12B Spent Fuel Pool Geometry and Dose Rate Calculation

This Appendix provides ~~site-specific~~ information regarding the ~~Spent Fuel Pool (SFP) geometry and~~ methodology used to generate Spent Fuel Pool (SFP) radiation sources and dose rate assessments. ~~This Appendix supplements the SFP radiation source information in Section 12.2, specifically Tables 12.2-5a and 12.2-5b.~~See FSAR Section 12.2, Tables 12.2-5a, 12.2-5b, and 12.2-5c; and FSAR Section 12.3, Figures 12.3-5 through 12.3-10.

~~The SFP geometry is provided in FSAR Section 12.2, Tables 12.2-5a and 12.2-5b; and FSAR Section 12.3, Figures 12.3-6 and 12.3-10. The SFP storage rack has the capacity for 2365 BWR fuel assemblies. The size and configuration of the storage rack are given in Figure 12B-1, which shows an irregular rectangular cross-section area extended to 10.7 m x 9.0 m. The fuel storage rack is situated at the lowest level of the SFP, where the total height of water is 11.6 m.~~

12B.1 Core Source Determination

The STP 3 & 4 core rated power is 3926 MWt. The core load consists of 872 fuel assemblies, with an assumed average enrichment of 4.2%. The reload batch size is 320 fuel assemblies. The equilibrium core is assumed to operate on a two-year fuel cycle, with approximately 704 effective-full-power-days (EFPD) of operation for each cycle. The core is composed of three batches of fuel, with varying residence time and burnup at the end of cycle (EOC). The latest batch of fresh fuel assemblies will be placed at high power density locations during operation and will average 23 GWD/T at EOC, 1.27 times the core average (18 GWD/t). The second oldest batch will have resided in the core for one cycle prior to the current cycle and will have a cycle exposure of 20 GWD/t (1.14 times average) and an average accumulated exposure of 43 GWD/T at EOC. The oldest batch will have a cycle exposure of 8 GWD/t (0.43 times average) and an average accumulated exposure of ~~resided in the core for two cycles prior to the current cycle and on average accumulated~~ 51 GWD/T at EOC. Radiation sources of the equilibrium core, as well as the discharge batch, are calculated using the code ORIGEN-S of the SCALE system (Reference 12B.6-1).

In addition to fuel elements, the mass of supporting structural materials is also included in the ORIGEN-S calculations. Therefore the calculated radiation source includes contributions from activation and corrosion products as well as fission products and actinides. In addition, a safety margin of 10% is added to the calculated radiation source to bound the uncertainties in structural material specification. The resulting equilibrium core gamma source is shown in Table 12B-1.

12B.2 SFP Fuel Source Determination

The SFP radiation source is calculated using ~~the maximum capacity of the storage rack~~five batches of spent fuel discharged at two-year intervals plus a full core discharge. ~~Under normal operating conditions, the maximum number of fuel assemblies accommodated by the storage rack is 1493, leaving enough room for a full core offload in the event of unexpected operating conditions.~~ The ~~maximum~~ SFP radiation source ~~when the storage rack is filled to its capacity will~~ includes the full equilibrium core radiation source conservatively assumed to be at one-day post shutdown plus the sources of existing fuel assemblies in the SFP. The existing fuel assemblies in the SFP are assumed to be the discharged batches from an equilibrium core at decay times of two-years, four-years, six-years, eight-years, and ten-years. Calculation of these sources has been described in 12B.1 above. Similar to the core radiation source, an extra 10% safety margin is added to the calculated results for conservatism. The resulting SFP peak gamma source and peak neutron source are given in Tables 12B 2 and 12B-3, respectively. The major radionuclides that provide the peak SFP source are listed in Table 12B-4.

