12.3A TMI SHIELDING STUDY

12.3A.1 INTRODUCTION

In July 1979, NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report) was published. One requirement of NUREG-0578 was a design review of plant shielding of areas necessary for plant operation. That review, as clarified by NUREG-0737, is presented here.

The design basis of this review includes the effects of large and small break LOCAs. The dose rates and integrated doses were calculated by standard shielding techniques, utilizing the actual geometries and expected source strengths as a function of time and space. All equipment, piping, and areas which could contain radioactivity under accident conditions were considered (systems considered are listed in Table 12.3A-8).

→(DRN 05-1249, R14)

Dose rate calculations were performed for those areas containing radioactive sources and the surrounding areas affected by these sources to ensure that vital areas (those necessary for post-accident operations) are accessible and habitable for the time necessary to perform the required tasks in that area. This was done by placing piping diagram transparencies over general arrangement drawings to determine the source term for each area. The original design calculations were done by 3 dimensional point kernel technique which is contained in the computer program SPAN-4[(1)] . This program has rectangular, cylindrical and spherical geometries available- thus, all the source and shielding shapes were accurately described in this analysis. The SPAN-4 program contains libraries of material densities and cross section, energy structures, buildup factors, flux to dose conversion factors and quadrature weights. The library data can be changed or added to (on a temporary basis) by the user through input data on cards. This calculational model is used for the direct dose rates from the contained sources. Other computer codes have been utilized for updated calculations of dose rates at various points of interest. These codes have been verified and validated throughout industry and by WSES to calculate both accurate and conservative radiation dose rate and shielding results.

←(DRN 05-1249, R14)

→(DRN 99-2362, R11; 03-2066, R14; 05-1249, R14; EC-5000082374, R301)

The original TMI design basis dose analyses were reviewed for impact based on EPU conditions. A comparison of the average or expected radioisotope activities based on ANSI N237 (FSAR design basis) and ANSI 18.1 (EPU basis) and an evaluation of the change in flux-to-dose conversion factors between ANSI 6.1.1 1977 and ANSI 6.1.1 1991 indicate that normal operation and anticipated occurrences dose rates due to EPU are bounded by the current FSAR design basis dose rates.

←(DRN 99-2362, R11; 03-2066, R14; 05-1249, R14; EC-5000082374, R301)

The dose rates calculated for this analysis used the worst conditions as their basis; thus, for practical operating conditions the Health Physics personnel will need to take dose rate measurements to guide personnel in their trips about the plant. Some areas may be found accessible which are indicated as inaccessible in this Appendix.

12.3A.2 SOURCE TERMS

→(DRN 99-2362, R11; 05-144, R14)

The source terms used for areas other than the control room and the diesel generator rooms are consistent with the requirements of NUREG-0737 and Regulatory Guide 1.4. The core inventories were taken from Table 4.3-1 of the Combustion Engineering document SYS80-PE-RG, Revision 4, <u>Radiation Design Guide</u> issued July 12, 1979 and are listed in Table 12.3A-1. The source data is tabulated in Table 12.3A-2 through 12.3A-7.

←(DRN 99-2362, R11)

The source term used for the main control room and the emergency diesel generator rooms are consistent with the requirements of Regulatory Guide 1.183 and are listed in Table 12.2-12.

←(DRN 05-144, R14)

The source strengths used in this analysis were divided into three categories; liquid, gaseous and plateout. The source strengths (γ/cc-sec) are calculated for nine energy groups, and 11 times following the accident.

