12.2 RADIATION SOURCES

12.2.1 CONTAINED SOURCES

The radiation sources used for the design and analysis of the shielding requirements are based on the design of plant operation including full power operation, shutdown conditions, refueling operations, and for various postulated accidents. They include the neutron and gamma fluxes outside the reactor vessel, the reactor coolant activation, fission and corrosion product activities, deposited corrosion product sources on reactor coolant equipment surfaces, spent fuel handling sources, and postulated core meltdown sources. In addition, radiation sources for various auxiliary systems are also tabulated.

12.2.1.1 Reactor Coolant Fission and Corrosion Product Activity

The activity utilized for shielding calculations is shown in Table 11.1-2.

The basis for the above reactor coolant radioisotope concentrations is specified in Table 11.1-1, and the manner in which the concentrations have been derived is given in Subsection 11.1.1.1.

12.2.1.2 Neutron Fluxes Outside the Reactor Pressure-Vessel

The maximum neutron spectra during full power operation, divided into 10 energy groups, for points on the top and bottom of the reactor pressure vessel along the core centerline, and a point on the side of the vessel adjacent to the maximum axial power, are shown in Table 12.2-1. The neutron spectra in this table include the neutrons scattered from the concrete cavity wall and are used to determine the thickness of the primary shield wall.

To determine the neutron flux streaming up the annulus between the reactor vessel and the reactor vessel cavity wall, a better definition of the neutron spatial and angular fluxes emergent from the vessel is developed by the computer program DOT[(1)].

The calculation utilizes S_4 quadrature, 22 neutron energy groups, and 16 axial intervals from the core bottom to the upper guide structure. A vacuum boundary is used at the outer radius.

The DOT generated data, summarized for convenience in the form of neutron leakage rates from axial intervals spaced along the vessel surface, is shown in Table 12.2-2. The equivalent scalar fluxes can be obtained by division of the leakage rates in the Table 12.2-2 by the surface area of the region.

12.2.1.3 Gamma Fluxes at Full Power Operation

The maximum gamma spectra during full power operation, divided into 14 energy groups, for points on the top and bottom of the vessel along the core centerline, and a point for the side of the vessel adjacent to the maximum axial power, are shown in Table 12.2-3.

12.2.1.4 Reactor Coolant N-16 Activity

→(DRN 03-2066, R14)

The N-16 activity in the reactor coolant which determines the shielding requirements for the secondary shield wall and portions of the Chemical and Volume Control System (CVCS) is discussed in Subsection 11.1.3.1. The N-16 activity at various points in the primary system, as well as the CVCS, is determined from the given activity at the reactor outlet nozzle, taking into account the decay due to the transit time between the nozzle and the point of interest as calculated on the basis of the maximum flow rate and changes in density caused by cooling.

←(DRN 03-2066, R14)

12.2.1.5 Reactor Coolant System Sources at Shutdown

Following shutdown, residual radiation from the Reactor Coolant System is due to fission product decay gamma radiation emanating from the core; material activation sources, radioactive corrosion products which have deposited on surfaces, and the fission and corrosion products in the reactor water. The reactor vessel maximum decay gamma spectra, divided into eight energy groups for various times after shutdown, are shown in Table 12.2-4 for a point on the side of the vessel adjacent to the maximum axial peak.

The material activation spectra at the same location and decay time is shown in Table 12.2-5. The material activation sources include contributions from the vessel wall, barrel, shroud, and primary coolant water. Activation of the vessel insulation and supports is considered and evaluated on the basis of the fluxes computed to exist in the annular cavity space during full power operation.

The fission and corrosion product reactor coolant inventory assumed to be present for shutdown is that specified in Table 11.1-2, corrected for decay up to the point in time of interest.

The activity of radioactive crud and its thickness on Reactor Coolant System surfaces have been evaluated using data from six pressurized water reactors. For the circulating crud, the observed activities were compared to activity values recommended for design by the Draft ANSI N237 Source Term Specification, 1976.

→(DRN 03-2066, R14)

←(DRN 03-2066, R14)

Table 11.1-10 shows the expected maximum specific activities for deposited corrosion products. The residual activity due to deposited corrosion products is evaluated with the assumption of a thickness of 0.16 mg. crud/cm^2 for steam generator tubing and a 1.5 mg. crud/cm^2 for piping and system crud level.

→(DRN 03-2066, R14)

The current FSAR design basis source terms are based on draft ANSI standard N237. Extended Power Uprate used ANSI/ANS-18.1-1999 as the basis for source terms. An evaluation of the two source terms and the change in the flux-to-dose conversion factors between ANSI 6.1.1 1977 and ANSI 6.1.1 1991 indicate that the EPU source terms are bounded by the current FSAR design basis source terms.

←(DRN 03-2066, R14)

12.2.1.6 Pressurizer Activity

The liquid section of the pressurizer has source terms due to plateout of radioactive crud plus dispersed fission and corrosive product activity in the water. The maximum deposited activity for the liquid section, assuming a crud film thickness of 1.5 mg. crud/cm^2, is derived from the values of Table 11.1-10. As an upper limit, the liquid section will have

dispersed fission and corrosion product activity equal to the primary coolant activity with one percent failed fuel specified in Table 11.1-2.

The activity in the steam section is due to the buildup of the gaseous fission products Xe and Kr from one percent failed fuel. This activity is shown in Table 12.2-6.

12.2.1.7 Contained Sources in Other Plant Systems

The source intensity in equipment and pipelines handling radioactive fluids is determined from activity in the reactor water by considering the processes that the reactor water has undergone prior to entering equipment and piping (dilution, filtering, demineralization, delay, change of phase, etc.)

In all cases the process or combination of processes leading to the highest activity is considered for conservatism. The maximum inventory of activity in the various components of the Chemical and Volume Control System, Boron Management System, Fuel Pool System, Safety Injection System and Waste Management System are listed in Tables 12.2-7, 12.2-8, 12.2-9, 12.2-10 and 12.2-11, respectively.

12.2.1.8 Spent Fuel and Spent Fuel Pool

The maximum and expected fission and corrosion product activities in the spent fuel pool are specified in Table 11.1-17.

The source terms employed to determine the minimum water depth above spent fuel and shielding walls around the spent fuel pool, as well as shielding of the spent fuel transfer tube, are given in Table 12.2-12. The activities shown in that table are equilibrium core activities based on 105 percent of full power core conditions.

→(LBDCR 16-016, R309)
The spent fuel pool contains greater than 8 cores as described in Subsection 9.1.2.
← (LBDCR 16-016, R309)

→(DRN 99-2362, R11)
To size the shield for the transfer tube, the hottest element is assumed to be transferred. Its activity is that computed from the assumption of a homogenous core inventory (Table 12.2-12) and an overall peaking factor of 1.80.
←(DRN 99-2362, R11)

12.2.1.9 Accident Sources
→(DRN 03-2066, R14)
The accident source terms which are employed to determine shielding requirements for emergency accessways and containment shielding, as well as potential doses to equipment inside containment following a loss-of-coolant accident (LOCA) are shown in Table 12.2-13. Table 12.2-13 assumes a release to containment of the activity stated in TID 14844[2], namely 100 percent noble gasses, 50 percent halogens, and 1 percent remaining fission product inventory.

The accident sources for the main control room are shown in Table 15.6-18.
←(DRN 03-2066, R14)
The accident sources for the Safety Injection System are shown in Table 12.2-10.
→(LBDCR 13-009, R307)
12.2.1.10 Low Level Radioactive Waste Storage Facility

The Low Level Radioactive Waste Storage Facility (LLRWSF) was added as part of LDCR 95-0059 for storage of radioactive waste awaiting disposal.