12B.3 SFP Dose Rate Assessment

The SFP radiation source is used for the shielding design and analysis to ensure radiation levels at neighboring areas surrounding the SFP meet the design criteria. A simplified model of a rectangular parallelepiped of (8.1 m x 8.1 m x 11.2 m) is assumed to represent the SFP. The SFP peak radiation source is assumed to be homogenized over the bottom 3.8 m of the SFP to represent the active fuel length of a typical BWR fuel assembly. 3.8 meters was assumed to represent the height of the active fuel region, since this height results in the minimum required shielding of water, 11.2 m – 3.8 m = 7.4 m (see Table 12.2-5c). The~~re are 2 m thick~~ concrete walls and floor surrounding the SFP ~~on all sides. The floor of the SFP is~~are 2 meters thick ~~as well~~(see Table 12.2-5c). The water shielding and distance between the spent fuel racks and the surrounding walls in the pool are not considered. (See Figure 12B-1). The dose rate calculations are carried out with the point-kernel shielding code QAD-CGGP-A (Reference 12B.6-2). The calculated dose rate results indicate all the areas surrounding the SFP meet the reactor building (R/B) radiation zone limits. Figure 12B-2 shows the dose rate profile for a set of detectors along the centerline of the SFP perpendicular to the water surface. The ~~D~~dose rate at the water surface is approximately 0.001 μrem/hr (1E-5 μSv/hr). The dose rates assessed are due to fuel assemblies in the SFP alone. Contributions from contaminants in the SFP water are not included.

12B.4 Single Bundle Refueling Operator Dose Rate Assessment

The maximum calculated radiation dose rate to the refueling operator from a single raised fuel assembly is calculated as follows:

As discussed in 12B.1 above, the core is rated at 3926 MWt. It consists of 872 fuel bundles, 320 of which are installed as new assemblies each refueling. 3926MWt / 872 bundles equals a core average of 4.502MWt per bundle. This average, however, consists of once, twice, and thrice burned bundles.

Similar to the SFP dose rate assessment, QAD-CGGP-A is used for the single bundle dose rate calculations. Three separate calculations are performed to address the contributions from (1) the high enrichment lower active fuel region, (2) the low enrichment (natural uranium) upper active fuel region (top node) where the power density is lower than the core average, and (3) the upper non-active fuel assembly plenum (referred to as the handle)~~the active fuel region, the top fuel node where the power density is lower than the core average, and the assembly handle~~ where it is assumed that majority of the activation products accumulate. The radiation source input for each of these components and the fuel assembly dimensions in the calculation model are presented in Table 12B-5 (the values in this table do not include an added safety margin).

DCD 9.1.4.1 requires that a fully retracted fuel grapple must maintain 2591 mm (8.5 ft) of water shielding over fuel. Therefore, the model assumes that the top node is just below the 8.5 feet of shielding water, and the handle element is just above that level. The movement of a raised fuel assembly from a once burned batch results in the highest dose rate to a worker on the refueling platform during movement of the assembly from the reactor core to the spent fuel pool.

The core design indicates that the average burnup of the once-burned batch is approximately 23 GWd/MTU, or 5.703 MWt/bundle. This value is 1.27 times the average per bundle across the entire core, as stated above. This factor would have to be increased by 10% to reflect a peak bundle of 1.4 times the core average (see DCD Table 1.3-1). However, the DCD radial peaking factor of 1.4 is for an 18-month cycle. To account for the 24-month cycle assumed in this assessment, a radial peaking factor for a 24-month core design of 1.65 is typical. Consequently, a factor of 30% was applied to the 1.27 times core average value to reflect the 1.65 peaking factor. ~~To obtain the highest activity bundle for determining the refueling source term, an additional 30% was added to the calculated dose rate from the once-burned batch of 1.27 times the core average. The net effect is equivalent to using a bundle source of 1.65 times the core average for this highest activity fuel assembly.~~ The dose rate is bounding for any assembly that will be moved by the refueling operator. (The calculated gamma source of a single bundle with one-cycle residency is compared with that of three-cycle residency in Table 12B-6, which further confirms that peak source has been chosen for the refueling dose rate assessment. The values in this table do not include an added safety margin.)

It is assumed that the fuel assembly is lifted to a height of 8.5 ft (2.6 m) below the pool water surface with the operator on the refueling machine trolley platform at a minimum of 8.8 ft (2.7 m) above the water surface. The resulting peak dose rate at 8.8 ft (2.7 m) above the water surface is approximately 1.2 mrem/hr, located at a radial distance from the fuel assembly of approximately 140 cm (4.6 ft) as shown in Figure 12B-3. Even when the maximum fuel pool water source (see Section 12B.5 below) is added, the dose rate remains below 2.5 mrem/hr. For an operator standing on the trolley platform, the dose rate will be less than that shown in Figure 12B-3; therefore the design criteria of ANSI/ANS-57.1-1992 has been satisfied.