→(DRN 99-2362, R11; 03-2066, R14)

For the liquid source, two conditions were studied to ensure that calculated dose rates are conservative. The first condition is one in which a large break LOCA occurs in the primary loop, and during the injection phase a large quantity of the injection water spills from the primary loop to the floor of the containment and the SIS sump. The injection water comes from four safety injection tanks, the boric acid make-up tanks and the refueling water storage pool. The reactor coolant water volume is 11,100 cubic feet, and the minimum combined water 6 volume of reactor coolant and injection water is 69,525 cubic feet ($1.97x10^9$ cc). The minimum time for the injection phase to be completed (and the beginning of the recirculation phase) is 20 minutes. The total activity in the SIS sump water and the primary loop water is specified in Regulatory Guide 1.4 and NUREG-0737 as:

←(DRN 99-2362, R11; 03-2066, R14)

a) 100% of the core inventories for the noble gases,
b) 50% of the core inventories for the halogens, and
c) 1% of the core inventories for the other isotopes

None of this activity leaves the containment until the recirculation phase is started. In the first few minutes of recirculation, it is assumed that this activity will be uniformly mixed in the total water volume of the primary loop and the SIS sump.

The second condition for the water volume activity concentration is one in which a small break occurs in the primary loop so that only a small amount of water is injected into the primary loop. It is also assumed that all the core inventories listed above in the first assumption are released into this smaller volume of water in the primary loop. This ratio of the two volumes is 6.264; thus, the source strengths in the second condition will be 6.264 times greater than in the first. The greater water/activity concentrations resulting from the second condition were used in the analysis of the hot leg sampling lines and the shutdown cooling system. They were not used for the Safety Injection or Containment Spray Systems because these systems are not aligned to their recirculation mode for a small break LOCA.

→(DRN 03-2066, R14)

Waterford 3 implemented Alternative Source Term (AST) dose methodology for use in Chapter 15 analyses. Continued use of the older dose methodology based on Regulatory Guide 1.4 and NRC document TID-14844 is allowed by the NRC. Current TMI Action Plan doses and Equipment Qualification doses remain bounding and are not revised using AST with the exception of the CVAS and SBVS charcoal filter trains. The revised CVAS and SBVS filter train analyses use AST methodologies and assumptions and are similar to the LBLOCA dose analyses documented in Section 15.6.3.3.

←(DRN 03-2066, R14)

For the small break scenario, Waterford 3 assumed that Shutdown Cooling is initiated six hours into the LOCA. Cooldown is assumed to be initiated two hours after the LOCA; the cooldown is assumed to take four hours, corresponding to a 50°F/hour cooldown rate from an initial RCS temperature of 550°F to SDC entry conditions at 350°F.

The source strengths for the liquid source are summarized in Tables 12.3A-2, 3, 4 and 5 for the SIS sump water total noble gases, halogens, and all other inventories respectively. Table 12.3A-2 is the total source strength and Tables 12.A-3, 4 and 5 are the ratios relative to the total.

→(DRN 03-2066, R14)

The gaseous source model used 100 percent of the core noble gases and 25 percent of the core halogens as an instantaneous release into the containment vessel. The free volume of the containment, vessel has been conservatively assumed to be 2,500,000 cubic feet ($7.08x10^9$ cc), and it is assumed that the activity is uniformly distributed within this volume. The source strengths for the gaseous source are summarized in Table 12.3A-6.

←(DRN 03-2066, R14)

The plateout source model assumed 25 percent of the core iodine and one percent of the core rubidium and cesium instantaneously deposited on the internal surface areas of the containment. In addition to these source terms, 100 percent of the core inventories of gaseous Krypton-88, Krypton-89 and Xenon-138 are assumed in the containment and their daughter product decay to Rubidium-88, Rubidium-79 and Cesium-138, respectively are included as plateout isotopes. The internal surface area has been conservatively estimated to be 434,000 square feet ($4.03x1O^8$ cm^2), and the activity was assumed to be uniformly distributed on that surface area. The source strengths are summarized in Table 12.3A-7.