The radiation protection design of the LLRWSF, in terms of shielding and dose estimates is based upon the dose rates of the waste containers in the facility when the facility is filled to capability. Table 11.4-11 includes anticipated waste volumes by type for a fully loaded facility. The design basis radiation levels for the wastes to be stored in the facility are based on Waterford 3 historical source terms.
←(LBDCR 13-009, R307)

→(LBDCR 13-009, R307)
The radiation shielding configuration of the LLRWSF is designed in accordance with the dose rate criteria per 10CFR20, FSAR and site procedures for both the site boundary and for restricted areas. The nearest site boundary to the LLRWSF is River Road which is approximately 980 ft away. The facility is designed so that the maximum offsite dose rate at the site boundary from the waste stored in the LLRWSF will be maintained $\leq$ 0.05 rem/yr which is 10% of the allowable. The restricted area is the area surrounding the LLRWSF which will be controlled by the Radiation Protection Group for purposes of protection of individuals from exposure to radiation and radioactive materials. The facility is designed so that the maximum dose when the facility is fully loaded at approximately 60 ft. away is 0.3 mrem/hr.

The projected waste containers curie content is listed in Table 12.2-11. The location of the LLRWSF is shown in Figure 1.2-2.
←(LBDCR 13-009, R307)
→(LBDCR 13-010, R307)

12.2.1.11 Original Steam Generator and Reactor Vessel Head Storage Facility

In support of the Waterford 3 (W3) SG/RVCH Replacement Project, the Original Steam Generator Storage Facility (OSGSF) was added as a part of Steam Generators / Reactor Vessel Head replacement project. The Original Steam Generator Storage Facility (OSGSF) is located outside of the Protected Area and within the site Owner Controlled Area. The OSGSF is located well away from any safety-related onsite systems, structures, or components. The OSGSF is designed as a non-safety related structure to be used to provide secure storage until site decommissioning of the two Original Steam Generators (OSGs), the Original Reactor Vessel Closure Head (ORVCH) and Original Control element Drive Mechanisms (OCEDMs) that were removed and replaced during the W3 SG/RVCH Replacement Outage.

The location of the OSGSF is shown on FSAR Figure 1.2-2.

The movements of the OSGs, ORVCH and OCEDMs into the OSGSF were accomplished through an open side to the building facing Plant North towards Warehouse 5B. The closure of these openings has been accomplished by means of pre-cast concrete panels sealed to the OSGSF walls that minimize airflow through the panel joints.

The design of a reinforced concrete floor accounted for supporting the loads anticipated during facility construction and offloading and storage of the OSGs, ORVCH and OCEDMs. The floor slab was elevated two (2) feet above finished grade of 16' so that surface water could drain away from the building. The OSGSF floor was coated to create a barrier to prevent leaching of contamination into the floor.

The 24" wall concrete thickness has been established to allow unrestricted personnel access to the outside of the building in accordance with the NRC unrestricted access criteria found in 10 CFR 20. The OSGSF roof will not be accessed by members of the public, and thus, the 10 CFR 20 dose limit of 100 mR/yr is not applicable to the OSGSF roof.

The OSGSF is designed to provide adequate clearance between the stored components and wall surfaces to permit personnel to visually inspect these areas if needed.

The perimeter fence is equipped with a lockable gate which is controlled by Radiation protection. The dose rate at the perimeter fence is as depicted on Table 12.3-1. The OSGSF is designed for adequate flood protection for the OSGs, ORVCH, and OCEDMs. The OSGSF is not designed for occupation except for the capability to perform inspections. Accordingly, the interior of the OSGSF is not required to meet emergency egress requirements or to have installed ventilation systems. Similarly for the purpose of long term storage of the OSGs, ORVCH and OCEDMs, which requires no occupancy of the facilities by personnel, the OSGSF is not designed for any water, wastewater, electrical, or telephone services.

The building concrete shielding design meets the radiological requirements of 40 CFR 190, 10 CFR 20 and plant license requirements, and provides adequate shielding to limit the contact dose rate to 0.1 mR/hour on the exterior wall surface of the building.
←(LBDCR 13-010, R307)

→(LBDCR 13-010, R307)
The OSGSF roof is equipped with watertight membrane roofing system to preclude moisture instrusion in the storage facility.

The floor sumps with an external access are provided as common collection point for any liquid within the OSGSF. The sampling port will be used to sample any inventory that collects within the sump, as well as to remove such inventory without requiring entry into the OSGSF.

Two ground water sampling wells have been added down gradient from the OSGSF to monitor groundwater impact assuming an unlikely release of contaminated water from the OSGSF. Monitoring will be performed to identify and mitigate radiological contamination that could reach groundwater.
←(LBDCR 13-010, R307)
→(LBDCR 13-0020, R308)

12.2.1.12 Independent Spent Fuel Storage Installation (ISFSI)

In order to provide adequate spent fuel storage capacity for WF3, Entergy has established an ISFSI at WF3 on a site located south of the four large water storage tanks that are situated at the south end of the WF3 plant area, just west of the switchyard, within the Protected Area. The ISFSI pad is sized to store 72 HI-STORM storage casks, with each cask capable of storing 32 spent fuel assemblies, which is adequate to meet the projected WF3 spent fuel storage needs over the life of the nuclear power plant.

The WF3 ISFSI operates under the conditions of the general license in accordance with 10 CFR Part 72 regulations. The spent fuel storage cask designs that are approved for use under the general license are listed in 10 CFR 72.214. The HI-STORM 100 Cask System has been approved for use, and is listed in 10 CFR 72.214. The design basis for the HI-STORM 100 Cask System is provided in the Final Safety Analysis Report (FSAR) for the HI-STORM 100 Cask System and as supplemented by changes made by Entergy Operations, Inc., the general licensee, from the HI-STORM FSAR under the provisions of 10 CFR 72.48. Amendment No. 5 to the HI-STORM 100 CoC is being used as the licensing basis for the WF3 ISFSI and dry cask storage activities.

ISFSI operations is evaluated under the WF3 ISFSI 10 CFR 72.212 Evaluation Report, which includes the Radiological Evaluation for the ISFSI as required by 10 CFR 72.104.
←(LBDCR 13-0020, R308)

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Equipment cubicles, corridors, and areas normally occupied by operating personnel can contain small amounts of airborne radioactivity as a result of equipment leakage. For the purpose of evaluating the potential exposure to personnel from this activity. This subsection presents a description of the sources of activity and the models and parameters used to evaluate airborne radionuclide levels in the Reactor Auxiliary Building, the Reactor Building, the Fuel Handling Building and the Turbine Building. Table 12.2-14 includes assumptions, parameters and sources of airborne radioactivity used in the analysis. The sources are determined for each area assuming that leakage occurs in that area and that the leaking fluid contains a fraction of the reactor coolant activity. This fraction is determined from process consideration of leaking fluid (amount of filtering, degassing, demineralization etc. prior to leak). The leak rate is based on typical data from operating plants. The equilibrium airborne concentration is then determined by use of the standard equation of build-up and removal, where build-up is caused by leakage and removal both by radioactive decay and ventilation.

The isotopic airborne concentrations as a fraction of maximum permissible concentration in air (10CFR20) were calculated for those areas normally occupied by operating personnel and where a potential for high exposure exists. These values are presented in Table 12.2-15 and indicate that the dose to a critical organ of a worker, adjusted on the basis of weekly occupancy, would be well below the maximum allowable limit. Furthermore, the dose calculated based on these values would be highly conservative because the maximum permissible concentrations (MPCs) given in 10CFR20 are based on infinite cloud assumptions while volumes of the applicable areas are limited.

→(DRN 99-2362, R11)

A more detailed room by room analysis is performed for the Reactor Auxiliary Building. In this case airborne radionuclide concentrations in the form of ΣC/MPC were calculated as well as the whole body dose, in mrem per hour occupancy, resulting from inhalation and external exposure. The inhalation and external whole body dose conversion factors were taken from Table E-7 of Regulatory Guide 1.109, for the adult group Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the purpose of Evaluating Compliance with 10CFR Part 50, Appendix I (Revision 1) and Table D-3 of WASH 1258[3], respectively. The WASH-1258 values for a semi-infinite cloud model result in highly conservative dose values. The assumptions used in the analysis are essentially the same as those listed in Table 12.2-14. The differences are that for gaseous releases the iodine partition factor is assigned a valve of 1.0, and 0.05 for liquid leakage associated with the steam generator blow down heat exchanger, tank and pumps. Table 12.2-16 lists the results of the equipment and equipment leakage rates.

