8

Radiological Dose Assessment

Brookhaven National Laboratory (BNL) routinely evaluates site operations to ensure that the radiological dose impact to the members of the public, BNL workers, and the environment is "As Low As Reasonably Achievable (ALARA)." All scientific and operational processes and activities that can in any way impact the health and safety or potentially contribute to radiological dose are reviewed for their environmental impacts. The potential radiological dose to the public is calculated as the maximum dose to a hypothetical Maximally Exposed Individual (MEI) at the BNL site boundary. Doses are calculated by considering all direct and indirect possible sources and pathways, such as inhalation of air emissions, ingestion of deer meat and fish, and any immersion dose. The dose assessment has routinely shown that the total effective dose equivalent from BNL activities is well below the EPA regulatory dose limits for the public, BNL workers, and the environment.

The average annual external dose from all potential ambient sources was 66 ± 11 mrem (660 ± 110 μSv) on site and 62 ± 10 mrem (620 ± 100 μSv) at off-site locations. Both measurements include contributions from natural background and cosmic radiation sources. A statistical comparison of the average doses measured at 47 on-site and 16 off-site thermoluminescent dosimeters (TLDs) showed that there was no additional external dose contribution from BNL operations above the dose from natural background radiation. In addition to these "background" measurements, nine TLDs were used to monitor known radiation source areas. The results of these measurements are described below.

The effective dose from air emissions was calculated as 4.40E-02 mrem (0.44 μSv) to the MEI. The ingestion pathway dose was estimated as 1.31 mrem (13.1 μSv) from consumption of deer meat and 0.37 mrem (3.7 μSv) from consumption of pumpkinseed fish caught on the BNL site. The total annual dose to the MEI from all pathways was estimated as 1.72 mrem (17.2 μSv). The BNL dose from the air inhalation pathway was 0.4 percent of EPA's annual regulatory dose limit of 10 mrem (100 μSv), and less than 2 percent of DOE's annual dose limit of 100 mrem (1,000 μSv) from all pathways.

Doses to aquatic and terrestrial biota were also evaluated and found to be well below the DOE regulatory limits. Other short-term projects, such as remediation work and waste management disposal activities, conducted in 2004 were also evaluated for their radiological emissions and potential dose impact; there was no radiological risk to the public or the environment from these activities. In conclusion, the overall dose impact from all BNL activities in 2004 was indistinguishable from natural background radiation levels.

8.1 DIRECT RADIATION MONITORING

A direct radiation-monitoring program is used to measure the external dose contribution to members of the public and workers from radiation sources at BNL. This is achieved by measuring direct penetrating radiation exposures both on and off site. The direct measurements taken at the off-site locations are with the

premise that off-site exposures are true natural background radiation (contribution from cosmic and terrestrial) exposures and represent no contribution from BNL operations. On- and off-site external doses were measured, averaged, and then compared using the statistical t-test to evaluate any variations and the contribution, if any, from BNL operations above natural background radiation.

Direct penetrating beta-gamma radiation is measured using TLDs. The principle of TLD operation is that when certain crystals are exposed to radiation, impurities in the crystals' low-temperature trapping sites are excited to higher energy states. These electrons remain in a high-energy state at normal ambient temperature. When the TLDs are heated (annealed), the electrons return to the lower energy state. As the electrons return to the lower energy state, they emit photon energy (i.e., light), which is measured with a photomultiplier tube; the light intensity given out is directly proportional to the absorbed radiation dose. The environmental TLDs used at BNL are composed of calcium fluoride and lithium fluoride crystals. The TLDs' accuracy is verified by comparing the absorbed dose of the TLD to a known radiation source. Also, BNL participates in the inter-comparison proficiency testing programs sponsored by DOE.

