor includes a channel calibration performed at least once per 22.5 months (18 months plus 25%), and a channel operational test is performed periodically as required by the Technical Specifications, the ODCM, or the TVA calibration program procedures. Testing of the Reactor Building upper and lower compartment post accident monitors is described by plant Technical Specifications.

The channel calibration is the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input from a radioactive calibration source. The channel calibration encompasses the entire channel, including the required sensor, alarm, interlock, display and trip functions. The channel calibration may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

The channel operational test (COT) is the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the operability of required alarm, interlock, display and trip functions. The COT includes adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

The built-in checksource function exposes the channel detector to a radioactive source for all channels except those employing an ion chamber detector. The checksource function simulates a detector signal at the channel electronics in channels employing an ion chamber detector. The built-in checksource feature is used to verify functional response of the detector and/or electronics during the performance of the channel operational test. This function is also used by operations personnel at other times, such as after maintenance, to rapidly determine channel operability.

Maintenance is performed, as necessary, if abnormalities are detected during any of the above checks. Unscheduled maintenance will be performed as required.

12.3.4.2 Airborne Particulate Radioactivity Monitoring

12.3.4.2.1 Design Basis

The airborne radioactivity monitoring systems are one of the plant features provided to comply with 10 CFR 50, Appendix A, General Design Criteria 19, 63, and 64, to assure that the requirements of 10 CFR 20 are met.

Each of the systems monitor an air space to which one or more of the following descriptions are applicable.

(1) Spaces in which there is, during normal operation, a potential for airborne concentrations at DAC levels which, when integrated over a normal 40 hr/wk and 50 wk/yr, would exceed the ALI of any isotope or mixture of isotopes and for which there are requirements for either (a) frequent (i.e., once per shift) visits, each of which is for a duration of at least several minutes, or (b) infrequent but routine visits of at least an hour's duration and for which monitoring systems can be practicably supplied in lieu of provision for safely taking and analyzing grab samples for airborne activity prior to personnel entry.

(2) General spaces (e.g., spaces outside shielded equipment rooms) of buildings that contain equipment which bears, in process fluids, potentially significant radioactivity. (Although the plant ventilation systems normally supply clean air upstream of the spaces containing potential leakage points, monitoring is provided to detect airborne activity in the event of malfunction of the ventilation systems).

(3) Spaces which have requirements for routine occupancy into which significant airborne activity may be introduced directly (e.g., physical barriers to its introduction do not exist). If an existing physical barrier consists of a ventilation system, consideration is given to the magnitudes of possible airborne concentrations should the ventilation system malfunction.

(4) Spaces in which habitable conditions must be guaranteed at all times, even during accident conditions.

12.3.4.2.2 Airborne Monitoring Channels

Process and effluent radiation monitoring systems provide useful information about the airborne activity within the plant buildings. These systems, described in Section 11.4, are the following:

(1) Containment Building lower and upper compartment air monitors.

(2) Auxiliary Building ventilation monitor.

Monitoring of the Auxiliary Building airborne activity during normal conditions is accomplished with four portable airborne monitors that monitor the spent fuel pool area, the holdup valve gallery general area, the safety injection pump general area and the shipping bay in the waste packaging area. The decontamination area is monitored by Site RADCON in accordance with site procedures when access to this area is required. Monitoring of the control room, Unit 1 and Unit 2 hot sample rooms, and the waste packaging area is accomplished using portable continuous air monitors (CAMs) having a range of 0.1 to 100 DAC. The portable CAM pumping system provides sample flow through a fixed filter. They are provided with local alarms that annunciated upon instrument malfunction or high radiation.

The Auxiliary Building ventilation monitor real time particulate and iodine channels are used to supplement Auxiliary Building portable airborne monitors listed in Table 12.3-5. These channels detect airborne radioactivity from particulates and iodines in excess of 10 DAC-Hr from any area in the Auxiliary Building which may be normally occupied, taking into account dilution in the ventilation system.

The Containment Building upper and lower compartment monitors (described in Section 11.4) can be used to monitor for airborne activity in the Reactor Building under normal conditions and accident conditions provided containment isolation has not occurred, or conditions allow the lines to be reopened. RADCON surveys will supplement these monitors, as required, to determine if the Containment Building can be entered.

The locations of the four portable airborne monitors that are in the Auxiliary Building can be determined by their respective coordinates from Table 12.3-5 applied to Figures 12.3-4, 12.3-8, 12.3-10, and 12.3-12.

12.3.4.2.3 Operational Characteristics

Paragraphs 12.3.4.2.3 through 12.3.4.2.5 discuss the four portable airborne monitors. For operational characteristics of the Containment Building lower and upper compartment air monitors and the Auxiliary Building ventilation monitors, refer to Section 11.4.

12.3.4.2.4 Component Descriptions

The following component descriptions apply to the four portable airborne monitoring channels:

Local Display Unit

Local alarms are provided on the display unit for high radiation and instrument malfunction. Visual and audible alarms are provided in the main control room on detection of a high radiation or instrument malfunction condition.

Multipoint Recorder (0-M-12)

The air particulate monitor activity signals are recorded on a common multipoint recorder located on panel 0-M-12 in the main control room.

12.3.4.2.5 Sensitivity, Range and Set Point

The four portable particulate monitors located in the Auxiliary Building have a required range of 1.99E-8 to 7.96E-6 microcuries per cubic centimeter. This required range is sufficient to detect 10 DAC hours of airborne radionuclides expected in the area (i.e., Co-60, Cs-137, etc.).

12.3.4.3 Deleted by Amendment 84.

12.3.4.4 Special Radiation Monitors

The types of special radiation monitors are described in the following sections.

12.3.4.4.1 Portal Monitors

The portal monitor is a radiation monitoring device for providing a visual and aural warning when the radiation contamination of an individual exceeds a preset level, especially on the heels and soles of the shoes and a general overall body scan. The portal monitors are located at the gatehouse.

12.3.4.4.2 Personnel Contamination Monitors

The personnel contamination monitor is a radiation monitoring device designed to detect the presence and general location of beta-gamma contamination on the hands, shoes, and clothing of personnel. The instrument is designed to allow an individual to monitor himself.