←(DRN 99-2362, R11)

A negligible amount of radioactivity is expected to be released due to removal of reactor vessel head, movement of spent fuel or relief valve venting. Therefore, contribution from these sources to airborne activity are not considered.

The airborne concentration of a radioisotope in an area having a constant leak rate, source strength and exhaust rate, can be calculated by the equation given below. Radionuclide concentrations in other areas such as corridors are calculated assuming the air in the corridor can be contaminated by exhaust from nearby areas.

→(DRN 02-110, R12)

FSAR Table 12.2-15a and 12.2-16a provide isotopic airborne concentrations as a function of derived air concentration (DAC) from 10 CFR 20, Appendix B.

←(DRN 02-110, R12)

→(DRN 99-2362, R11)

$$C_i\left(t\right)/\,MPC_i \;=\; W\,a_i\;\;SS\,PF_i\;\frac{\left(1-e^{-\lambda\tau_i t}\right)}{V\,\lambda\tau_i\,MPC_i}$$

←(DRN 99-2362, R11)

where,

W = leak rate of fluid in cm^3/hr

a_i = concentration of i^{th} isotope in the primary coolant in μci/cm^3

SS = source strength defined as fraction of primary coolant present in the leaking liquid

PF_i = partition factor of i^{th} isotope

$\lambda\tau_i$ = total removal rate constant for i^{th} isotope in $hr.^{-1}$.

= $\lambda d_i + \lambda e$

λd_i = decay constant for i^{th} isotope in $hr.^{-1}$

λe = removal rate constant due to exhaust in hr.

t = time interval in hours

V = free volume of the area where leak occurs in cm^3

$C_i(t)$ = airborne concentration of the i^{th} radioisotope at time t in μci/cm^3

For small rooms and other operating areas where λd for most of the radionuclides, the peak or equilibrium activity (Ceq) is given by the following equation:

$$\mathrm{Ceq}_i\,/\,\mathrm{MPC}_i \;=\; \frac{\mathrm{W\;a_i\;SS\;PF_i}}{\mathrm{V}\;\left(\lambda \mathrm{d}_i \;+\; \lambda \mathrm{e}\right)\mathrm{MPC}_i}$$

$$=\; \frac{\mathrm{W\;a_i\;SS\;PF_i}}{\left[\mathrm{CFM}\right]1.7x10^6\;\mathrm{MPC}_i}$$

In order to calculate tritium concentration in the Fuel Handling Building the following equations were used to calculate evaporation rate from the fuel pool:

→(DRN 99-2362, R11)

$$\mathrm{wp} \;=\; \frac{\mathrm{A}\left(95 + 0.425\upsilon\right)}{y}\left(\rho \mathrm{w} \;-\; \rho \mathrm{a}\right)$$

←(DRN 99-2362, R11)

and ventilation rate:

$$(\text{CFM}) = \frac{\text{wp}}{4.5\,(\text{Wi} - \text{Wo})}$$

where,

→(DRN 99-2362, R11)

wp	=	Evaporation rate of water in lbm/hr.

←(DRN 99-2362, R11)

ν	=	Air velocity over surface in ft/min.
y	=	Latent heat at pool surface water temperature in Btu/lb.

→DRN 99-2362, R11)

pa	=	Saturation pressure of vapor at room air dew point temperature in in. of mercury.
pw	=	Saturation pressure of vapor at the surface water temperature in in. of mercury.

←(DRN 99-2362, R11)

A	=	Surface area of pool in ft^2
Wo	=	Moisture content of out-door air in lbm/lbm of dry air
Wi	=	Moisture content of in-door air in lbm/lbm of dry air
CFM	=	Ventilation rate in ft^3/min.

SECTION 12.2: REFERENCES

1 - "DOT IIW User's Manual" WANL-TME-1982, December 1969.

2 - DiNunnho, Anderson, Bakes and Anderson, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, Atomic Energy Commission, March 23, 1962.

3 - "Numerical Guides for Design Objectives and Limiting Conditions for Operation to meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents", WASH 1258, AEC, July 1973.

→(DRN 99-2362, R11)

4 - C-CE-4034, February 16, 1977, information utilized in development of Section 12.2.1 Tables.

←DRN 99-2362, R11)

TABLE 12.2-1

NEUTRON FLUXES OUTSIDE THE REACTOR PRESSURE VESSEL

Maximum Neutron spectra (n/cm^2-sec)

E(mev)	Side	Top	Bottom
18	1.60(+4)*	-**	1.09(-2)
14	9.33(+5)	-	8.08(-1)
10	2.42(+7)	-	1.35(+1)
8	5.36(+7)	-	1.00(+1)
6	9.98(+7)	-	1.06(+1)
4	9.50(+7)	-	1.16(+1)
3	6.68(+7)	-	1.33(+1)
2	7.00(+7)	-	1.17(+1)
1	7.98(+7)	-	1.53(+1)
0.33	9.60(+7)	-	3.42(+1)
Epithermal	4.83(+10)	-	-
Thermal	1.15(+10)	-	-

* Denotes power of ten (10

** - denotes insignificant

TABLE 12.2-2

NEUTRON LEAKAGE RATE FROM AXIAL REGIONS OF THE REACTOR VESSEL*

Axial Region Boundaries		Height (cm)	Neutron Leakage Rate (n/sec.)
Upper (cm)	Lower (cm)		
341.6	281.6	60.0	1.8 (+12)**
281.6	251.6	30.0	4.4 (+12)
251.6	236.6	15.0	8.9 (+12)
236.6	229.1	7.5	8.0 (+12)
229.1	221.6	7.5	1.1 (+13)
221.6	214.1	7.5	1.5 (+13)
214.1	206.6	7.5	1.8 (+13)
206.6	191.6	15.0	5.0 (+13)
191.6	161.6	30.0	1.51 (+14)
161.6	131.6	30.0	2.10 (+14)
131.6	101.6	30.0	2.55 (+14)
101.6	71.6	30.0	2.90 (+14)
71.6	11.6	60.0 around core centerline	6.12 (+14)
11.6	-48.4	60.0	5.54 (+14)
-48.4	-108.4	60.0	4.73 (+14)
-108.4	-168.4	60.0	2.13 (+14)

* Bottom of the core is at elevation - 168.4 cm, and top of the core is at elevation 168.4 cm.

** Denotes power of ten (10).

TABLE 12.2-3

GAMMA FLUXES AT FULL POWER OPERATION

Maximum Gamma Spectra (γ/cm^{2}sec.)			
E(mev)	Side	Top	Bottom
10.00	3.05(+7)*	3.92(+2)	7.34(+3)
9.00	2.62(+8)	2.90(+3)	1.08(+7)
8.00	4.97(+8)	3.08(+3)	1.93(+7)
7.00	5.40(+8)	5.18(+3)	3.42(+7)
6.00	5.99(+8)	6.33(+3)	4.91(+7)
5.00	7.40(+8)	8.03(+3)	6.71(+7)
4.00	9.45(+8)	1.05(+4)	8.88(+7)
3.00	1.40(+9)	1.37(+4)	1.25(+8)
2.00	2.51(+9)	2.00(+4)	1.94(+8)
1.38	1.14(+9)	1.21(+4)	1.14(+8)
1.00	8.23(+8)	1.08(+4)	1.07(+8)
0.75	1.96(+9)	1.49(+4)	1.51(+8)
0.50	3.29(+9)	2.42(+4)	2.51(+8)
0.25	3.79(+9)	2.95(+4)	3.21(+8)

* Denotes power of ten (10)