12B.5 Spent Fuel Pool Radionuclides and Dose

The Fuel Pool Cooling and Cleanup (FPC) system described in FSAR Subsections 9.1.3 and 12.3.1.4.3 maintains the SFP water at a low radioactive nuclide level. In support of this statement, representative data from an operating ABWR plant is presented. Measurements at 1.2 m above the refueling floor indicate a maximum of 0.007 mSv/hr (0.7 mrem/hr) during plant outages with 1) fuel assemblies fully seated in the storage racks, and 2) maximum levels of measured radionuclides in the SFP water. This maximum dose rate is measured at 1.2 m (3.9 ft) above the refueling floor. During routine operations, the dose rate is expected to be less for normal operation of the FPC system. Also, the dose rate is considerably less at the operating trolley platform due to the increased distance from 1.2 m (3.9 ft) to 2.7 m (8.8 ft) and attenuation through the refueling machine lower structure and platform. A listing of SFP water radionuclides for a representative ABWR is summarized in Table 12B-7.

12B.6 References

12B.6-1 NUREG/CR-0200, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Rev 7, May 2004.

12B.6-2 CCC-645, "QAD-CGGP-A: Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor," Oak Ridge National Laboratory, December 1995.

12B.6-3 CN-REA-10-53, "STP Units 3 & 4 ABWR Spent Fuel Pool Radiation Source", Revision 1, Westinghouse Electric Company LLC, December 15, 2010.

12B.6-4 CN-REA-10-64, "STP Units 3 & 4 ABWR Spent Fuel Dose rates", Revision 0, Westinghouse Electric Company LLC, October 15, 2010.

12B.6-5 CN-REA-10-67, "Dose rate Evaluation from a Single ABWR Fuel Bundle in STP Units 3 & 4 Spent Fuel Pool", Revision 0, Westinghouse Electric Company LLC, October 10, 2010.

12B.6-6 LTR-ABWR-LIC-11-001, "Scaling of Dose rate for a Single Fuel Bundle Dose to Worker on Refueling Platform - STP 3&4", March 31, 2011.

Table 12B-1 Gamma Source* of an Equilibrium Core vs. Time Post-Shutdown

Energy Range (MeV)	Gamma Source (MeV/s-MW)	
	1-Day	30-Day
0.02 - 0.035	1.4E+14	2.7E+13
0.035 - 0.05	1.3E+14	3.2E+13
0.05 - 0.075	1.1E+14	2.6E+13
0.075 - 0.125	1.6E+15	5.5E+13
0.125 - 0.175	4.5E+14	1.2E+14
0.175 - 0.25	1.0E+15	3.6E+13
0.25 - 0.40	1.9E+15	1.4E+14
0.40 - 0.90	1.1E+16	3.9E+15
0.90 - 1.35	1.5E+15	2.1E+14
1.35 - 1.80	3.4E+15	6.9E+14
1.80 - 2.20	2.5E+14	5.3E+13
2.20 - 2.60	2.5E+14	5.1E+13
2.60 - 3.00	4.9E+12	1.1E+12
3.00 - 3.50	1.9E+12	4.5E+11
3.50 - 4.00	1.3E+10	3.3E+07
4.00 - 4.50	2.1E+09	2.5E+06
4.50 - 5.00	1.8E+10	1.6E+06
5.00 - 10.00	1.3E+08	2.8E+06
Total	2.2E+16	5.4E+15

* Based on rated power 3926 MWt and approximately 704 effective-full-power-days (EFPD) of operation each cycle. 10% safety margin added to ORIGEN-S results.