12.3.4.4.3 Deleted by Amendment 84.

REFERENCES:

(1) SDC, A Shielding - Design Calculation Code for Fuel Handling Facilities (RSIC Code Package CCC-60).

(2) Reactor Shielding Design Manual, Theodore Rockwell III, D. Van Nostrund Company, Incorporated, New York, N. Y., 1956.

(3) ANSI N13.10-1974, Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents.

(4) QAD-P5Z, "Source Shield Detector Problems", ID No. 262361.

(5) MCNP, "A General Monte Carlo N-Particle Transport Code," Version 4B.

Table 12.3-1 Deleted by Amendment 84

Table 12.3-2 Deleted by Amendment 84

Table 12.3-3 Ventilation Air Exhaust Points
(Page 1 of 2)

Location	Volume (ft^3)
REACTOR BUILDING (General Purge)	1,053,800
Vicinity of Reactor Vessel	
Refueling Canal	
Instrument Room[1]	
MAIN CONTROL ROOM (Level of Control Building)	260,000
AUXILIARY BUILDING - GENERAL	3,480,000
Pipe Shafts	
Elevation 737	
Heat Exchanger Rooms	
Hot Instrument Shop	
General Spaces	
Spent Fuel Pit Skimmer Filter Room	
Elevation 713	
Sample Room	
Waste Gas Compressor Rooms	
Decontamination Equipment Room	
Demineralizer and Filter Rooms	
Ion Exchange and Filter Rooms	
Sample Exhaust Hood	
Waste Gas Analyzer Room	
Valve Gallery	
General Spaces	
Boric Acid Filter Rooms	
Demineralizer Ion Exchange and Filter Valve Room	
Elevation 692	
Gas Stripper Rooms	
Hold-Up Tank Rooms	
Waste Evaporator Rooms	
Gas Decay Tank Rooms	
Turbine Driven Auxiliary Feedwater Pump Rooms	
Charging Pump Rooms	
Safety Injection System Pump Rooms	
Refueling Purification Filter Rooms	
Sample Exhaust Hood	
General Areas	
Concentrates Filter Rooms	
Construction Tool Room	

Table 12.3-3 Ventilation Air Exhaust Points
(Page 2 of 2)

Location	Volume (ft^3)
AUXILIARY BUILDING - GENERAL (Continued)	
Elevation 676	
Gas Stripper Feed Pump Room	
Waste Hold-up Tank Room	
Floor Drain Collector Tank Room	
Waste Evaporator Feed Pump Room	
Sump Tank and Pump Room	
Containment Spray Pump Rooms	
Residual Heat Removal Pump Rooms	
Waste Evaporator Feed Filter Rooms	
Auxiliary Waste Evaporator Feed Filter Rooms	
FUEL HANDLING AND RADWASTE PACKAGING AREA	1,012,900
Fuel Handling Area	
Spent Fuel Pool Area	
Waste Package Area	
Fuel Transfer Canal	
Cask Loading Area	
Cask Decontamination Tank Room	
Nitrogen Storage Area	
EGTS Room	
Blowdown Treatment Room	
Spent Resin Tank Room	
Post Accident Sampling Facility	
Cask Decontamination Room	
Condensate Demineralizer Waste Evaporator Building	

Table 12.3-4 Location of Plant Area Radiation Monitors
(Page 1 of 2)

Monitor No.	Location Building and Elevation		Building Coordinates	Area	Range
1-RE-90-1	Auxiliary	El. 757.0	A5-w	Spent Fuel Pool Area	10^{-1} to 10^{4} mR/hr
1-RE-90-2	Auxiliary	El. 757.0	A5-w	Personnel Air Lock	10^{-1} to 10^{4} R/hr
0-RE-90-3	Auxiliary	El. 729.0	A6-y	Waste Packaging Area	10^{-1} to 10^{4} mR/hr
0-RE-90-4	Auxiliary	El. 719.0	A2-q	Equipment Decon Area	10^{-1} to 10^{4} mR/hr
0-RE-90-5	Auxiliary	El. 737.0	A9-v	Spent Fuel Pool Pump Area	10^{-1} to 10^{4} mR/hr
1-RE-90-6	Auxiliary	El. 737.0	A5-s	Comp Clg Ht Exch Area	10^{-1} to 10^{4} mR/hr
1-RE-90-7	Auxiliary	El. 713.0	A5-w	Sample Room	10^{-1} to 10^{4} mR/hr
1-RE-90-8	Auxiliary	El. 713.0	A4-t	Aux FW Pumps Area	10^{-1} to 10^{4} mR/hr
0-RE-90-9	Auxiliary	El. 692.0	A5-w	Waste Evap Cnds Tk Area	10^{-1} to 10^{4} mR/hr
1-RE-90-10	Auxiliary	El. 692.0	A4-t	CVCS Board Area	10^{-1} to 10^{4} mR/hr
0-RE-90-11	Auxiliary	El. 676.0	A7-u	Cntmt Spray & RHR Pump Area	10^{-1} to 10^{4} mR/hr
1-RE-90-59	Reactor	El. 756.63	Az 315°	Cntmt Refueling Floor	10^{-1} to 10^{4} mR/hr
1-RE-90-60	Reactor	El. 756.63	Az 225°	Cntmt Refueling Floor	10^{-1} to 10^{4} mR/hr
1-RE-90-61	Reactor	El. 736.0	Az 88°	Lower Compt Inst Rm.	10^{-1} to 10^{4} mR/hr
0-RE-90-135	Control	El. 757.0	C7-q	Main Cntl Rm Rad Mon	10^{-1} to 10^{4} mR/hr
0-RE-90-230	Turbine	El. 685.0	T8-E	Condensate Demin Area	10^{-1} to 10^{4} mR/hr
0-RE-90-231	Turbine	El. 685.0	T8-E	Condensate Demin Area	10^{-1} to 10^{4} mR/hr
1-RE-90-271	Reactor	El. 806.0	Az 180°	RB Upper Comp Post Accident	10E to 10^{8} R/hr
1-RE-90-272	Reactor	El. 806.0	Az 360°	RB Upper Comp Post Accident	10E to 10^{8} R/hr
1-RE-90-273	Reactor	El. 728.0	Az 170°	RB Lower Comp Post Accident	10E to 10^{8} R/hr
1-RE-90-274	Reactor	El. 728.0	Az 170°	RB Lower Comp Post Accident	10E to 10^{8} R/hr
2-RE-90-1	Auxiliary	El. 757.0	A11-W	Spent Fuel Pool Area	10^{-1} to 10^{4} mR/hr
2-RE-90-2(1)	Auxiliary	El. 757.0	A11-W	Personnel Air Lock	10^{-1} to 10^{4} R/hr
2-RE-90-6	Auxiliary	El. 737.0	A11-s	Comp Clg Ht Exch Area	10^{-1} to 10^{4} mR/hr
2-RE-90-7	Auxiliary	El. 713.0	A11-W	Sample Room	10^{-1} to 10^{4} mR/hr
2-RE-90-8	Auxiliary	El. 713.0	A12-t	Aux FW Pumps Area	10^{-1} to 10^{4} mR/hr
2-RE-90-10	Auxiliary	El. 692.0	A12-t	CVCS Board Area	10^{-1} to 10^{4} mR/hr
2-RE-90-59(1)	Reactor	El. 756.63	Az 315°	Contmt Refueling Floor	10^{-1} to 10^{4} mR/hr
2-RE-90-60(1)	Reactor	El. 756.63	Az 225°	Contmt Refueling Floor	10^{-1} to 10^{4} mR/hr