TABLE 12.2-4

REACTOR COOLANT SYSTEM SOURCES (GAMMA SPECTRA)
AT SHUTDOWN

E(mev)	Maximum Decay Gamma Spectra (γ/cm^2 -sec.)				
	1hr.	5 hr.	20 hr.	2d	10d
4.00	1.15(+7)*	1.56(+5)	1.08(+4)	8.28(+3)	5.54(+3)
3.00	2.21(+7)	1.98(+6)	1.59(+5)	1.38(+5)	9.63(+4)
2.00	4.40(+7)	8.53(+6)	3.42(+6)	3.14(+6)	1.96(+6)
1.38	3.04(+7)	6.28(+6)	2.72(+6)	2.45(+6)	1.53(+6)
1.00	2.81(+7)	5.98(+6)	2.62(+6)	2.34(+6)	1.47(+6)
0.75	3.90(+7)	8.46(+6)	3.75(+6)	3.34(+6)	2.09(+6)
0.50	6.26(+7)	1.35(+7)	5.99(+6)	5.34(+6)	3.33(+6)
0.25	7.79(+7)	1.66(+7)	7.32(+6)	6.51(+6)	4.04(+6)

* Denotes power of ten (10)

TABLE 12.2-5

REACTOR COOLANT SYSTEM SOURCES (MATERIAL ACTIVATION SPECTRA) AT SHUTDOWN

Maximum Material Activation Spectra (γ/cm^2-sec)

E(mev)	1 hr	5 hr	20 hr	2d	10d
2.00	1.33(+7)*	4.49(+6)	7.87(+4)	--	--
1.38	2.25(+6)	2.25(+6)	2.25(+6)	2.25(+6)	2.25(+6)
1.00	3.71(+7)	1.52(+7)	3.99(+6)	3.80(+6)	3.80(+6)

* Denotes power of ten (10)

TABLE 12.2-6

PRESSURIZER STEAM SECTION ACTIVITY

Isotope	Activity *(m Ci/cc)
Kr - 85 m	4.1 (-1) **
Kr - 85	1.28 (+3)
Kr - 87	6.6 (-2)
Kr - 88	4.5 (-1)
Xe-133	1.73 (+3)
Xe-135	4.09 (+0)
Xe-138	6.00 (-3)

*Assumes that all Xe and Kr isotopes build up for 1 yr. and are supplied to the pressurizer at a continuous primary water spray rate of 1.5 gpm, with complete stripping of the gas from water.

**Denotes power of ten (10)

TABLE 12.2-7 (Sheet 1 of 2)

MAXIMUM ACTIVITY INVENTORY IN CVCS COMPONENTS (c)
(CURIES)

Nuclide	Regenerative Heat Exchanger	Letdown Heat Exchanger	Purification Filter	Purification Ion-Exchanger

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 12.2-7 (Sheet 2 of 2)

Nuclide	Deborating Ion-Exchanger	Volume Control Tank	Boric Acid Makeup Tank	Charging Pump

Security-Related Information
Text Withheld Under 10 CFR 2.390

MAXIMUM ACTIVITY INVENTORY IN BMS COMPONENTS
(CURIES)

→(DRN 00-1046; 00-805)

Nuclide	Reactor Drain Tank	Equipment Drain Tank	Flash Tank (1)	Boric Acid Condensate Tank

←(DRN 00-805)

Volume

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 12.2-8 (Sheet 2 of 2)

Nuclide	Pre-Concentrator Filter	Pre-Concentrator Ion-Exchanger	Boric Acid Condensate Ion-Exchanger	Holdup Tanks

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 12.2-9

MAXIMUM ACTIVITY INVENTORY IN FPS COMPONENTS

Nuclide	Fuel Pool Ion-Exchanger	Pool Purification Filter	Fuel Pool Heat Exchanger

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 12.2-10

MAXIMUM AND EXPECTED ACTIVITY INVENTORY IN SIS COMPONENTS
(CURIES)

Shutdown Heat Exchanger

	MAXIMUM	
	Shutdown	LOCA

Security-Related Information
Text Withheld Under 10 CFR 2.390

→(LBDCR 13-009, R307)

TABLE 12.2-11 (Sheet 1 of 6) Revision 307 (07/13)

←(LBDCR 13-009, R307)

MAXIMUM ACTIVITY INVENTORY IN WMS COMPONENTS
(Curies)

→(DRN 01-1249, R11-B)

Nuclide	Waste Tank	Laundry Tank	Waste Condensate Tank	Waste Demineralizer System	Waste Condensate Ion Exchanger

←(DRN 01-1249, R11-B)

Security-Related Information
Text Withheld Under 10 CFR 2.390

→(LBDCR 13-009, R307)

TABLE 12.2-11 (Sheet 2 of 6) Revision 307 (07/13)

←(LBDCR 13-009, R307)

Nuclide	Waste Filter	Waste Oil Filter	Laundry Filter	Spent Resin Tank	Gas Decay Tank	Gas Surge Tank

Security-Related Information
Text Withheld Under 10 CFR 2.390

(LBDCR 13-009, R307)

TABLE 12.2-11 (Sheet 3 of 6) Revision 307 (07/13)

Maximum Anticipated Activity per Container Stored in the
Low Level Radioactive Waste Storage Facility
(Curies)

Dry Active Waste Container
Maximum Anticipated Activity per Container

Nuclide	**1280 Cubic Feet Container**	**Curies per Container**

Security-Related Information
Text Withheld Under 10 CFR 2.390

(LBDCR 13-009, R307)

(LBDCR 13-009, R307)

TABLE 12.2-11 (Sheet 4 of 6) Revision 307 (07/13)

Maximum Anticipated Activity per Container Stored in the
Low Level Radioactive Waste Storage Facility
(Curies)

Plant Bead Resin Container
Maximum Anticipated Activity per Container

Nuclide	**Container 145 Cubic Feet**	**Curies per Container**

Security-Related Information
Text Withheld Under 10 CFR 2.390

Liquid Waste Management Bead Resin Container
Maximum Anticipated Activity per Container

Nuclide	**Container 205 Cubic Feet**	**Curies per Container**

Security-Related Information
Text Withheld Under 10 CFR 2.390

(LBDCR 13-009, R307)

→(LBDCR 13-009, R307)

TABLE 12.2-11 (Sheet 5 of 6) Revision 307 (07/13)

Maximum Anticipated Activity per Container Stored in the
Low Level Radioactive Waste Storage Facility
(Curies)

Plant Filters, Class A & B Container
Maximum Anticipated Activity per Container

Nuclide	Container 120 Cubic Feet	Curies per Container

Security-Related Information
Text Withheld Under 10 CFR 2.390

Secondary Blowdown Bead Resin Container
Maximum Anticipated Activity per Container

Nuclide	Container 205 Cubic Feet	Curies per Container

Security-Related Information
Text Withheld Under 10 CFR 2.390

←(LBDCR 13-009, R307)

➔(LBDCR 13-009, R307)

TABLE 12.2-11 (Sheet 6 of 6) Revision 307 (07/13)

Maximum Anticipated Activity per Container Stored in the
Low Level Radioactive Waste Storage Facility
(Curies)

Condensate Polisher Powder Resin Container
Maximum Anticipated Activity per Container

Nuclide	**Container 205 Cubic Feet**	**Curies per Container**

Security-Related Information
Text Withheld Under 10 CFR 2.390

🡐(LBDCR 13-009, R307)

TABLE 12.2-12 Revision 14 (12/05)

LOCA CORE INVENTORY (Curies/MWt) (a)

Nuclide	(Ci)/MWt	Nuclide	(Ci)/MWt	Nuclide	(Ci)/MWt

→(DRN 03-2066, R14)

Security-Related Information
Text Withheld Under 10 CFR 2.390

(a) Core inventories are based on 100.5% of full power core conditions.