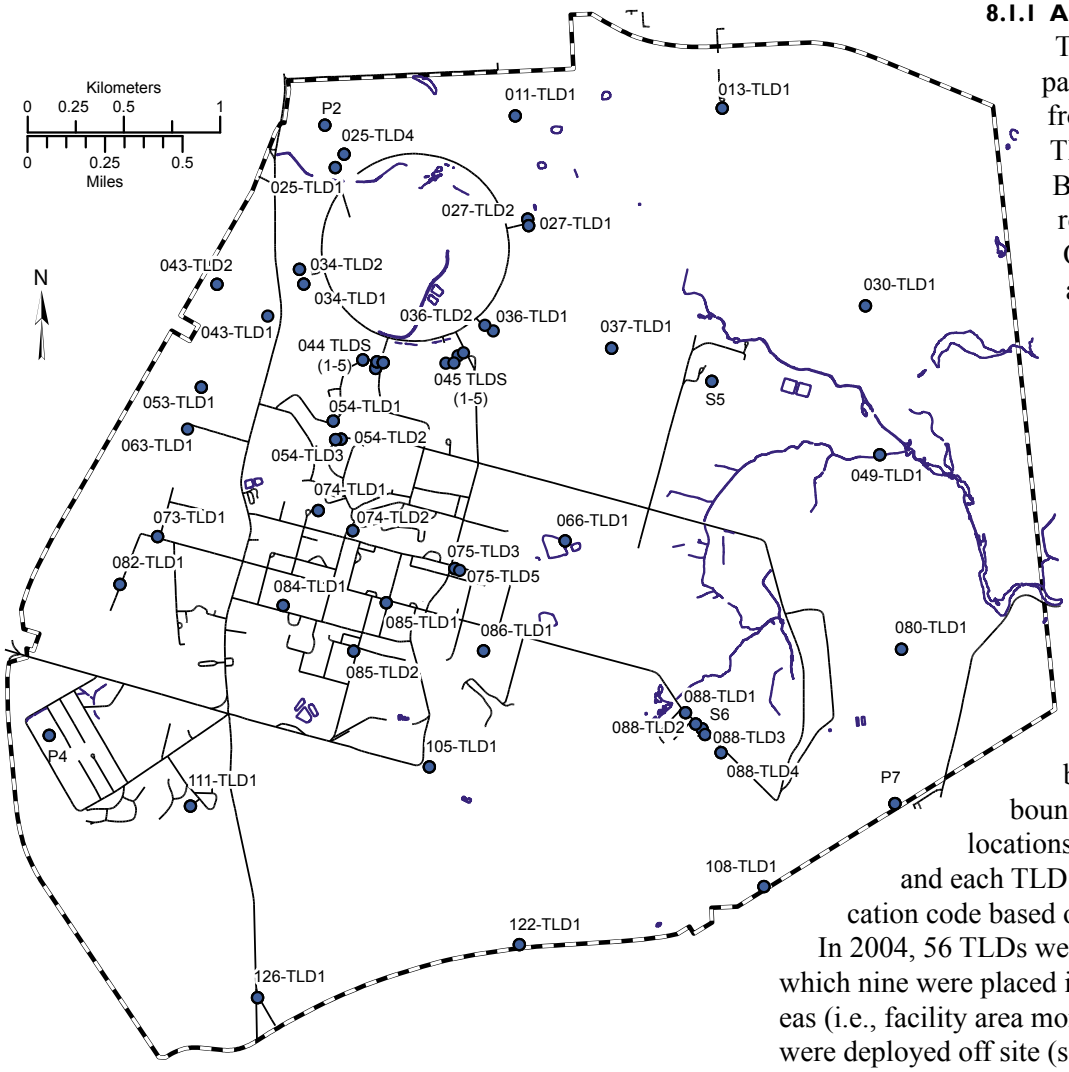

Figure 8-1. On-Site TLD Locations.

8.1.1 Ambient Monitoring

To assess the dose impact of direct radiation from BNL operations, TLDs are deployed at the BNL site and in the surrounding communities. On-site TLD locations are determined based on the potential for exposure to gaseous air plumes, atmospheric particulates, scattered radiation, and the location of historical radiation-generating facilities. The BNL perimeter is also posted with environmental TLDs to assess the dose impact, if any, beyond the Laboratory boundary. On- and off-site locations are divided into grids and each TLD is assigned an identification code based on these grids.

In 2004, 56 TLDs were deployed on site, of which nine were placed in known radiation areas (i.e., facility area monitoring TLDs) and 16 were deployed off site (see Figures 8-1 and 8-2 for locations). An additional 30 control TLDs were stored in a lead-shielded container in