Table 12.3-4 Location of Plant Area Radiation Monitors
(Page 2 of 2)

Monitor No.	Location Building and Elevation		Building Coordinates	Area	Range
2-RE-90-61(1)	Reactor	El. 736.0	Az 88°	Lower Compt Inst. Rm.	10^{-1} to 10^{4} mR/hr
2-RE-90-271(1)	Reactor	El. 806.0	Az 180°	Upper Cont High Range	10^{0} to 10^{8} R/hr
2-RE-90-272(1)	Reactor	El. 806.0	Az 360°	Upper Cont High Range	10^{0} to 10^{8} R/hr
2-RE-90-273(1)	Reactor	El. 728.0	Az 170°	Lower Cont High Range	10^{0} to 10^{8} R/hr
2-RE-90-274(1)	Reactor	El. 728.0	Az 7°	Lower Cont High Range	10^{0} to 10^{8} R/hr
0-RE-90-102	Auxiliary	El. 757.17	A11-x	Spent Fuel Pit Area	10^{-1} to 10^{4} mR/hr
0-RE-90-103	Auxiliary	El. 757.17	A5-w	Spent Fuel Pit Area	10^{-1} to 10^{4} mR/hr

(1) Not required for Unit 1 operation.

Table 12.3-5 Airborne Particulate Activity Monitoring Channels

Monitor No.	Location Building and Elevation	Building Coordinates	Area	Range
Spent Fuel Pool Area	Aux. Building / EL. 757.0	A8-y	Spent Fuel Pool Area	10^1 to 10^7 cpm
Holdup Valve Gallery Area	Aux Building / EL. 713.0	A6-s	Holdup Valve Gallery General Spaces	10^1 to 10^7 cpm
Safety Injection Pump Area	Aux Building / EL. 692.0	A9-u	Safety Injection Pump General Spaces	10^1 to 10^7 cpm
Shipping Bay Area	Aux Building / EL. 729.0	A9-y	Shipping Bay	10^1 to 10^7 cpm

Table 12.3-6 Shield Wall Thickness
(Page 1 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
Elev. 674' 0"					
A1	Concrete Fill	2' 0"	2' 3"	3'9" / 5'0"	41N366-1 41N328-1
A2	Concrete Fill	2' 0"	5'0"	2' 3"	41N328-1 41N366-1
Elev. 676' 0"					
A2	3' 0"	5' 0"	3' 0"	5' 0"	41N306-1 41N473-1 41N470-1
A3	3' 0"	5' 0"	3' 0"	3' 0"	41N306-1 41N473-1 41N470-1
A4	3' 0"	2' 0"	2' 6"	3' 0"	41N306-1 41N309-1
A4a	2' 0"	5' 0"	2' 6"	3' 0"	41N306-1 41N309-1 4IN470-1
A5	1'6"	3' 0"	1'6"	1'6"	41N309-1 41N366-1
A10	2' 0"	2' 0"	2' 3"	3' 9"	41N366-1 41N306-1

Table 12.3-6 Shield Wall Thickness
(Page 2 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
A11	2' 0"	2' 0"	2' 3"	3' 9"	41N366-1 41N309-1
A12	2' 0"	2' 0"	3' 9"	2' 3"	41N366-1 41N309-1
A13	2' 0"	2' 0"	3' 9"	2' 3"	41N366-1 41N309-1
Elev. 692' 0"					
A3	3' 6"	4' 0"	3' 6"	3' 0"	41N330-1 41N310-1 41N368-1 41N330-2
A4	1' 0"	1' 0"	1’0"	3’0"	41N368-1 41N310-1
A5	3' 6"	4' 0"	3' 0"	3' 6"	41N310-1 41N368-1
A9	3' 0"	2' 0"	2’6"	2' 0"	41N368-1 41N310-1
A10	3' 0"	2' 0"	2' 0"	2’0"	41N368-1 41N330-1
A11	2' 2"	2' 0"	2' 0"	2' 0"	41N368-1
A15	2' 3"	4' 0"	2' 6"	4' 0"	41N328-2 41N368-2 41N337-1
Al6	4' 0"	2' 3"	2' 6"	4' 0"	41N328-2 41N368-2
A18	2' 0"	1' 6"	2’6"	2' 0"	41N368-1