←(DRN 03-2066, R14)

TABLE 12.2-12A Revision 14 (12/05)

CORE INVENTORY FOR STEAMING EVENTS (Curies)

→(DRN 03-2066, R14)

Isotope	Curies

Security-Related Information
Text Withheld Under 10 CFR 2.390

←(DRN 03-2066, R14)

TABLE 12.2-13

FISSION PRODUCT GAMMA SOURCE IN CONTAINMENT BUILDING (Mev/sec)
(assuming 100% noble gases, 50% halogens, 1% solids)

Energy Interval (Mev)

Time	.1 - .4	.4 - .9	.9 - 1.35	1.35 - 1.8	1.8 - 2.2	2.2 - 2.6	2.6
0	3.08(18)*	1.84(19)	7.30(18)	1.37(19)	8.90(18)	6.41(18)	2.90(18)
.5 hr.	2.93(18)	1.62(19)	6.65(18)	5.02(18)	4.63(18)	5.19(18)	3.71(17)
1 hr.	2.82(18)	1.43(19)	5.81(18)	4.45(18)	3.12(18)	4.30(18)	1.36(17)
2 hr.	2.68(18)	1.14(19)	4.70(18)	3.56(18)	2.14(18)	3.05(18)	4.38(16)
8 hr.	2.09(18)	6.41(18)	2.16(18)	1.58(18)	6.91(17)	5.76(17)	1.56(16)
24 hr.	1.16(18)	4.45(18)	8.68(17)	6.23(17)	2.32(17)	1.03(17)	2.57(14)
1 wk.	3.06(17)	1.07(18)	1.63(17)	1.75(17)	5.88(16)	3.11(16)	1.74(14)
1 mo.	4.10(16)	1.28(17)	3.17(15)	2.42(16)	1.46(15)	1.85(15)	5.52(13)
2 mo.	5.98(15)	7.86(16)	5.37(14)	4.30(15)	7.72(14)	2.85(14)	1.78(13)
4 mo.	7.51(14)	4.35(16)	1.98(14)	4.87(14)	6.45(14)	3.41(13)	7.72(12)

* Denotes power of ten (10)

TABLE 12.2-14 (Sheet 1 of 2) Revision 12 (10/02)

ASSUMPTIONS AND PARAMETERS USED TO CALCULATE AIRBORNE CONCENTRATIONS

→(DRN 99-2362)

Leakage Rates and Source Stream (SS):	
←(DRN 99-2362)	
Leakage into containment	1% of the noble gas inventory/d .001% of the iodine inventory/d SS = 1
→(DRN 00-1046; 02-110)	
Steam leakage into turbine building	1700 lb/h*
←(DRN 02-110)	
Leakage into Reactor Auxiliary Building	160 lb./d, SS = 1
Leakage from the gas surge tank	0.005 scfm, **
Leakage from the waste gas compressor room	0.02 scfm, **
Leakage from letdown heat exchanger room	6.6 gal./d, SS = 1
←(DRN 00-1046)	
Corridor elevation - 4 ft. MSL in RAB	10% of the exhaust from nearest rooms flows into the corridor
Partition Factors:	
Turbine Building	1 for iodine, 1 for noble gases
Reactor Auxiliary Building	0.0075 for iodines, 1 for noble gases
Letdown heat exchanger	0.1 for iodines, 1 for noble gases.
Ventilation Rates (cfm):	
Containment	Isolated case
Turbine Building	636,000
Fuel Handling Building	26,000
Reactor Auxiliary Building	77,000
Gas surge tank room	365
Waste gas compressor room	600
Letdown heat exchanger room	450
Volumes (Cu. Ft.):	
Containment	2.7×10^6
Turbine Building	2.5×10^6
Fuel Handling Building	4.08×10^5
Reactor Auxiliary Building	2.0×10^6
Other Factors:	
Failed fuel fraction	0.12%
Plant load	80%
Outside air condition (winter)	56.1 F, 76.7% Relative Humidity

TABLE 12.2-14 (Sheet 2 of 2)

Other Factors: (Cont'd)

Fuel pool parameters:	
Surface temperature	150°F
Surface Area	1152 ft^2
Air velocity over surface	10 ft./min.

* Concentration of the liquid leaking into the turbine was assumed to be the same as that for secondary coolant.

** Gaseous source terms were calculated assuming degassification of 0.540 gpm of reactor coolant and an over all decontamination factor of 2000 for iodine and waste gas (flow rate 30,000 scf/yr.)

TABLE 12.2-15 Revision 12 (10/02)

AVERAGE AIRBORNE C/MPC IN REACTOR AUXILIARY BUILDING, TURBINE BUILDING, CONTAINMENT AND FUEL HANDLING BUILDING

ISOTOPE	CONTAINMENT (C/MPC)	FUEL HANDLING BUILDING (C/MPC)	TURBINE BUILDING (C/MPC)	REACTOR AUXILIARY BUILDING (C/MPC)

Security-Related Information
Text Withheld Under 10 CFR 2.390

→(DRN 02-110, R12)

Table 12.2-15a (Sheet 1 of 2) Revision 14 (12/05)

Average Airborne C/DAC in Reactor Auxiliary Building, Turbine Building,
Containment and Fuel Handling Building

→(DRN 05-455, R14)

Isotope	Containment Building		Fuel Handling Building		Turbine Building		Reactor Auxiliary Building	
	C (uCi/cc)	C / DAC	C (uCi/cc)	C / DAC	C (uCi/cc)	C / DAC	C (uCi/cc)	C / DAC

Security-Related Information
Text Withheld Under 10 CFR 2.390

←(DRN 02-110, R12; 05-455, R14)

→(DRN 02-110, R12)

Table 12.2-15a (Sheet 2 of 2) Revision 14 (12/05)

Average Airborne C/DAC in Reactor Auxiliary Building, Turbine Building, Containment and Fuel Handling Building

→(DRN 05-455, R14)

Isotope	Containment Building		Fuel Handling Building		Turbine Building		Reactor Auxiliary Building	
	C (uCi/cc)	C / DAC	C (uCi/cc)	C / DAC	C (uCi/cc)	C / DAC	C (uCi/cc)	C / DAC

Security-Related Information
Text Withheld Under 10 CFR 2.390

←(DRN 02-110, R12; 05-455, R14)

TABLE 12.2-16 (Sheet 1 of 4) Revision 11-A (02/02)

REACTOR AUXILIARY BUILDING ROOM BY ROOM C/MPC AND WHOLE BODY DOSE COMMITMENT VALUES

Item	Location and/or Component	Elevation (ft. MSL)	Leakage Rate (gpd)	ΣC/MPC	Dose Commitment (mrem/hr occupancy) Inhalation Whole Body	External Whole Body
1	Shutdown Cooling Heat Exchanger A and B	-35	1.0(1)[(a)]	2.29	3.15(-2)	7.98(-1)
2	Valve Operating Closure A and B	-15.5	1.0	4.58(-1)	6.30(-3)	1.60(-1)
3	Below Valve Operating Closure B Sump #6 and Pumps	-35	0	4.64	6.37(-2)	1.62
4	Containment Spray Pump A, LPSI Pump A, HPSI Pumps A and A/B, Sumps #7 and #8 and Pumps	-35	6.6	1.56	2.15(-2)	5.45(-1)
5	Containment Spray Pump B, LPSI Pump B, HPSI Pump B, Equip. Drain Tk Pump, Reactor Drain Tk Pump, Sump #5 and Pump	-35	1.96(1)	4.64	6.37(-2)	1.62
6	Sump #1 and Pumps	-35	6.6	8.09(-1)	6.30(-2)	2.63(-1)
7	Equip. Drain Tk	-35	3.3	1.48(-1)	1.16(-2)	4.82(-2)
8	Emergency FW Pump B	-35	0	-(b)		
9	Emergency FW Pump A	-35	0	-	-	-
10	Emergency FW Pump (Turbine Driven)	-35	0	-	-	-
11	Component Cooling Water Makeup Pumps A and B, Oil Sump #3 and Pump	-35	0	-	-	-
→(DRN 00-1046)						
12	Gas Surge Tank	-35	5.0(-3)scfm	5.46	1.53(-1)	1.30
←(DRN 00-1046)						
13	Gas Decay Tank C	-35	2.0(-2)scfm	1.66(2)	4.66	39.5
14	Waste Gas Compressor B	-35	2.0(-2)scfm	3.32	9.32(-2)	7.90(-1)
15	Gas Decay Tank B	-35	2.0(-2)scfm	1.66(2)	4.66	39.5
16	Waste Gas Compressor A	-35	2.0(-2)scfm	3.32	9.32(-2)	7.90(-1)
17	Gas Decay Tank A	-35	2.0(-2)scfm	1.66(2)	4.66	39.5
18	Charging Pump A	-35	6.6	4.79	4.72(-2)	1.96
19	Charging Pump A/B	-35	6.6	4.79	4.72(-2)	1.96
20	Charging Pump B	-35	6.6	4.79	4.72(-2)	1.96
21	Waste Tank, Waste Tk Pump B	-35	9.9	7.76(-2)	2.58(-2)	8.0(-4)
22	Waste Tank, Waste Tk Pump A, Sump #11 and Pump	-35	9.9	7.76(-2)	2.58(-2)	8.0(-4)