Table 12B-2 SFP Peak Gamma Source (MeV/s) vs. Time Post-Shutdown

Energy Group	Energy Range (MeV)	1-Day	1-Year	2-Year
1	0.02 - 0.035	5.5E+17	3.9E+16	2.5E+16
2	0.035 - 0.05	5.5E+17	4.1E+16	2.3E+16
3	0.05 - 0.075	4.5E+17	4.1E+16	2.4E+16
4	0.075 - 0.125	6.2E+18	8.5E+16	5.1E+16
5	0.125 - 0.175	1.8E+18	8.3E+16	4.1E+16
6	0.175 - 0.25	3.9E+18	5.6E+16	3.2E+16
7	0.25 - 0.40	7.4E+18	1.2E+17	6.8E+16
8	0.40 - 0.90	4.5E+19	3.6E+18	2.5E+18
9	0.90 - 1.35	6.0E+18	2.1E+17	1.5E+17
10	1.35 - 1.80	1.3E+19	8.3E+16	5.1E+16
11	1.80 - 2.20	9.9E+17	4.7E+16	2.0E+16
12	2.20 - 2.60	9.7E+17	4.1E+15	2.0E+15
13	2.60 - 3.00	1.9E+16	7.3E+14	3.6E+14
14	3.00 - 3.50	7.6E+15	1.0E+14	5.1E+13
15	3.50 - 4.00	5.2E+13	1.1E+11	7.0E+10
16	4.00 - 4.50	8.4E+12	2.4E+10	2.3E+10
17	4.50 - 5.00	6.9E+13	1.6E+10	1.5E+10
18	5.00 - 10.00	5.4E+11	2.7E+10	2.5E+10
Total		8.8E+19	4.4E+18	3.0E+18

Note: The data represents one full core offload plus existing pool batches. 10% safety margin is added.

Table 12B-3 SFP Peak Neutron Source

Energy Group	Neutron Energy (KeV)	(n/s) 1-Day Post-Shutdown
1	1.0E-08 - 1.0E-05	1.0E-02
2	1.0E-05 - 3.0E-05	8.0E-03
3	3.0E-05 - 5.0E-05	1.4E-00
4	5.0E-05 - 1.0E-04	1.2E-00
5	1.0E-04 - 2.25E-04	9.5E+00
6	2.25E-04 - 3.25E-04	8.7E+00
7	3.25E-04 - 4.0E-04	7.9E+00
8	4.0E-04 - 8.0E-04	5.1E+01
9	8.0E-04 - 1.0E-03	2.7E+01
10	1.0E-03 - 1.13E-03	2.6E+01
11	1.13E-03 - 1.3E-03	3.1E+01
12	1.3E-03 - 1.77E-03	9.7E+01
13	1.77E-03 - 3.05E-03	3.3E+02
14	3.05E-03 - 0.01	2.9E+03
15	0.01 - 0.03	1.5E+04
16	0.03 - 0.1	9.3E+04
17	0.1 - 0.55	1.3E+06
18	0.55 - 3.0	1.7E+07
19	3.0 - 17.0	2.3E+08
20	17.0 - 100.0	3.2E+09
21	100 - 400	2.1E+10
22	400 - 900	4.7E+10
23	900 - 1400	4.7E+10
24	1400 - 1850	3.8E+10
25	1850 - 3000	7.2E+10
26	3000 - 6430	6.5E+10
27	6430 - 2.0E+04	5.9E+09

Note: The data represents one full core offload plus existing pool batches. 10% safety margin is added.

Table 12B-4 Peak Source Radioisotopes in the Spent Fuel Assemblies

Isotopes	Curies	Isotopes	Curies
I-131	1.07E+08	Sr-92	3.25E+05
I-132	1.36E+08	Y-91	1.46E+08
I-133	1.09E+08	Y-91M	1.56E+07
I-134	6.20E+00	Y-92	4.81E+06
I-135	1.79E+07	Y-93	3.35E+07
Total I	**3.71E+08**	Zr-93	3.78E+02
		Zr-95	2.02E+08
Na-24	3.19E+03	Nb-95M	2.24E+06
P-32	3.16E+04	Nb-95	1.99E+08
Cr-51	1.76E+07	Mo-99	1.67E+08
Mn-54	7.62E+05	Tc-99M	1.60E+08
Mn-56	6.13E+04	Tc-99	2.61E+03
Fe-55	6.38E+06	Ru-103	1.75E+08
Fe-59	3.88E+05	Rh-103M	1.75E+08
Co-58	9.22E+05	Ru-106	6.63E+07
Co-60	1.54E+05	Rh-106	6.63E+07
Ni-63	1.71E+05	Ag-110M	3.50E+05
Cu-64	1.59E+04	Ag-110	4.77E+03
Zn-65	3.70E+01	Te-129M	5.68E+06
Sr-89	1.11E+08	Te-129	5.21E+06
Sr-90	1.51E+07	Te-131M	1.44E+07
Y-90	1.55E+07	Te-131	2.89E+06
Sr-91	2.46E+07	Te-132	1.32E+08
Total	**1.93E+08**	Ba-137M	1.89E+07
		Ba-140	1.93E+08
Kr-83m	6.41E+04	La-140	2.06E+08
Kr-85m	7.52E+05	Ce-141	1.90E+08
Kr-85	1.89E+06	Ce-144	1.61E+08
Kr-87	1.27E+02	Pr-144M	2.25E+06
Kr-88	2.30E+05	Pr-144	1.61E+08
Total Kr	**2.94E+06**	W-187	1.91E+05
		Np-239	1.65E+09
Xe-131m	1.25E+06	Pu-239	5.03E+04
Xe-133m	6.63E+06	**Total**	**4.16E+09**
Xe-133	2.28E+08		
Xe-135m	2.92E+06	Cs-134	2.65E+07
Xe-135	5.84E+07	Cs-135	8.82E+01
Total Xe	**2.98E+08**	Cs-136	5.21E+06
Noble Gas Totals	**3.00E+08**	Cs-137	2.01E+07
		Total Cs	**5.18E+07**