Table 12.3-6 Shield Wall Thickness
(Page 3 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
A21	2' 0" / 3'0”	2' 0"	2' 0"	2' 0"	41N368-1 41N328-1
A22	3' 0"	2' 0"	2'0"	2' 0"	41N368-1 41N310-1
A23	3' 0"	2' 0"	2' 0"	2'0"	41N368-1 41N310-1
A29	2' 3"	3' 6"	2' 3"	2' 3"	41N368-1 41N310-1 41N470-1
A30	2' 3"	3' 6"	2' 3"	3' 0"	41N368-1 41N330-1 41N470-1
A31	3' 0"	3' 6"	3' 0"	3' 0"	41N368-1 41N330-1
Elev. 713' 0"					
A6	3' 0"	4'8" / 3' 0"	2' 3" / 1'6"	4'0"	41N370-1 41N315-2 41N344-3
A7	4' 0"	4' 0"	2' 3"	4' 0"	41N315-2 41N370-1
A9	4' 0"	2' 0"	4' 0" / 3' 0"	2' 0"	41N370-1
A10	2' 3"	2' 6"	2' 3"	2' 3"	41N370-1
A11	2' 3"	2' 3"	2' 3"	2' 6"	41N370-1
A12	2' 3"	2' 3"	2' 3"	2' 6"	41N370-1
A13	2' 3"	2' 3"	0' 6" / 1' 0"	2' 6"	41N370-1 41N373-1

Table 12.3-6 Shield Wall Thickness
(Page 4 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
A14	2' 3"	2' 3"	2' 6"	0' 6" / 1' 0"	41N370-1 41N373-1
A15	2' 3"	2' 3"	2' 6"	2' 3"	41N370-1
A16	2' 3"	2' 3"	2' 6"	2' 3"	41N370-1
A17	2' 3"	2' 6"	2' 3"	2' 3"	41N370-1
A18	4' 0"	2' 0"	2' 0"	4' 0" / 3' 0"	41N370-1
A19	3' 0"	4' 0" / 3' 0"	1' 6"	2' 3" / 4' 0"	41N370-1 41N315-1 41N344-3
A20	4' 0"	4' 0"	4' 0"	2' 3"	41N370-1 41N315-1
A22	1' 9"	1' 9"	1' 9"	1' 9"	41N370-1
A23	1' 6" / 1' 0"	2' 6" / 1' 0"	1' 6" / 1' 0"	1' 0"	41N370-1
A24	1' 9"	2' 6"	2' 1"	2' 6"	41N370-1
A25	2' 6"	3' 0"	2' 6"	2' 6"	41N370-1 41N315-2
A26	2' 6"	3' 0"	2' 6"	2' 6"	41N370-1 41N315-2
A28	6' 0"	2' 3"	2' 3" / 2' 6"	3' 0"	41N358-1 41N315-2
A29	6' 0"	2' 3"	3' 0"	2' 3" / 2' 6"	41N358-1 41N315-2
Elev. 713' 0" DEMIN PITS					
P1	1' 0"	3' 0"	1' 0"	2' 6"	41N370-1
P2	2' 6"	3' 0"	2' 6"	2' 6"	41N370-1

Table 12.3-6 Shield Wall Thickness
(Page 5 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
P3	2'6"	3' 0"	2' 6"	2' 6"	41N370-1
P4	2' 6"	3' 0"	2' 6"	2' 6"	41N370-1
P5	2' 6"	3' 0"	2' 6"	2' 6"	41N370-1
P6	2' 6"	3' 0"	2' 6"	2' 0"	41N370-1
P7	1' 0"	3' 0"	2' 0"	1' 0"	41N370-1
P8	1' 0"	3' 0"	1' 0"	1' 0"	41N370-1
P9	4' 0"	3' 3"	3' 0"	3' 0"	41N370-1
P10	4' 0"	2' 6"	3' 0"	2' 6"	41N370-1
P11	4' 0"	2' 6"	2' 6"	2' 6"	41N370-1
P12	4' 0"	2' 6"	2' 6"	2' 6"	41N370-1
P13	4' 0"	4' 0"	4' 0"	4' 0"	41N370-1
P14	4' 0"	4' 0"	4' 0"	4' 0"	41N370-1
P15	3' 3"	4' 0"	4' 0"	3' 3"	41N370-1
P16	4' 0"	2' 6"	2' 6"	2' 6"	41N370-1
P17	4' 0"	2' 6"	2' 6"	2' 6"	41N370-1
P18	4' 0"	2' 6"	2' 6"	3' 0"	41N370-1
P19	4' 0"	3' 3"	3' 0"	3' 0"	41N370-1
P20	3' 3"	4' 0"	3' 3"	4' 0"	41N370-1
P21	4' 0"	4' 0"	4' 0"	4' 0"	41N370-1
P22	4' 0"	4' 0"	4' 0"	4' 0"	41N370-1

Table 12.3-6 Shield Wall Thickness
(Page 6 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
ELEV. 713' 0" BATS FILTER					
	1' 6"	4' 0"	1' 6"	1' 6"	41N370-6
ELEV. 729' 0"					
A3	2' 9"	2' 0" / 1' 6"	2' 9"	2' 9"	41N388-1
A4	2' 6"	5'0" / 4'0"	2' 9"	2' 6"	41N388-1
A5	5'0" /4'0"	1' 6"	2' 9"	2' 9"	41N337-1 41N388-1 41N318-1
ELEV. 737' 0"					
A5	3' 0"	3' 0"	2' 3" / 2' 6"	3' 0"	41N318-2
A7	2' 3"	3' 0"	2' 3"	2' 6"	41N372-1
A8	2' 3"	3' 0"	2' 6"	2' 3" / 2' 6"	41N372-1
A9	3' 0"	3' 0"	3' 0"	2' 6"	41N318-1
A15	1' 6"	1' 6"	1' 6"	1' 6"	41N372-2
A16	1' 6"	1' 6"	1" 6"	1' 6"	41N372-2
ELEV. 730' 6"					
DE3	2' 0"	2' 0"	2' 0"	2' 0"	41N391-7
DE4	2' 0"	2' 0"	2' 0"	2' 0"	41N391-7
DE5	2' 0"	2' 0"	2' 0"	2' 0"	41N391-7
DE6	2' 0"	2' 0"	2' 0"	2' 0"	41N391-7
ELEV. 750' 6"					

Table 12.3-6 Shield Wall Thickness
(Page 7 of 7)