The Reactor Building contains a very large source of radiation following an accident and its effects can be noted at great distances. The dose rates at selected points within the plant site are shown on Figure 12.3A-1 and are a result of the gaseous atmosphere and plateout from within the containment. As can be seen from the figure, the five minute dose rates are high, but by 24 hours the dose rates are reduced to 0.4 percent of the five minute dose rates. This rapid decline in the dose rates is a result of the short half lives of most of the noble gases. Sixty five percent of the one year integrated dose from the containment source is received in the first six hours and 90 percent in the first day.

12.3A.3 ANALYZED SYSTEMS AND AREA DOSE RATES

The systems analyzed as sources of radiation are those listed in Table 12.3A-8. All other systems such as the Chemical and Volume Control System (CVCS), Waste Management, and Boron Management are not necessary for post-LOCA operation. Degassing of the Reactor Coolant System (RCS) will be done using the RCS vent system rather than the CVCS system. The RCS vents discharge directly to the containment atmosphere or the quench tank (see Subsection 5.4.15).

The following is a description of assumptions made in computing the maximum possible dose rates in each vital area, problems associated with unacceptable dose rates, and projected solutions to those problems. The occupancy requirements and dose levels for each vital area are summarized in Table 12.3A-9.

→(EC-5000082374, R301)

Dose rates are acceptable if they meet the requirements of NUREG 0737, i.e., less than 15 mrem/hour (averaged over 30 days) for areas requiring continuous occupancy, and GDC-19 requirements (less than 5 rem for the duration of the accident) for areas requiring irregular, not continuous, occupancy.

←(EC-5000082374, R301)

12.3A.3.1 <u>Control Room</u>

→(DRN 03-2066, R14)

The Control Room and Technical Support Center (item I on Table 12.3A-9) are located at elevation +46.00' in the RAB (Figure 12.3A-8). Continuous occupancy is required in the control room for the duration of the accident. The bounding event with respect to control room dose are DBA events resulting in potential ADV releases based on the proximity of the east atmospheric dump valve release location and the east control room air intake. The bounding event with respect to shine doses from filter trains, containment and airborne sources is the large break LOCA analysis documented in Section 15.6.3.3. There are six sources of radiation considered for this area. The individual contributors to the dose in the control room from each source for a large break LOCA are:

←(DRN 03-2066, R14)

→(DRN 03-2066, R14)

		Approximate % Dose Contribution
1)	Shield Building Ventilation System (SBVS)	negligible
2)	Control Room Air Conditioning System (CRACS)	1.1
3)	Inhalation / Submersion	49.0
4)	Outside Cloud	0.6
5)	Containment Gas and Plateout	0.2
6)	Controlled Ventilation Area System (CVAS)	49.1

→(EC-5000082374, R301) [items 2–6]

←(EC-5000082374, R301)

The SBVS filters are shielded by 2.5 ft. thick concrete walls on the west side and a portion of the north side of the control room. The CVAS filters are shielded by 2.5 ft. thick concrete walls on the west side of the control room. The CRACS emergency filtration unit filters are shielded by a 1.0 ft. thick concrete wall on the north side of the control room. Additional shielding was provided to shield against shine from the CRACS Emergency Filtration Units. The dose from inhalation cannot be reduced completely; however, air tight air locks limit leakage into the area and the emergency filters remove the iodine from the control room air. The control room is shielded from the cloud outside the containment by the 2.0 ft. thick roof slab above the control room and the internal walls separating the Reactor Building and the RAB.

←(DRN 03-2066, R14)

Therefore, with the additional shielding provided on the north side of the control room, the dose rates in the control room are acceptable.

12.3A.3.2 Valve Operation

In order to operate the Shutdown Cooling System, valves SI-125A, SI-125B, SI-412A, SI-412B (providing a flow path between the LPSI pump discharge lines and the shutdown heat exchangers) and SI-135A and SI-135B (provides a flow path from the LPSI pump discharge lines to the LPSI pump suction lines in order to reduce thermal shock) must be opened (items 2, 3 and 4 on Table 12.3A-9). As a result of the shielding review, these valves have been provided with motor operated actuators.