TABLE 12.2-16 (Sheet 2 of 4)

Item	Location and/or Component	Elevation (ft. MSL)	Leakage Rate (gpd)	ΣC/MPC	Dose Commitment (mrem/hr occupancy) Inhalation Whole Body	External Whole Body
23	Laundry Filter	-35	6.6	1.78(-1)	1.76(-1)	5.87(-6)
24	Oil Separator	-35	6.6	5.69(-1)	1.89(-1)	5.87(-3)
25	Laundry Tank A and B, Laundry Pump A and B, Detergent Sump #1 and Pumps	-35	19.8	5.20(-2)	5.11(-2)	1.71(-6)
26	Waste Filter	-35	6.6	5.69(-1)	1.89(-2)	5.87(-3)
27	Waste Condensate Pumps A and B, Chem. Waste Tank and Pump Sample Recovery Tank and Pump	-35	6.6	2.76(-2)	2.73(-2)	8.84(-8)
28	Waste Condensate Tanks A and B	-35	6.6	8.10(-3)	7.99(-3)	2.59(-8)
29	Sump #10 and Pumps, Plumbing Valve Pit, Refueling Storage Pool Leak Detection Station, Condensate Storage Pool Leak Detection Station	-35	0	-	-	-
30	Elevator Machine Room	-35	0	-	-	-
31	Holdup Tank 1-D	-35	3.3	2.81(-1)	2.05(-2)	9.02(-3)
32	Holdup Tank 1-B	-35	3.3	2.81(-1)	2.05(-2)	9.02(-3)
33	Holdup Tank 1-C	-35	3.3	2.81(-1)	2.05(-2)	9.02(-3)
34	Holdup Tank 1-A	-35	3.3	2.81(-1)	2.05(-2)	9.02(-3)
35	Acid Neutralizing Tank	-35	0	-	-	-
36	Boric Acid Makeup Tanks A and B, Boric Acid Pumps A and B	-35	19.8	4.59(-2)	3.17(-2)	2.58(-4)
37	Holdup Drain Pump, Holdup Recirc Drain Pump, Holdup Recirc Pump	-35	19.8	1.58	1.16(-1)	5.08(-2)
38	Boric Acid Preconcentrator Filter B	-35	6.6	2.67	1.95(-1)	8.57(-2)
39	Boric Acid Preconcentrator Filter A	-35	6.6	2.67	1.95(-1)	8.57(-2)
40	Shield Door Area	-35	0	-	-	-
41	Boric Acid Cond. Tanks A, B, C and D, Boric Acid Cond. Pumps A and B, Sump #9 and Pumps	-35	2.64(1)	2.86(-2)	2.81(-2)	9.12(-7)
42	Waste Condensate Ion Exchanger	-35	3.3	2.08(-2)	2.04(-2)	6.63(-7)
43	Spent Resin Tank	-35	3.3	2.88(1)	1.01	4.31(-1)

TABLE 12.2-16 (Sheet 3 of 4)

Revision 305 (11/11)

Item	Location and/or Component	Elevation (ft. MSL)	Leakage Rate (gpd)	ΣC/MPC	Dose Commitment (mrem/hr occupancy) Inhalation Whole Body	External Whole Body
44	Corridor	-35	1.98	3.59(-2)	3.55(-4)	1.47(-2)
45	Purification Filter	-4	6.6	2.27(1)	3.12(-1)	7.90
→(DRN 00-805, R11-B)						
46	Flash Tank (c)	-4	3.3	2.06	2.83(-2)	7.19(-1)
47	Flash Tk Pumps A and B (c)	-4	1.32(1)	1.43	1.04(-1)	4.57(-2)
←(DRN 00-805, R11-B)						
→(DRN 99-1032, R12)						
48	Boronometer (Note: This device has been functionally abandoned-in-place by DC 3432.)	-4				
←(DRN 99-1032, R12)						
49	Volume Control Tank	-4	3.3	8.71(-1)	8.60(-3)	3.57(-1)
50	Fuel Pool Filter	-4	3.3	9.05(-2)	7.38(-2)	2.44(-4)
→(EC-4019, R305)						
51	Chemical Addition Tank and Strainer / Zinc Inj. Skid	-4	9.9	3.64	2.84(-1)	1.19
←(EC-4019, R305)						
52	Deborating Ion Exchanger, Purification Ion Exchanger B	-4	6.6	4.45	6.12(-2)	1.55
53	Purification Ion Exchanger A, Fuel Pool Ion Exchanger	-4	3.3	1.86	2.56(-2)	6.48(-1)
54	Preconcentrator Ion Exchanger B, Boric Acid Cond. Ion Exchanger B	-4	6.6	1.07	7.81(-2)	3.43(-2)
55	Preconcentrator Ion Exchanger A. Boric Acid Cond. Ion Exchanger A	-4	6.6	1.07	7.81(-2)	3.43(-2)
56	Letdown Heat Exchanger and Strainer	-4	1.32(1)	3.21(1)	1.49	3.83
57	Blowdown Pumps A and B	-4	6.6	9.48(-2)	9.47(-2)	3.99(-5)
58	Filter Flush Tank and Pump	-4	9.9	1.21	9.47(-2)	3.95(-1)
59	Blowdown Heat Exchanger A and B	-4	3.0	4.26(-2)	3.34(-2)	1.37(-4)
60	Blowdown Filters A and B	-4	1.32(1)	7.24(-2)	5.90(-2)	1.96(-4)
61	Blowdown Demineralizers A and B	-4	6.6	3.95(-2)	3.22(-2)	1.07(-4)
62	Acid Storage Tank, Caustic Storage Tank and Heaters, Chemical Feed Tank and Pump	-4	0	-	-	-
63	Boric Acid Concentrator A	-4	1.65(1)	8.21(-2)	2.06(-2)	9.17(-3)
64	Boric Acid Concentrator B	-4	1.65(1)	8.21(-2)	2.06(-2)	9.17(-3)
65	Waste Concentrator	-4	1.65(1)	5.37(-2)	1.79(-2)	5.53(-4)
66	Pipe Penetration Area	-4	1.0(1)	1.03	4.18(-2)	1.86(-1)

TABLE 12.2-16 (Sheet 4 of 4) Revision 11-B (06/02)

Item	Location and/or Component	Elevation (ft. MSL)	Leakage Rate (gpd)	ΣC/MPC	Dose Commitment (mrem/hr occupancy) Inhalation Whole Body	External Whole Body
67	Corridor	-4	1.32	1.08(-1)	5.0(-3)	1.29(-2)
68	Component Cooling Water Chemical Feed Tank	+21	0	-	-	-
69	Boric Acid Batching Tank and Strainer	+21	9.9	2.29(-1)	4.82(-2)	2.79(-3)
70	Waste Concentrate Storage Tank and Metering Pump	+21	6.6	1.02	7.81(-2)	1.47(-2)
71	Vault Area	-35	3.3	1.52(-3)	1.24(-3)	4.11(-6)
72	Aux. Component Cooling Water Pumps A and B	-35	0	-	-	-
73	Refueling Water Pool Purification Pump, Sump #3 and Pumps	-35	3.3	1.52(-3)	1.24(-3)	4.11(-6)
74	Blowdown Tank	-4	3.3	1.41(-1)	1.35(-1)	5.70(-5)

Notes

[a] represents powers of 10

[b] represents clean

→(DRN 00-805)

[c] The Flash Tank and pumps have been made inactive per ER-W3-00-0225-00-00.