Note: The data represent one full core offload plus existing pool batches. 10% safety margin is added.

Table 12B-5 Single Bundle Source for Dose Assessment (Photons/s)

Energy Group	Energy Range (MeV)	Enriched Fuel Region	Top Node	Handle
1	0.02 - 0.035	2.5E+16	1.9E+14	6.0E+13
2	0.035 - 0.05	1.5E+16	1.1E+14	7.4E+12
3	0.05 - 0.075	8.4E+15	7.0E+13	8.7E+12
4	0.075 - 0.125	7.0E+16	1.1E+15	7.4E+12
5	0.125 - 0.175	1.6E+16	1.1E+14	2.2E+13
6	0.175 - 0.25	2.2E+16	2.7E+14	3.4E+12
7	0.25 - 0.40	2.8E+16	3.2E+14	3.5E+13
8	0.40 - 0.90	8.7E+16	6.4E+14	4.8E+14
9	0.90 - 1.35	6.0E+15	4.2E+13	1.6E+13
10	1.35 - 1.80	1.1E+16	8.0E+13	1.6E+12
11	1.80 - 2.20	4.9E+14	3.4E+12	4.2E+11
12	2.20 - 2.60	5.4E+14	3.7E+12	1.2E+10
13	2.60 - 3.00	8.9E+12	6.6E+10	5.5E+10
14	3.00 - 3.50	3.2E+12	2.2E+10	1.6E+09
15	3.50 - 4.00	2.1E+10	1.3E+08	2.5E+07
16	4.00 - 4.50	3.1E+09	1.7E+07	1.7E+04
17	4.50 - 5.00	2.3E+10	1.3E+08	1.3E+05
18	5.00 - 10.00	1.1E+08	6.0E+05	6.0E+02
Total		2.9E+17	2.9E+15	6.4E+14
Source Height (cm)*		366	15	15

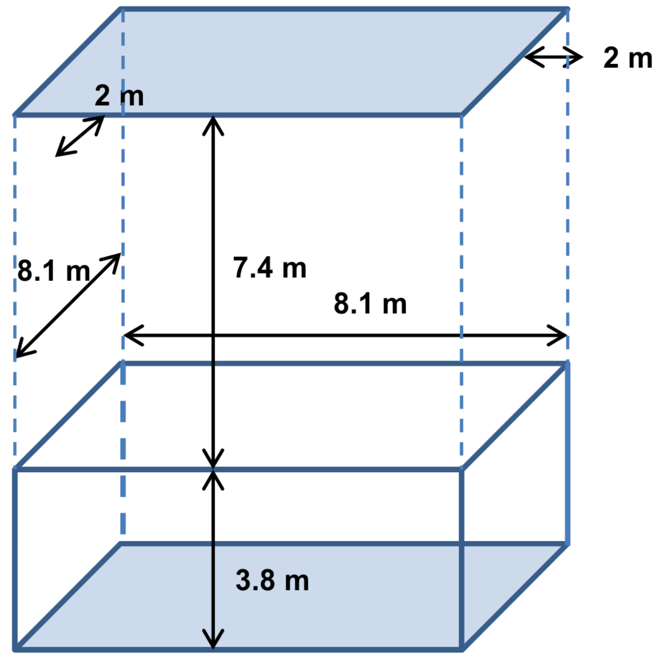
* Bundle cross-section is 15.24 cm x 15.24 cm (6 in x 6 in).