Operation of the Shutdown Cooling System also requires that valves CS-117A and CS-117B be closed manually from the RAB -15' level valve gallery to isolate the Containment Spray headers. Analysis indicates that the dose rate near these valve operators is less than or equal to 23 R/hr at six hours. The resulting dose for a 10 minute entry is less than 4 Rem. Manual operation of these valves results in radiation dose within the criteria of GDC 19 and is therefore acceptable. (Reference: Calculation OSA-RC-CALC-91-001).

→(EC-30976, R307)

Operation of the Shutdown Cooling System, after a loss of instrument air, requires that valves SI-129A and/or SI-129B be closed manually and remotely. The location of the supplemental air supplies used to close the SI-129A and SI-129B valves remotely are located in the RAB -15’ level valve gallery near the handwheel operators for valves CS-117A and CS-117B. Therefore, the dose is the same experienced to close CS-117A or CS-117B at six hours.

←(EC-30976, R307)

12.3A.3.3 Sampling Area

The dose rates in the present reactor coolant system sampling area (item 5A on Table 12.3A-9) in the southeast corner of elevation -4.00' of the RAB are unacceptable. Post-accident sampling will, therefore, be conducted in the wing area of the RAB at elevation +21.00'. Based on the maximum dose rate of 16.3 R/hr, the post-accident sampling station is accessible following an accident for sampling and maintenance for entry durations of up to 18 minutes. The expected dose rates at the post-accident sampling station are given in Table 12.3A-9, Item 5B. Refer to FSAR Subsection 9.3.8 for further detail on post-accident sampling.

12.3A.3.4 Radiochemistry Laboratory

The Radiochemistry Laboratory area (item 6 on Table 12.3A-9) at elevation -4.00' of the RAB (Figure 12.3A-5) will have post-LOCA dose rates of less than 10 mrem/hr for the first 24 hours post accident and less than 1 mrem/hr thereafter. At the elevation below the radiochemistry lab, there is highly radioactive equipment such as the shutdown heat exchangers and equipment for the Containment Spray and Safety Injection Systems. However, the radiochemistry lab is well shielded from the high activity sources by the long slant paths in the walls and floors separating the two areas.

→ (DRN 99-2362)

It is important to note that the reactor coolant system post-accident sampling will not be conducted in the -4 RAB sampling area. Therefore, the dose rates generated for the Radiochemistry Laboratory Area reflect this fact. Consequently the Radiochemistry Laboratory Area will be available for continuous post-accident use for analyzing samples. This area is accessible from the control room by way of stairway A to elevation +21.00' and then stairway B to elevation -4.00'.

← (DRN 99-2362)

12.3A.3.5 Electrical Equipment Area

The electrical equipment area (Table 12.3A-9) on the east side of elevation +21.00' of the RAB (Figure 12.3A-7) contains switches which must be activated before equipment on the diesel generator manual load block may be sequenced onto the diesel generator. This is normally done between 30 minutes and one hour into the accident.

The maximum post-accident dose rate in the "B" Switchgear Room, which extends into the RAB Wing Area, is 16.3 R/hr at 20 minutes into the postulated large break LOCA. The maximum small break LOCA dose of 14.7 R/hr occurs at 6 hours into the event, the assumed time of SDC initiation. Personnel can enter the "B" Switchgear area for as long as 18 minutes without exceeding the GDC19 dose limit of 5 Rem.

Dose rates for the other Electrical Equipment areas are considerably lower. The Electrical Equipment areas other than the "B" Switchgear Room are continuously accessible.

The Electrical Equipment area is accessible from the control room via stairway A.

12.3A.3.6 Security Room

The doses in the security room (item 8 on Table 12.3A-9) are acceptable for continuous use. However, the location of the security room is considered confidential information and as such is not included here. The dose rates and accessibility to the security room is discussed in FSAR Subsection 13.6A.