←(DRN 00-805)

→(DRN 02-110, R12) Table 12.2-16a (Sheet 1 of 4) Revision 14 (12/05)

Reactor Auxiliary Building Room by Room C/DAC and Dose Commitment Values

→(DRN 05-455, R14)

Item	Location and / or Component	Elevation (Ft. MSL)	Ventilation Rate (CFM)	Leakage Rate (gpd)	Conc. C (uCi/cc)	C/DAC	Dose Commitment (mRem/hr Occupancy) DDE Submersion	CEDE Inhalation	CDE-Thyroid Inhalation
1	Shutdown Cooling Heat Exchanger A&B	-35	1,500	10.0	1.61E-06	4.28E-02	6.94E-02	5.68E-03	9.14E-02
2	Valve Operating Closure A and B	-35	750	1.0	3.01E-07	8.56E-03	1.39E-02	1.14E-03	1.83E-02
3	Below Valve Operating Closure B Sump #6 and Pumps	-35		0	3.05E-06	8.68E-02	1.41E-01	1.15E-02	1.85E-01
4	Containment Spray Pump A, LPSI Pump A, HPSI Pumps A and A/B, Sumps #7 and #8 and Pumps	-35	1,450	6.6	1.03E-06	2.92E-02	4.74E-02	3.88E-03	6.24E-02
5	Containment Spray Pump B, LPSI Pump B, HPSI Pump B Equip. Drain Tk Pump, Rx Drain Tk Pump, Sump #5 & Pump	-35	1,450	19.6	3.05E-06	8.68E-02	1.41E-01	1.15E-02	1.85E-01
6	Sump #1 and Pumps	-35	300	6.6	4.96E-07	1.41E-02	2.29E-02	1.87E-03	3.03E-02
7	Equipment Drain Tank	-35	820	3.3	9.08E-08	2.58E-03	4.19E-03	3.43E-04	5.52E-03
8	Emergency FW Pump B	-35	-	0	CLEAN	-	-	-	-
9	Emergency FW Pump A	-35	-	0	CLEAN	-	-	-	-
10	Emergency FW Pump (Turbine Driven)	-35	-	0	CLEAN	-	-	-	-
11	Component Cooling Water Makeup Pumps A and B, Oil Sump #3 and Pump	-35	-	0	CLEAN	-	-	-	-
12	Gas Surge Tank	-35	365	0.05 scfm	3.33E-05	8.70E-01	1.50E+00	6.02E-02	1.35E-01
13	Gas Decay Tank C	-35	120	0.02 scfm	4.05E-04	1.06E+01	1.82E+01	7.31E-01	1.63+00
14	Waste Gas Compressor B	-35	600	0.02 scfm	8.09E-06	2.12E-01	3.64E-01	1.46E-02	3.28E-02
15	Gas Decay Tank B	-35	120	0.02 scfm	4.05E-04	1.06E+01	1.82E+01	7.31E-01	1.63E+00
16	Waste Gas Compressor A	-35	600	0.02 scfm	8.09E-06	2.12E-01	3.64E-01	1.46E-02	3.28E-02
17	Gas Decay Tank A	-35	120	0.02 scfm	4.05E-04	1.06E+01	1.82E+01	7.31E-01	1.63E+00
18	Charging Pump A	-35	400	6.6	3.66E-06	9.43E-02	1.68E-01	1.41E-03	2.26E-02
19	Charging Pump A/B	-35	400	6.6	3.66E-06	9.43E-02	1.68E-01	1.41E-03	2.26E-02
20	Charging Pump B	-35	400	6.6	3.66E-06	9.43E-02	1.68E-01	1.41E-03	2.26E-02
21	Waste Tank, Waste Tank Pump B	-35	1,100	9.9	2.38E-08	1.21E-03	1.17E-03	7.67E-04	1.23E-02
22	Waste Tank, Waste Tank Pump A, Sump #11 and Pump	-35	1,100	9.9	2.38E-08	1.21E-03	1.17E-03	7.67E-04	1.23E-02
23	Laundry Filter	-35	100	6.6	1.46E-07	3.72E-03	6.69E-03	5.62E-06	9.05E-05

←(DRN 02-110, R12; 05-455, R14)

→(DRN 02-110, R12)

Table 12.2-16a (Sheet 2 of 4) Revision 14 (12/05)

Reactor Auxiliary Building Room by Room C/DAC and Dose Commitment Values

→(DRN 05-455, R14)

Item	Location and / or Component	Elevation (Ft. MSL)	Ventilation Rate (CFM)	Leakage Rate (gpd)	Conc. C (uCi/cc)	C/DAC	Dose Commitment (mRem/hr Occupancy) DDE Submersion	CEDE Inhalation	CDE-Thyroid Inhalation
24	Oil Seperator	-35	100	6.6	1.75E-07	8.90E-03	8.55E-03	5.62E-03	9.05E-02
25	Laundry Tank A and B, Laundry Pump A and B, Detergent Sump #1 and pumps	-35	1,025	19.8	4.28E-08	1.09E-03	1.96E-03	1.65E-06	2.65E-05
26	Waste Filter	-35	100	6.6	1.75E-07	8.90E-03	8.55E-03	5.62E-03	9.05E-02
27	Waste Condensate Pumps A and B, Chem Waste Tank and Pump, Sample Recovery Tank and Pump	-35	645	6.6	2.26E-08	5.77E-04	1.04E-03	8.72E-08	1.40E-06
28	Waste Condensate Tanks A and B	-35	2,015	6.6	7.25E-09	1.85E-04	3.32E-04	2.79E-08	4.49E-07
29	Sump #10 and Pumps, Plumbing Valve Pit, Refueling Storage Pool Leak Detection Station, Condensate Storage Pool Leak Detection Station	-35	-	0	CLEAN	-	-	-	-
30	Elevator Machine Room	-35	-	0	CLEAN	-	-	-	-
31	Holdup Tank 1-D	-35	540	3.3	3.61E-08	4.63E-03	2.11E-03	4.68E-03	7.54E-02
32	Holdup Tank 1-B	-35	600	3.3	3.61E-08	4.63E-03	2.11E-03	4.68E-03	7.54E-02
33	Holdup Tank 1-C	-35	540	3.3	3.61E-08	4.63E-03	2.11E-03	4.68E-03	7.54E-02
34	Holdup Tank 1-A	-35	600	3.3	3.61E-08	4.63E-03	2.11E-03	4.68E-03	7.54E-02
35	Acid Neutralizing Tank	-35	-	0	CLEAN	-	-	-	-
36	Boric Acid Makeup Tanks A and B, Boric acid Pumps A & B	-35	1,750	19.8	2.60E-08	8.15E-04	1.21E-03	1.93E-04	3.10E-03
37	Holdup Drain Pump, Holdup Recirc Drain Pump, and Holdup Recirc Pump	-35	810	19.8	1.60E-07	2.06E-02	9.38E-03	2.08E-02	3.35E-01
38	Boric Acid Preconcentrator Filter B	-35	160	6.6	2.71E-07	3.47E-02	1.58E-02	3.51E-02	5.66E-01
39	Boric Acid Preconcentrator Filter A	-35	160	6.6	1.20E-06	9.56E-01	1.58E-02	3.51E-02	5.66E-01
40	Shield Door Area	-35	-	0	CLEAN	-	-	-	-
41	Boric Acid Cond. Tanks A, B, C and D, Boric Acid Cond. Pumps A and B, Sump #9 and Pumps	-35	2,500	26.4	2.34E-08	5.96E-04	1.07E-03	8.99E-07	1.45E-05
42	Waste Condensate Ion Exchanger	-35	430	3.3	1.70E-08	4.33E-04	7.77E-04	6.54E-07	1.05E-05
43	Spent Resin Tank	-35	660	3.3	3.28E-06	4.21E-01	1.92E-01	4.26E-01	6.86E+00
44	Corridor	-35	19,840	1.98	3.71E-08	1.10E-03	1.71E-03	1.40E-04	2.26E-03