Room Number and Elevation	North Wall Thickness	South Wall Thickness	East Wall Thickness	West Wall Thickness	Ref. Drawing
DE2	2' 0"	2' 0"	2' 0'	2' 0"	41N391-7
DE3	2' 0"	2' 0"	2' 0"	2' 0"	41N391-7
DE4	2' 0"	2' 0"	2' 0"	2' 0"	41N391-7
ELEV. 685' 0" UNIT 1 TURB. BLD. CON DEMIN	1' 3"	1' 3"	1' 3"	8' 6"	41N233-3 41N238-2
ELEV. 685' 0" UNIT 2 TURB. BLD. CON DEMIN	1' 3"	1' 3"	8' 6"	1' 3"	41N238-2 41N233-3
ELEV. 685' 0" UNIT 1/2 TURB. BLD. CON DEMIN NEUT TANK	1' 3"	1' 3"	1' 3"	1' 3"	41N238-3

Table 12.3-7 Special Radiation Monitors

Monitor	Area
Portal Monitors	Gate Houses
Personnel Contamination Monitors	Aux. Building Entrance

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Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-1 Plans - Roof - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-2 Plan El. 772.0 and Above - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-3 Powerhouse, Auxiliary, Reactor and Control Buildings - Plan El. 772.0, 782.0 and 786.0 - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-4 Powerhouse, Auxiliary, Reactor and Control Building - Plan El. 755.0 and 757.0 - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-5 Powerhouse, Units 1 and 2 - Plan El. 757.0 and 755.0 - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-6 Powerhouse, Units 1 and 2 - Plan El. 737.0 and 729.0 - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-7 Auxiliary Building, Units 1 and 2 - Radiation Zone Map, Condensate Demineralizer

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-8 Powerhouse, Auxiliary, Reactor and Control Buildings - Radiation Zone Map - Plan El. 729.0 and 737.0

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-9 Powerhouse, Units 1 and 2 - Plan El. 713.0 and 708.0

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-10 Powerhouse, Auxiliary, Reactor and Control Buildings - Plan El. 708.0 and 713.0 - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-11 Powerhouse, Units 1 and 2 - Plan El. 685.5

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-12 Powerhouse, Auxiliary, Reactor and Control Buildings - Plan El. 676.0 and 692.0 - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-13 Powerhouse, Units 1 and 2 - Turbine Building - Plan El. 670.0 Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-14 Powerhouse, Units 1 and 2 - Reactor Building - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-15 Powerhouse, Units 1 and 2 - Reactor Building - Radiation Zone Map

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-16 Powerhouse, Auxiliary Building, Mechanical, Heating, Ventilating and Air Conditioning

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-17 Fuel Transfer Tube Shielding Details

Figure 12.3-18 Deleted by Amendment 97

Security-Related Information - Withheld Under 10CFR2.390

Figure 12.3-19 Deleted by Amendment 97

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12.4 DOSE ASSESSMENT

Annual Exposures

The anticipated maximum annual external exposure to any employee is expected to be far below the NRC limits set out in 10 CFR 20. Yearly collective radiation exposure (CRE) estimates to plant workers are made from data collected from historical plant operating experiences and records of dose received from routine operations and surveillances, minor maintenance and inspections, and refueling outage activities. Historical records indicate that the annual on-line and refueling outage exposures can be estimated at:

On-line CRE < 10 person-rem/yr

Refueling Outage CRE < 120 person rem/outage (average 80 person-rem/yr based on 18-month fuel cycle)

A requirement imposed upon all features of the WBN design, maintenance and operation, including anticipated operational occurrences, is that radiation doses to plant personnel be maintained as low as reasonably achievable (ALARA). In satisfying this requirement, the shielding design of the WBN incorporates (1) effective locations for many shield wall penetrations, (2) operation of valves from behind shield walls, (3) provisions to drain equipment from behind shield walls prior to maintenance, and (4) arrangements for shielding of spent filters during the removal process.

Operating and maintenance procedures are designed to ensure that the plant staff receives the minimum radiation dose allowed by the plant design.

The combination of shielding design and operation and maintenance of the plant in accordance with written procedures gives WBN a lower person-rem total. Design and operating features and practices should minimize the need for staff augmentation.

Estimate of Personnel Internal Exposure

The anticipated maximum internal exposure to any employee is expected to be far below the NRC limits set out in 10 CFR 20. Minimizing internal exposures is accomplished by controlling ventilation air flow throughout the buildings such that the supplied clean air is exhausted from clean areas to the more contaminated atmospheres and by controlling personnel exposure time in affected areas..

Monitoring internal exposures is performed in accordance with 10 CFR 20. For work in contaminated atmospheres, Radiation Protection will evaluate the need for respiratory protection equipment using the guidance of an established Total Effective Dose Equivalent (TEDE) ALARA policy (further discussion of this evaluation is in Section 12.5.2).

Boundary of the Restricted Area

Estimated annual dose rate in mrem/yr at the boundary of the restricted area (located inside the exclusion area defined in 2.1.2) are given in Tables 12.4-1 and 12.4-2. The total dose rates are the sum of the adult inhalation dose rate plus the gamma dose rate from the gaseous plume plus ground contamination dose rate plus the gamma dose rate from outdoor storage tanks shown in Figure 12.4-1. The gaseous effluents are releases from the Auxiliary Building, the Containment Building, and the Turbine Building. Dose rate contributions from shielded sources inside the plant building are negligible. The model used for estimating the plume dose rates is described in the Offsite Dose Calculation Manual for WBN. The dose rates from the two makeup water storage tanks and the two refueling water storage tanks were estimated with the computer code QAD-P5Z. As indicated in Table 12.4-2, the highest total dose rate at the boundary of the restricted area is 105 mrem/yr, based on a continuous 2000 hours/yr occupancy. Use of a more realistic occupancy, reflective of the transient traffic expected for this location, would result in a much lower dose estimate.It is, therefore, considered highly unlikely that a member of the public would receive $\geq$ 100 mrem/yr at or beyond the restricted area boundary.