→ (DRN 99-2362)

12.3A.3.7 -4 Access Point Offices

The dose rates in the present -4 Access Point offices (item 9 on Table 12.3A-9; Figure 12.3A-5) at elevation -4.00' of the RAB are unacceptable. This area is located directly above the heat exchangers for the shutdown cooling system and the containment spray system which can become highly radioactive. An additional health physics area has been established at the

← (DRN 99-2362)

southeast corner of the +7.00 foot elevation in the RAB. This health physics area will be used during post accident conditions. It is conveniently located near the sampling areas and the chemistry laboratories. Access will be required in this room on a 24 hour basis, and the dose rates are not expected to exceed 1.0 mrem/hr.

12.3A.3.8 Diesel Generator Rooms

→(DRN 05-144, R14)

The Diesel Generator rooms (item 10 on Table 12.3A-9) are located on the west side of the RAB at elevation +21.00' (Figure 12.3A-7). The diesel generators are immediately to the south of the Component Cooling Water (CCW) area, and they are shielded from the CCW area and the highly radioactive Shield Building Ventilation System (SBVS) and Controlled Ventilation Area System (CVAS) filters at elevation +46.00' by the 1.0 ft. thick ceiling slab. The slant paths and additional distance significantly reduces the dose rates in the diesel generator area so that the highest dose rate is 240 mrem/hr. Access is to these rooms for 15 minutes every eight hours for an operating parameter review. This area is accessible from the control room by stairway A and then through the electrical equipment area.

←(DRN 05-144, R14)

12.3A.3.9 Waste Management System (WMS) and Boron Management System (BMS) Control Panels

The WMS and BMS control panels (item 11 on Table 12.3A-9) are in the middle of elevation -4.00' of the RAB (Figure 12.3A-5). This area must be accessible for two to four hours to run the systems in a "cold" state prior to the introduction of radioactive fluid and an additional hour to align the systems for radioactive fluid processing. The dose rates in this area are acceptable for continuous post-accident use. This area is accessible from the control room by using stairway A to elevation +21.00' and then stairway B to elevation -4.00'.

12.3A.3.10 Heating, Ventilation and Air Conditioning (HVAC)

→(DRN 03-2066, R14)

The HVAC equipment room at elevation +46 of the RAB is not accessible following an accident, and the air intake valves are controlled by remote manual operation. The emergency operation of the equipment is discussed in Subsections 9.4.3.4 and 9.4.5.9.

←(DRN 03-2066, R14)

REFERENCES: SECTION 12.3A

1) 0. J. Wallace, "SPAN-4: A Point Kernal Computer Program for Shielding," WAPD- TM-809(L), October 1972.

2) G. Martin, Jr., D. Michlewicz and J. Thomas, "FISSION 2120: A Program for Assessing the Need for Engineered Safety Feature Grade Air Cleaning Systems in Post Accident Environments", Proceedings of the 15th DOE Nuclear Air Cleaning Conference, pp. 266 to 278, August 7-10, 1978.

→(DRN 03-2066, R14)

3) Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

←(DRN 03-2066, R14)

TABLE 12.3A-1 (Sheet 1 of 2) Revision 14 (12/05)

→(DRN 03-2066, R14)

TMI SOURCE TERM FOR
SHIELDING EVALUATION

Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 12.3A-1 (Sheet 2 of 2) Revision 14 (12/05)

→(DRN 03-2066, R14)

TMI SOURCE TERM FOR
SHIELDING EVALUATION

Nuclide	Inventory (Curies)	Nuclide	Inventory (Curies)

Security-Related Information
Text Withheld Under 10 CFR 2.390

*Numbers in parentheses denote powers of ten
**I-127 is a Stable Isotope therefore the inventory is expressed in atoms

→(DRN 03-2066, R14)
Note: This table was evaluated for EPU and the TMI source term was determined to be bounding for EPU.
←(DRN 03-2066, R14)