←(DRN 02-110, R12; 05-455, R14)

→(DRN 02-110, R12)

Table 12.2-16a (Sheet 3 of 4) Revision 305 (11/11)

Reactor Auxiliary Building Room by Room C/DAC and Dose Commitment Values

→(DRN 05-455, R14)

Item	Location and / or Component	Elevation (Ft. MSL)	Ventilation Rate (CFM)	Leakage Rate (gpd)	Conc. C (uCi/cc)	C/DAC	Dose Commitment (mRem/hr Occupancy) DDE Submersion	CEDE Inhalation	CDE-Thyroid Inhalation
45	Purification Filter	-4	100	6.6	1.49E-05	4.24E-01	6.87E-01	5.62E-02	9.05E-01
46	Flash Tank	-4	550	3.3	1.35E-06	3.85E-02	6.25E-02	5.11E-03	8.23E-02
47	Flash Tank Pumps A and B	-4	600	13.2	1.44E-07	1.85E-02	8.45E-03	1.87E-02	3.02E-01
48	Boronometer	-4	140	3.3	5.32E-06	1.51E-01	2.45E-01	2.01E-02	3.23E-01
49	Volume Control Tank	-4	1,100	3.3	6.65E-07	1.71E-02	3.05E-02	2.56E-04	4.11E-03
50	Fuel Pool Filter	-4	120	3.3	6.21E-08	1.77E-03	2.86E-03	2.34E-04	3.77E-03
→(EC-4019, R305)									
51	Chemical Addition Tank and Strainer / Zinc Inj. Skid	-4	100	9.9	2.23E-06	6.36E-02	1.03E-01	8.43E-03	1.36E-01
←(EC-4019, R305)									
52	Deborating Ion Exchanger, Purification Ion Exchanger B	-4	510	6.6	2.92E-06	8.31E-02	1.35E-01	1.10E-02	1.77E-01
53	Purification Ion Exchanger A, Fuel Pool Ion Exchanger	-4	610	3.3	1.22E-06	3.47E-02	5.63E-02	4.61E-03	7.42E-02
54	Preconcentrator Ion Exchanger B, Boric Acid Cond. Ion Exchanger B	-4	400	6.6	1.08E-07	1.39E-02	6.33E-03	1.41E-02	2.26E-01
55	Preconcentrator Ion Exchanger A, Boric Acid Cond. Ion Exchanger A	-4	400	6.6	1.08E-07	1.39E-02	6.33E-03	1.41E-02	2.26E-01
56	Letdown Heat Exchanger and Strainer	-4	450	13.2	6.70E-06	3.91E-01	4.07E-01	1.97E-01	5.36E+00
57	Blowdown Pumps A and B	-4	1,350	6.6	2.89E-11	1.94E-05	9.23E-06	1.73E-05	4.47E-04
58	Filter Flush Tank and Pump	-4	300	9.9	7.45E-08	2.12E-03	3.44E-03	2.81E-04	4.53E-03
59	Blowdown Heat Exchanger A and B	-4	1,300	3.0	5.24E-09	2.22E-04	2.77E-04	9.82E-04	2.11E-03
60	Blowdown Filters A and B	-4	600	13.2	4.96E-08	1.41E-03	2.29E-03	1.87E-04	3.02E-03
61	Blowdown Demineralizers A and B	-4	550	6.6	2.71E-08	7.70E-04	1.25E-03	1.02E-04	1.65E-03
62	Acid Storage Tank, Caustic Storage Tank and Heaters, Chemical Feed Tank and Pump	-4	-	0	CLEAN	-	-	-	-
63	Boric Acid Concentrator A	-4	900	16.5	4.85E-08	2.47E-03	2.37E-03	1.56E-03	2.51E-02
64	Boric Acid Concentrator B	-4	900	16.5	4.85E-08	2.47E-03	2.37E-03	1.56E-03	2.51E-02
65	Waste Concentrator	-4	1,100	16.5	3.97E-08	2.02E-03	1.94E-03	1.28E-03	2.06E-02
66	Pipe Penetration Area	-4	6,710	10	3.38E-07	1.43E-02	1.79E-02	5.29E-03	1.36E-01
67	Corridor	-4	16,250	1.9	4.61E-10	9.20E-05	2.13E-05	1.74E-06	2.80E-05

←(DRN 02-110, R12; 05-455, R14

→(DRN 02-110, R12)

Table 12.2-16a (Sheet 4 of 4) Revision 14 (12/05)

Reactor Auxiliary Building Room by Room C/DAC and Dose Commitment Values

→(DRN 05-455, R14)

Item	Location and / or Component	Elevation (Ft. MSL)	Ventilation Rate (CFM)	Leakage Rate (gpd)	Conc. C (uCi/cc)	C/DAC	Dose Commitment (mRem/hr Occupancy) DDE Submersion	CEDE Inhalation	CDE-Thyroid Inhalation
68	Component Cooling Water Chemical Feed Tank	+21	-	0	CLEAN	-	-	-	-
69	Boric Acid Batching Tank and Strainer	+21	630	9.9	4.84E-08	3.35E-03	2.48E-03	2.68E-03	4.31E-02
70	Waste Concentrate Storage Tank and Metering Pump	+21	400	6.6	4.37E-07	2.22E-02	2.14E-02	1.41E-02	2.26E-01
71	Vault Area	-35	7,130	3.3	1.04E-09	2.97E-05	4.82E-05	3.94E-06	6.35E-05
72	Aux Component Cooling Water Pumps A and B	-35	-	0	CLEAN	-	-	-	-
73	Refueling Water Pool Purification Pump, Sump #3 and Pumps	-35	7,130	3.3	1.04E-09	2.97E-05	4.82E-05	3.94E-06	6.35E-05
74	Blowdown Tank	-4	350	3.3	3.88E-08	2.14E-03	2.92E-05	3.57E-03	8.62E-04
	Reactor Auxiliary Building		77,000	160 (Lb/day)	8.01E-08	2.23E-03	3.68E-03	2.98E-04	4.78E-03

←(DRN 02-110, R12; 05-455, R14

Table 12.2-17 Revision 14 (12/05)

→(DRN 99-1098, R11; 03-2066, R14)

18-GROUP GAMMA-RAY SOURCE STRENGTHS PER FUEL ASSEMBLY
3 DAYS AFTER SHUTDOWN

←(DRN 03-2066, R14)

E Mean (Mev)	Photons/sec

Security-Related Information
Text Withheld Under 10 CFR 2.390

← (DRN 99-1098, R11)

→(LBDCR 14-007, R308)

Table 12.2-18 (Sheet 1 of 2) Revision 309 (06/16)

Waterford 3 Original Steam Generator Storage Facility Radioactive Isotope Inventory

Reactor Head

Isotope	Smear (Ci)	Head (Ci)	Isotope	Smear (Ci)	Head (Ci)

Security-Related Information
Text Withheld Under 10 CFR 2.390

←(LBDCR 14-007, R308)
→(LBDCR 15-018, R309)

Note:

Quantity of Isotopes identified in Table 12.2-18 above are estimates only and not intended to be used for calculating actual quantity of material stored.
← (LBDCR 15-018, R309)

→(LBDCR 14-007, R308)

Table 12.2-18 (Sheet 2 of 2) Revision 309 (06/16)

Waterford 3 Original Steam Generator Storage Facility Radioactive Isotope Inventory

Old Steam Generators

Isotope	Calculated Activity [Ci]	Isotope	Calculated Activity [Ci]	Isotope	Calculated Activity [Ci]

Security-Related Information
Text Withheld Under 10 CFR 2.390

←(LBDCR 14-007, R308)
→(LBDCR 15-018, R309)

Note:

Quantity of Isotopes identified in Table 12.2-18 above are estimates only and not intended to be used for calculating actual quantity of material stored.
← (LBDCR 15-018, R309)