Inside Restricted Area

Estimated dose rates from the gaseous plume and tanks at outside locations have also been calculated at locations within the restricted area. Dose points inside the restricted area have been selected as being possible points at which the maximum dose rate might occur and locations where unmonitored workers might occupy. Dose rate contributions, from shielded sources inside the plant buildings are negligible. Table 12.4-2 lists each point of interest. The dose rates listed are for two units in operation. Figure 12.4-1 shows the locations of the dose points inside the restricted area. Access to areas inside the restricted area that require personnel monitoring in accordance with 10 CFR 20 are posted and controlled in accordance with Radiation Protection procedures.

REFERENCES

None.

Table 12.4-1 Restricted Area Boundary Dose Rates For 2000 Hr/Year Exposure

Compass Sector	Distance (meters)	Inhalation Dose Rate (mrem/yr)	Plume Dose Rate (mrem/yr)	Ground Exposure Dose Rate (mrem/yr)	Total Effluent Dose Rate (mrem/yr)
N	169	4.2E-01	3.1E+01	1.0E+00	32.9
NNE	174	7.2E-01	4.2E+01	2.0E+00	44.6
NE	203	4.6E-01	2.5E+01	6.9E-01	26.3
ENE	240	4.4E-01	2.3E+01	4.7E-01	24.0
E	234	5.0E-01	2.6E+01	5.4E-01	27.5
ESE	240	4.5E-01	2.4E+01	4.9E-01	24.7
SE	282	4.7E-01	2.4E+01	5.2E-01	25.0
SSE	436	1.4E-01	6.5E+00	2.2E-01	6.9
S	431	8.9E-02	4.5E+00	2.3E-01	4.8
SSW	364	1.2E-01	6.1E+00	3.7E-01	6.6
SW	217	3.0E-01	1.7E+01	5.7E-01	17.4
WSW	217	4.7E-01	2.5E+01	5.9E-01	26.3
W	167	3.0E-01	1.7E+01	5.7E-01	17.4
WNW	168	1.6E-01	9.1E+00	1.7E-01	9.4
NW	195	2.0E-01	1.1E+01	2.2E-01	11.4
NNW	173	3.2E-01	1.8E+01	4.1E-01	18.3

Table 12.4-2 Onsite Dose Rates (Mrem/Yr)

Location	Gaseous Effluents	U1 PWST	U2 PWST	U1 RWST	U2 RWST	CSTs	Total
RA 1	1.74E+01	2.66E-01	-	0.88E+02	-	-	1.05E+02
RA 2	3.29E+01	4.06E-02	4.06E-02	9.41E+00	9.41E+00	-	5.18E+01
RA 3	2.75E+01	-	8.90E-02	-	2.93E+01	2.66E-02	5.69E+01
P 1	0.99E+02	1.77E-01	1.77E-01	4.67E+01	4.67E+01	-	1.93E+02
P 2	1.27E+02	-	8.28E+00	-	5.19E+02	3.30E-02	6.54E+02
P 3	1.57E+02	-	3.36E+01	-	8.81E+03	7.16E-02	9.00E+03
P 4	1.10E+02	-	3.66E-01	-	3.19E+02	6.40E-01	4.30E+02
P 5	2.28E+01	-	1.79E-02	-	7.99E+00	7.16E-02	3.09E+01
P 6	2.41E+01	7.20E-02	-	3.67E+01	-	-	6.09E+01
P 7	1.83E+01	4.86E-02	-	2.37E+01	-	-	4.20E+01
P 8	0.81E+02	-	-	8.81E+03	-	-	8.89E+03
SB	2.81E+00	9.26E-06	9.29E-06	2.67E-03	2.67E-03	9.33E-06	2.82E+00

Location Descriptions

RA 1 - West restricted area boundary
RA 2 - North restricted area boundary
RA 3 - East restricted area boundary
P 1 - 80' North of waste package area "A"
P 2 - 45' NE of U2 primary water storage tank (PWST)
P 3 - Between U2 refueling water storage tank (RWST) and PWST
P 4 - 90' NE of U2 condensate storage tank (CST)
P 5 - North end of switchyard
P 6 - West corner of main office building
P 7 - SE corner of main parking lot
P 8 - Between U1 RWST and PWST
SB - Land site boundary (1250 meters SE)

Figure 12.4-1 Dose Points Inside Restricted Area

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12.5 RADIATION PROTECTION PROGRAM

12.5.1 Organization

The Radiation Protection program consists of four elements that are directed toward essential support to TVA's nuclear power program.

(1) Radiological impact assessments.

(2) Radiation protection planning and radiological safety evaluation, including preliminary safety analysis reports, final safety analysis reports, and radiological emergency plans.

(3) Radiological environmental monitoring.

(4) Radiological control activities

The Radiation Protection Section is under the supervision of the Plant Manager.

The Radiation Protection Section is responsible for the radiation protection activities at the plant. It applies radiation standards and procedures; reviews proposed methods of plant operation; participates in development of plant documents; and assists in the plant training program, providing specialized training in radiation protection. It provides coverage for all operations involving radiation or radioactive materials including maintenance, fuel handling, waste disposal, and decontamination. It is responsible for personnel and inplant radiation monitoring, and maintains continuing records of personnel exposures, plant radiation, and contamination levels.

The Radiation Protection Manager (RPM) is the onsite supervisor of the Radiation Protection Section and is responsible for implementation, development, and direction of an adequate program of radiological health surveillance for all plant operations involving potential radiation hazards. He keeps the plant manager informed at all times of radiation hazards and conditions related to potential exposure, contamination of plant and equipment, or contamination of site and environs. His duties include training and supervising Radiation Protection technicians; planning and scheduling monitoring and surveillance services, scheduling technicians to ensure around-the-clock shift coverage as required; maintaining current data files on radiation and contamination levels, personnel exposures, and work restrictions; and ensuring that operations are carried out within the provisions of developed radiological control standards and procedures. He critiques plant operations and reviews suggestions from employees to identify areas in which exposures can be reduced. As an alternate member of the Plant Operations Review Committee, he reviews and consents on operating procedures. He provides assistance and advice to the Site Emergency Director during radiological emergencies.

As a minimum, the guidance of Regulatory Guides 8.2, 8.8, 8.10, and 1.8 has been followed in developing the Radiation Protection program and the personnel qualifications.

The minimum qualification requirements for the Radiation Protection Manager are stated in Section 13.1.3.