TABLE 12.3A-2

SIS SUMP WATER TOTAL SOURCE STRENGTHS

Includes 100% Core Noble Gases, 50% Core Halogens, 1% Core All Others Diluted in 69,525 Ft^3 Water - (Source Strengths γ/cc-sec)

Upper Mev	0.106	0.440	0.865	1.332	1.720	2.210	2.754	3.930	4.702
ENG GP #	8	7	6	5	4	3	2	1	0
Time									
5 MINS	1.69+9	6.30+9	8.13+9	6.36+9	1.55+9	1.88+9	1.02+9	1.08+8	8.99+6
20 MINS	1.69+9	4.87+9	7.09+9	5.44+9	1.24+9	1.18+9	8.59+8	2.12+7	3.42+5
1 HR	1.68+9	3.57+9	5.81+9	3.96+9	1.04+9	5.88+8	6.30+8	7.51+6	5.60+1
6 HRS	1.63+9	1.93+9	2.56+9	1.14+9	4.66+8	1.89+8	1.30+8	1.09+4	0+0
12 HRS	1.58+9	1.42+9	1.65+9	5.57+8	2.33+8	8.00+7	2.76+7	4.25+0	0+0
1 DAY	1.48+9	1.06+9	1.07+9	1.56+8	7.88+7	1.96+7	2.13+6	0+0	0+0
3 DAYS	1.14+9	7.63+8	3.41+8	3.55+6	1.12+7	1.48+5	3.46+5	0+0	0+0
1 WK	6.76+8	5.31+8	1.51+8	1.26+6	2.13+6	2.06+3	6.61+4	0+0	0+0
2 WKS	2.72+8	2.88+8	1.12+8	7.81+5	2.50+5	4.25+1	3.66+3	0+0	0+0
1 MO	3.51+7	7.40+7	7.50+7	4.33+5	1.39+5	5.96-3	4.90+0	0+0	0+0
1 YR	2.16+5	1.19+6	8.92+6	8.77+4	1.02+5	0+0	0+0	0+0	0+0

TABLE 12.3A-3

SIS SUMP WATER NOBLE GAS SOURCE STRENGTH RATIOS

Source Strength Ratios Relative To Total Source Strengths Includes 100% Core Noble Gases, 50% Core Halogens and 1% Core All Others

Upper Mev	0.106	0.440	0.865	1.332	1.720	2.210	2.754	3.930	4.702
ENG GP #	8	7	6	5	4	3	2	1	0
Time									
5 MINS	0.964	0.813	0.208	0.052	0.205	0.610	0.796	0.795	1.000
20 MINS	0.964	0.772	0.139	0.012	0.138	0.549	0.843	0.171	1.000
1 HR	0.967	0.720	0.099	0.003	0.164	0.484	0.926	0.000	1.000
6 HRS	0.967	0.528	0.046	0.000	0.150	0.429	0.994	0.000	-
12 HRS	0.968	0.354	0.017	0.000	0.068	0.249	0.976	0.000	-
1 DAY	0.968	0.156	0.004	0.000	0.010	0.056	0.722	-	-
3 DAYS	0.966	0.006	0.000	0.000	0.000	0.000	0.000	-	-
1 WK	0.959	0.000	0.000	0.000	0.000	0.000	0.000	-	-
2 WKS	0.940	0.000	0.000	0.000	0.000	0.000	0.000	-	-
1 MO	0.844	0.001	0.000	0.000	0.000	0.000	0.000	-	-
1 YR	0.000	0.000	0.000	0.000	0.000	-	-	-	-

TABLE 12.3A-4

SIS SUMP WATER HALOGEN SOURCE STRENGTH RATIOS

Source Strength Ratios Relative To Total Source Strengths Includes 100% Core Noble Gases, 50% Core Halogens and 1% Core All Others