Table 12B-6 Single Bundle Gamma Source (MeV/s) Comparison

Decay	Once Burned	Thrice Burned
1 Day	1.1E+17	5.8E+16
30 Days	2.6E+16	1.5E+16
1 Year	2.1E+15	3.6E+15
2 Years	1.0E+15	2.4E+15
3 Years	7.0E+14	1.8E+15

Table 12B-7 SFP Water Radionuclides

Nuclide	Bq/cm^3	Nuclide	Bq/cm^3
Cr-51	9.4E-02	Nb-95	8.1E-03
Mn-54	9.7E-02	Cs-134	1.4E-02
Mn-56	3.6E-02	Cs-137	7.7E-03
Co-58	5.1E-01	Sb-124	5.1E-01
Co-60	1.3E+00	Sb-125	3.4E-02
Cu-64	1.0E-01		



8.1m x 8.1m x 3.8m Active Fuel Volume
8.1m x 8.1m x 7.4m Water Shielding Volume
2.0m Thick walls and Floor

Figure 12B-1 ~~SFP Layout and Dimensions~~Spent Fuel Pool Analytical Model

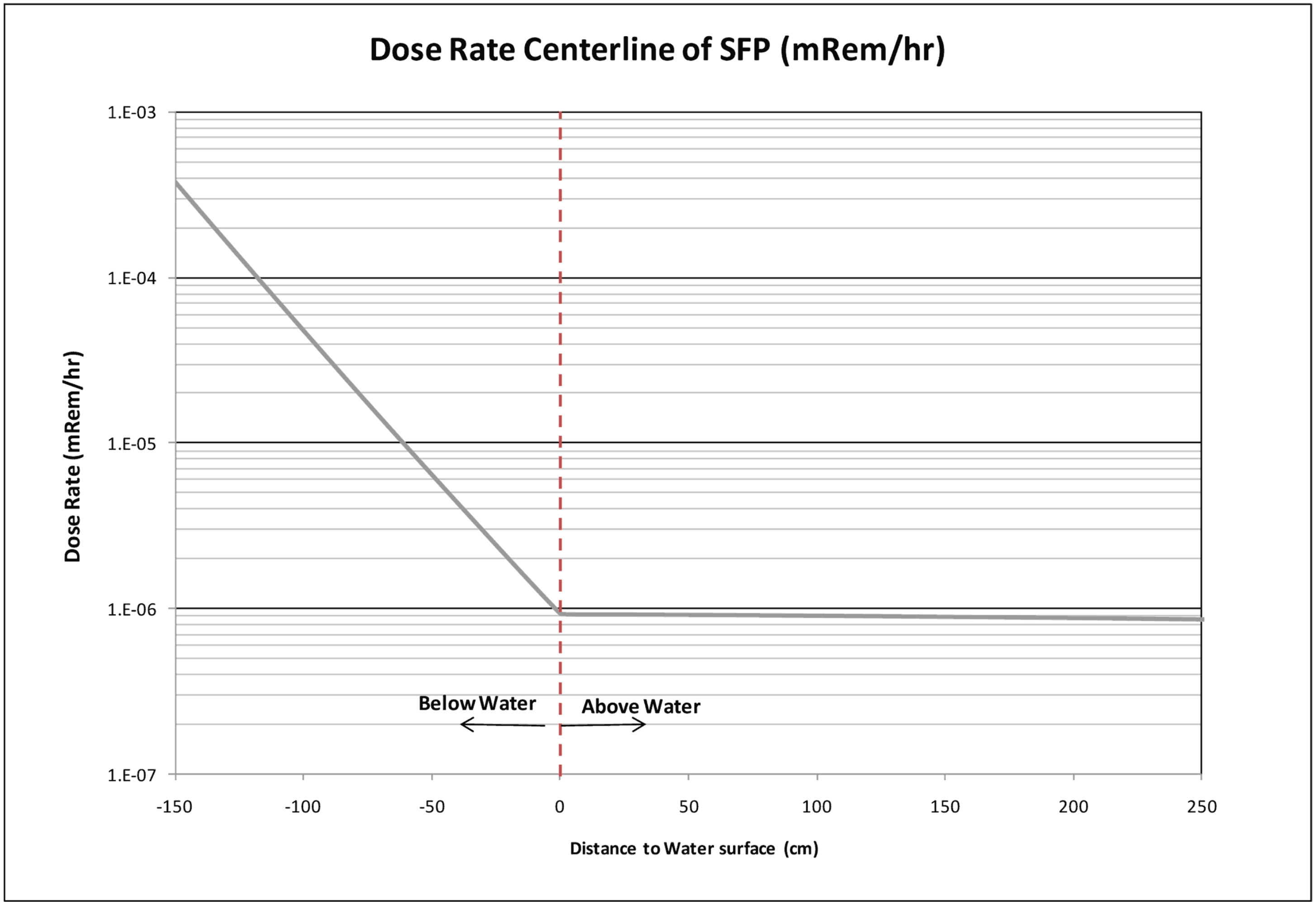


Figure 12B-2 SFP Centerline Dose Rate

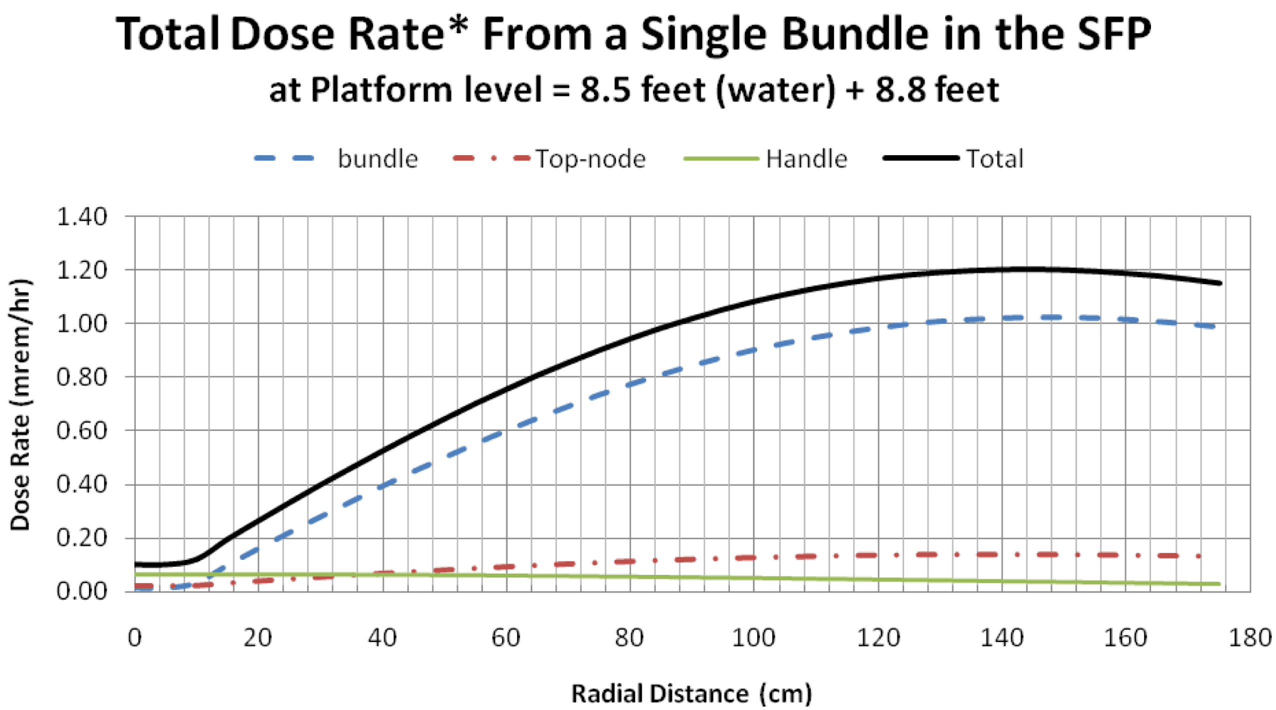


Figure 12B-3 Maximum Single Bundle Dose Rate at Refueling Platform