The minimum requirements for Radiation Protection technicians responsible for a shift are appropriate technical training and two years of applied health physics experience dealing with radiological problems similar to those encountered in a nuclear power station. Applicable experience may be granted as equivalency for the technical training.

Further information on the training and qualifications of Radiation Protection personnel may be found in Chapter 13.

12.5.2 Equipment, Instrumentation, and Facilities

The Radiation Protection facilities consist of office space; short-term record storage; and a service center.

The dosimetry laboratory is located outside of the restricted area within the plant training center. This facility provides radiological services for the in-processing and out-processing of personnel, issuance of dosimetry and coordination of bioassay services. The dosimetry laboratory is equipped with bioassay services. The dosimetry laboratory is equipped with bioassay equipment (whole body counters) and respirator fit testing equipment.

The service center is located between the Auxiliary Building and service shop areas. The Radiation Protection technicians use the service center as their base of operations and communications for work and activities performed in radiologically controlled areas. The service center is also used as the normal point of access and egress controll to and from radiologically controlled areas. Remote access control points may also be established during periods of high maintenance such as refueling outages or as necessary at remote locatons when additional access/egress control measure are warranted.

The service center is equipped with instrumentation, supplies, cabinets, and storage space. Portable and laboratory radiation monitoring instruments, and other Radiation Protection supplies including signs, personnel decontamination supplies, air sampling equipment, etc., are kept in this area.

Adjacent to the Radiation Protection service center is a personnel decontamination room equipped with a shower, sink and appropriate personnel decontamination supplies. Service center drains are piped to the Liquid Radwaste System for processing. Radiation monitoring instruments for detection of very low levels of radioactive contamination are readily available.

The portable and laboratory equipment located in the service center will allow the Radiation Protection personnel to measure dose rates and contamination levels throughout the plant in all routine and emergency situations. The portable Radiation Protection survey instrumentation and the fixed Radiation Protection and chemistry laboratory counting equipment are equivalent to the instrumentation described in Regulatory Guide 8.8, Position C.4.

Each portable survey instrument is calibrated and checked periodically with standard radioactive sources in accordance with instrument specific calibration and maintenance procedures. Accurate records on the performance of each instrument during each calibration are maintained. Each laboratory counting system is checked at regular intervals with standard radioactive sources for proper counting efficiencies, background count rates, and operating parameters.

TVA provides appropriate protective clothing dress out areas and protective clothing for use in radiological areas. Clothing required for a particular instance is prescribed by Radiation Protection based upon the actual or potential radiological conditions. Protective clothing is cleaned, surveyed for contamination, checked for physical condition, and returned to service if acceptable. Additional protective clothing stock is available from the plant warehouse as required. Protective clothing available for use includes:

(1) Coveralls

(2) Lab coats

(3) Gloves

(4) Head covers

(5) Foot covers

Tape or equivalent may be provided so that openings in clothing can be sealed, if necessary.

Respiratory protection devices are available from the Radiation Protection service center. The Radiation Protection is responsible for the maintenance of the devices, although other groups may perform the actual work. The need for, and type of, respiratory protection equipment to be issued for specific tasks/activities is determined by Radiation Protection evaluations. Maintaining TEDE ALARA and minimizing of the total risk from all expected hazards is the goal of the evaluations. Considerations made in the performance of these evaluations include:

(1) Process controls (e.g.,system flushing, venting, isolation)

(2) Engineering controls (e.g., containment devices, ventilation, remote handling tools)

(3) Radiological hazards

(4) Industrial Safety hazards

(5) Effects of respirators on worker efficiency and total dose

(6) Environmental conditions

(7) Need for precise communications and/or visual perception

(8) Physiological and/or psychological affects of respirators

(9) Job duration (e.g., access controls, stay times)

(10) Worker acceptance and input

Available respiratory protection devices include:

(1) Full face mask with high efficiency filters

(2) Full face mask with constant or pressure demand air flow. A manifold unit is used that contains mist filters, a regulator, and relief valve.

(3) Powered air purifying respirators (PAPRs) with high efficiency filters

(4) Constant air flow hoods and/or suits

(5) Self contained pressure demand breathing apparatus (bottled air type).

12.5.3 Procedures

Routine radiological surveys to detect radiation, radioactive contamination, and airborne radioactivity are performed throughout the plant on periodic schedules. Survey frequencies are determined by the Radiation Protection Manager based upon the actual or potential radiological conditions. Schedules for completion of routine surveys are issued to the technicians. As plant conditions change, the schedule will be updated. Radiological surveys may be performed whenever personnel enter potential or actual radiological areas and there is any doubt as to the existing conditions. Retention of survey records follows the requirements of 10 CFR 20.2103.

Section 12.1.1 defines the TVA overall ALARA program. Inplant procedures involving radiological conditions are written such that keeping exposures ALARA is a major consideration.

Entry into Radiation Areas as defined by 10 CFR 20.1003 is administratively controlled. Radiation Areas are posted per 10 CFR 20.1902. Entry to these areas requires the issuance of a Radiation Work Permit (RWP). The RWP sets out entry requirements and other precautions. In addition, any entry into Radiation Areas requires possession of an appropriate dosimeter.

Access controls to prevent unplanned exposures in high radiation areas are implemented in accordance with Technical Specifications 5.11, High Radiation Area and applicable guidance of Regulatory Guide 8.38. High Radiation Areas are posted per 10 CFR 20.1902. When does rates in High Radiation Areas are greater than 1.0 rem/hr at 30 centimeters from radiation source or from any surface penetrated by the radiaiton, but less than 500 rads/hr at 1 meter from the radiation source or from any surface penetrated by the radiation, the area is posted as Locked High Radiation Area to emphasize the radiological significance of the area. Entry to High Radiation Areas requires issuance of a Radiation Work Permit (RWP). The RWP sets out entry requirements and other precautions. Entry into High Radiation Areas requires possession of an appropriate dosimeter.

In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in 1 hour or 1 meter from a radiation source or any surface through which the radiation penetrates:

(1) Conspicuously posted with a sign(s) stating

GRAVE DANGER - VERY HIGH RADIATION AREA

(2) Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the Radiation Protection Manager.

(3) Plant manager's (or designee) approval required for entry.