Upper Mev	0.106	0.440	0.865	1.332	1.720	2.210	2.754	3.930	4.702
ENG GP #	8	7	6	5	4	3	2	1	0
Time									
5 MIN	0.015	0.160	0.619	0.698	0.331	0.144	0.012	0.202	0.000
20 MINS	0.015	0.689	0.794	0.410	0.206	0.010	0.823	0.000	
1 HR	0.016	0.245	0.738	0.894	0.527	0.331	0.006	0.999	0.000
6 HRS	0.016	0.411	0.695	0.929	0.754	0.570	0.000	1.000	-
12 HRS	0.016	0.562	0.630	0.867	0.759	0.750	0.000	1.000	-
1 DAY	0.016	0.734	0.535	0.664	0.592	0.942	0.000	-	-
3 DAYS	0.018	0.872	0.227	0.016	0.018	0.885	0.000	-	-
1 WK	0.021	0.877	0.062	0.000	0.000	0.003	0.000	-	-
2 WKS	0.028	0.858	0.032	0.000	0.000	0.000	0.000	-	-
1 MO	0.053	0.773	0.009	0.000	0.000	0.000	0.000	-	-
1 YR	0.000	0.000	0.000	0.000	0.000	-	-	-	-

TABLE 12.3A-5

SIS SUMP WATER ALL OTHERS SOURCE STRENGTH RATIOS

Source Strength Ratios Relative To Total Source Strengths Includes 100% Core Noble Gases, 50% Core Halogens and 1% Core All Others

Upper Mev	0.106	0.440	0.865	1.332	1.720	2.210	2.754	3.930	4.702
ENG GP #	8	7	6	5	4	3	2	1	0
Time									
5 MINS	0.021	0.027	0.173	0.250	0.464	0.246	0.192	0.003	0.000
20 MINS	0.021	0.031	0.172	0.194	0.452	0.245	0.148	0.006	0.000
1 HR	0.017	0.035	0.253	0.103	0.309	0.185	0.068	0.001	0.000
6 HRS	0.017	0.061	0.259	0.071	0.096	0.001	0.006	0.000	-
12 HRS	0.016	0.084	0.353	0.133	0.173	0.001	0.024	0.000	-
1 DAY	0.016	0.110	0.461	0.336	0.398	0.002	0.278	-	-
3 DAY	0.016	0.122	0.773	0.984	0.982	0.115	1.000	-	-
1 WK	0.020	0.123	0.938	1.000	1.000	0.997	1.000	-	-
2 WKS	0.032	0.142	0.968	1.000	1.000	1.000	1.000	-	-
1 MO	0.103	0.226	0.991	1.000	1.000	1.000	1.000	-	-
1 YR	1.000	1.000	1.000	1.000	1.000	-	-	-	-

TABLE 12.3A-6

CONTAINMENT GASEOUS TOTAL SOURCE STRENGTHS

Includes 100% Core Noble Gases and 25% Core Halogens Diluted in 2,500,000 Ft^3
Air - (Source Strengths γ/cc-sec)

Upper Mev	0.106	0.440	0.865	1.332	1.720	2.210	2.754	3.930	4.702
ENG GP #	8	7	6	5	4	3	2	1	0
Time									
5 MINS	4.61+7	1.52+8	1.32+8	9.86+7	2.61+7	4.79+7	2.72+7	2.64+6	2.50+5
20 MINS	4.61+7	1.13+8	1.09+8	8.00+7	1.97+7	2.81+7	2.29+7	3.36+5	9.50+3
1 HR	4.59+7	7.91+7	8.56+7	5.58+7	1.63+7	1.21+7	1.70+7	1.04+5	1.56+0
6 HRS	4.47+7	3.74+7	3.56+7	1.57+7	6.61+6	3.44+6	3.57+6	1.51+2	0+0
12 HRS	4.32+7	2.45+7	2.18+7	7.72+6	2.94+6	1.30+6	7.38+5	0+0	0+0
1 DAY	4.05+7	1.60+7	1.36+7	2.17+6	7.55+5	2.81+5	3.72+4	0+0	0+0
3 DAYS	3.12+7	1.04+7	3.37+6	1.49+4	5.07+3	1.80+3	0+0	0+0	0+0
1 WK	1.84+7	7.31+6	7.94+5	0+0	0+0	0+0	0+0	0+0	0+0
2 WKS	7.38+6	4.00+6	3.86+5	0+0	0+0	0+0	0+0	0+0	0+0
1 MO	9.16+5	1.01+6	9.89+4	0+0	0+0	0+0	0+0	0+0	0+0
1 YR	0+0	0+0	1.88+3	0+0	0+0	0+0	0+0	0+0	0+0