(4) Radiation Protection personnel shall be in accompaniment of the person(s) making the entry. Radiation Protection shall assess the radiation exposure conditions at the time of the entry.

Areas where transferable radioactive contamination is present in levels greater than 1000 dpm/100 cm^2 beta-gamma or 20 dpm/100 cm^2 alpha are posted as "Contaminated Areas." Entry into a Contaminated Area requires a RWP which specifies protective clothing and measures dependent upon the conditions. Whenever practical, contaminated equipment will be packaged to prevent contamination spread and tagged with radioactive material tags when removed from a Contaminated Area. All materials and equipment leaving Contaminated Areas will be monitored and released only if there is no contamination present in excess of established limits. All items which have been in a radiologically controlled area and which have the potential for becoming contaminated are monitored prior to being released from the area.

Potential airborne radioactivity concentrations are kept to a minimum by process and engineering controls. Airborne radioactivity conditions are evaluated by using strategically located continuous air monitors, as well as routine and special grab sampling. Entry into an Airborne Radioactivity Area as defined by 10 CFR 20.1003 is administratively controlled. Airborne Radioactivity Areas are posted per 10 CFR 20.1902. Entry to these areas requires the issuance of an RWP that sets out entry requirements and other precautions.

Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," is used as guidance in implementing a bioassay program.

Planned Special Exposures (PSE) may be authorized. In the event WBN uses a PSE, the PSE will be conducted in accordance with guidance from Regulatory Guide 8.35, "Planned Special Exposures".

Occupational exposure limits for minors, declared pregnant women, and for radiation dose to the embryo/fetus are established following the guidance of Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", and Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus".

All personnel entering the Radiologically Controlled Area (RCA) unescorted will have completed a radiological orientation course. This course consists of introductory subjects, monitoring techniques and equipment, protective procedures and equipment, and the radiological emergency plan. The presentation methods, length of the particular course, material emphasized, and participation in demonstrations are based upon the needs of the individual.

The storage and handling of byproduct materials and special nuclear materials is detailed in procedures.

Prospective monitoring determinations for internal and external dose monitoring are performed for individuals or group of individuals ectering the restricted area. Personnel monitoring, for dose from sources external to the body, is conducted using appropriate dosimeters as required by 10CFR20. TVA maintains accreditation as a processing laboratory for dosimeters, as described in American Standards Institute (ANSI) N13.11–1983, “Personnel Dosimeter - Criteria for Performance”. This accreditation is under the National Voluntary Laboratory Accreditation Program conducted by the National Institute of Standards and Technology. Dosimeters may be processed onsite by WBN, an accredited TVA subfacility, or by another processing laboratory within the scope of TVA’s accreditation. Dose information for whole body (total effective dose equivalent), external exposure of the skin, lens of the eye, and extremities is recorded in a dose tracking system and retained in a permanent historical database for generating required reports. Real time control is generally implemented using information from direct reading docimeters. Official doses of record are taken from dosimeters. However, doses are calculated when dosimeter results are not available or do not accurately represent actual dose received.

Personnel monitoring and confirmatory monitoring for dose from intakes of radioactive materical is conducted using DAC-HR tracking and bioassays, including whole body counting. Monitoring is performed for each person requied to be monitored by 10CFR20. The whole body counter is calibrated with standard radiosotopes in configurations that approximate the human body. It is able to detect expected gamma emitting readonuclides per ANSI-N13.30, September 1989, Table-1, “Acceptable Minimum Detectable Activities.”

REFERENCES

None.

Appendix 12A

Radiation Protection Features for the Tritium Production Program

Beginning with Cycle 5, WBN will irradiate Tritium Producing Burnable Absorber Rods (TPBARs) in Unit 1 only to fulfill an Interagency Agreement with the U.S. Department of Energy (DOE).

TVA has performed an evaluation of the radiation sources for the tritium production program and determined that the core source term for the maximum irradiation level of 2,304 TPBARs is bounded by the existing source term of record for WBN. Additionally, the transition cycles to arrive at the maximum number of TPBARs have also been analyzed and are bounded by the existing source term of record.

Tritium is a radioactive isotope of hydrogen with a half-life of 12.3 years, which undergoes beta decay, with a maximum energy level of 18.6 KeV. The average energy level is 5.7 KeV. This low energy limits the maximum range of a tritium beta to about 6 millimeters in air and 0.0042 millimeters in soft tissue. Therefore, the primary radiological significance of exposure to tritium is in the form of internal exposure and the only potential hazard comes when personnel are exposed to open processes that have been wetted with tritiated liquids. Therefore, the design features of the plant that deal with contamination and airborne radioactive control, such as drain and ventilation systems are of potential concern. TVA's evaluation of these systems concluded that there is negligible impact to these systems by the tritium production program.

TVA has evaluated the current Operational Radiation Protection Program described in the preceding sections and determined that there will be no major impact due to the tritium production program. The program modifications are adjustments or changes in scope, rather than major program revisions. Additional monitoring instrumentation and sample equipment to allow better assessment of plant airborne tritium activity will be procured. Plant specific procedures addressing these actions will be developed before TPBAR irradiation. Tritium program enhancements include:

Tritium Bioassay Program - the tritium bioassay program will follow the guidance of U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 8.9 "Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program."

Tritium Monitoring - tritium monitoring will consist of Air Monitoring, Air Sampling, Surface Monitoring, Liquid Monitoring and Liquid Scintillation Counting.

Radiological Environmental Monitoring Program - changes to the REMP to accommodate tritium production have been identified. These changes will consist of:

- Atmospheric moisture - selected atmospheric sampling stations will be modified to include the collection of atmospheric moisture.
- Surface water - perform tritium analysis on samples collected every four weeks from the downstream and upstream sampling locations.

- Public water - perform tritium analysis on samples collected every four weeks from downsteam public water systems.
- Ground water - perform tritium analysis on samples collected every four weeks from the site monitoring wells. Add monthly grab sampling at locations for the nearest (within 5 mile radius) offsite users of ground water as the source of drinking water.

With the program changes listed above, the WBN Radiation Protection Program will continue to provide assurance of the health and safety of plant employees and the public during the tritium production program.