TABLE 12.3A-7

CONTAINMENT PLATEOUT TOTAL SOURCE STRENGTHS

*Includes 25% Core Iodine and 1% Core Cesium and Rubidium Plus Daughter Products Of Noble Gas Decay - (Source Strengths γ/cc-sec)

Upper Mev	0.106	0.440	0.865	1.332	1.720	2.210	2.754	3.930	4.702
ENG GP #	8	7	6	5	4	3	2	1	0
Time									
5 MINS	7.07+8	3.34+10	1.83+11	2.29+11	5.33+10	2.35+10	1.01+10	0+0	0+0
20 MINS	7.07+8	3.22+10	1.86+11	2.12+11	8.73+10	3.10+10	1.35+10	0+0	0+0
1 HR	7.05+8	2.98+10	1.52+11	1.24+11	7.13+10	2.18+10	7.33+9	0+0	0+0
6 HRS	6.92+8	2.50+10	5.84+10	2.80+10	9.48+9	3.59+9	5.60+7	0+0	0+0
12 HRS	6.78+8	2.36+10	3.76+10	1.37+10	4.66+9	1.72+9	9.97+6	0+0	0+0
1 DAY	6.50+8	2.19+10	2.42+10	3.88+9	1.31+9	4.66+8	5.15+5	0+0	0+0
3 DAYS	5.47+8	1.82+10	6.49+9	8.17+7	1.58+7	3.16+6	0+0	0+0	0+0
1 WK	3.89+8	1.29+10	1.96+9	4.60+7	6.91+6	1.50+2	0+0	0+0	0+0
2 WKS	2.14+8	7.06+9	1.23+9	3.34+7	6.87+6	0+0	0+0	0+0	0+0
1 MO	5.47+7	1.78+9	6.99+8	1.75+7	6.77+6	0+0	0+0	0+0	0+0
1 YR	0+0	0+0	3.99+8	4.24+6	4.97+6	0+0	0+0	0+0	0+0

*Uniformly distributed over the estimated 434,000 Ft^2 of surface area inside the containment.

TABLE 12.3A-8 Revision 11 (05/01)

SYSTEMS CONTAINING RADIOACTIVE MATERIAL

→ (DRN 99-2362)

System and Major Components	Location	Figure

Security-Related Information
Figure Withheld Under 10 CFR 2.390

TABLE 12.3A-9 (Sheet 1 of 2) Revision 307 (07/13)

AREAS REQUIRING ACCESSIBILITY FOLLOWING AN ACCIDENT

AREA	LOCATION	OCCUPANCY REQUIREMENTS	MAXIMUM DOSE	ACCEPTABLE	REMARKS

Security-Related Information
Figure Withheld Under 10 CFR 2.390

TABLE 12.3A 9 (Sheet 2 of 2) Revision 14 (12/05)

AREAS REQUIRING ACCESSIBILITY FOLLOWING AN ACCIDENT

AREA	LOCATION	OCCUPANCY REQUIREMENTS	MAXIMUM DOSE	ACCEPTABLE	REMARKS

Security-Related Information
Figure Withheld Under 10 CFR 2.390