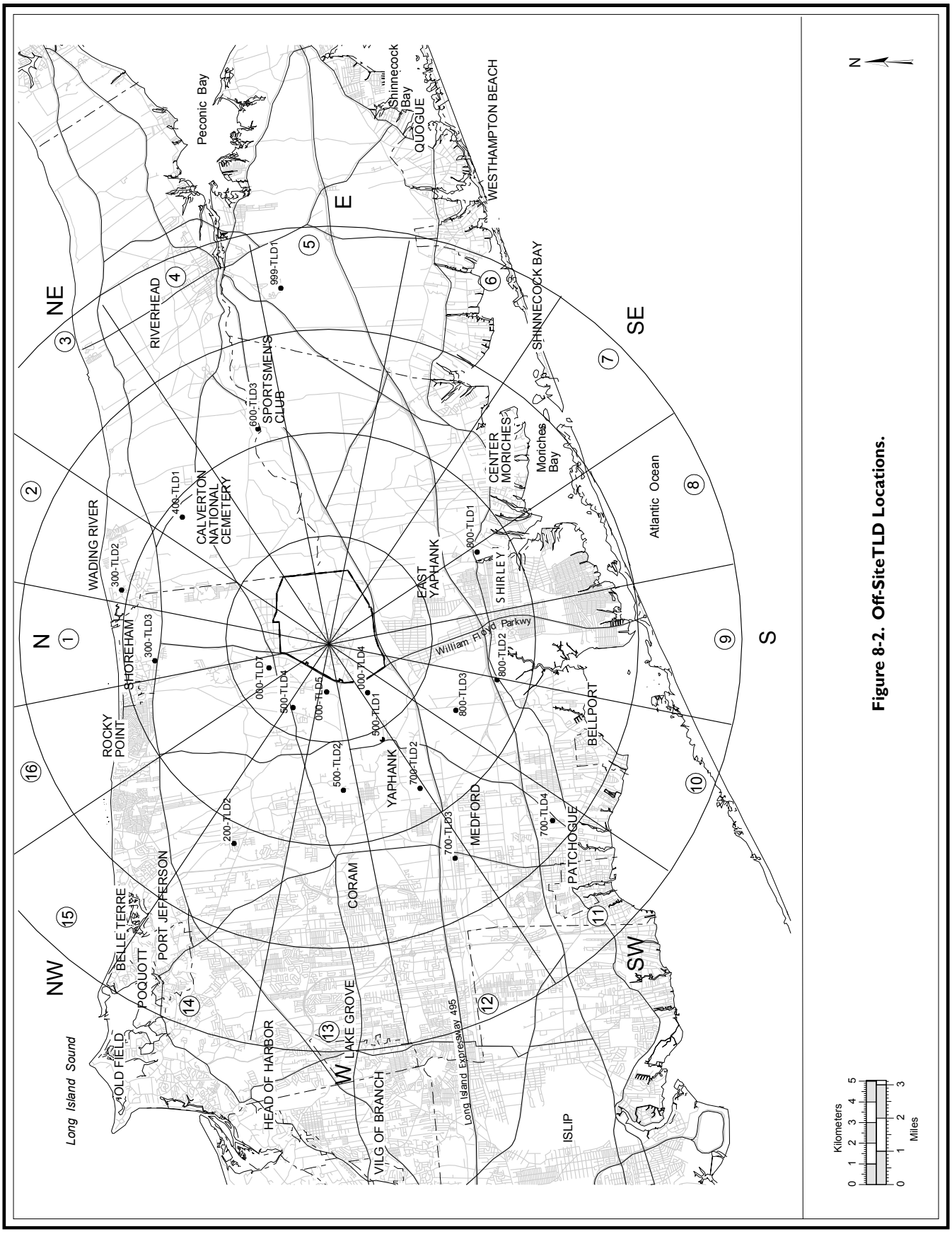

Figure 8-2. Off-SiteTLD Locations.

Building 490; the average of the control TLDs is reported as "075-TLD4" in Tables 8-1 and 8-2, for comparison. Note that some residual dose remains on the control TLDs when they are annealed and it is impossible to completely shield the control devices from all natural background and cosmic radiation sources. Therefore, small doses are measured by the control TLDs. The on- and off-site TLDs are collected and read quarterly to determine the external radiation dose measured.

Table 8-1 shows the quarterly and yearly on-site radiation dose measurements. The on-site average external dose for the first, second, third, and fourth quarters was 17.2 ± 3.3, 15.8 ± 3.6, 15.7 ± 3.2, and 17.5 ± 3.2 mrem, respectively. The annual on-site external dose from all potential environmental sources, including cosmic and terrestrial radiation sources, was 66 ± 11 mrem. Table 8-2 shows the quarterly and yearly off-site radiation dose measurements. The off-site average external dose for the first, second, third, and fourth quarters was 15.9 ± 2.7, 14.7 ± 2.4, 15.2 ± 3.3, and 16.1 ± 2.7 mrem, respectively. The annual off-site average ambient dose was 62 ± 10 mrem. To determine the BNL contribution to the external direct radiation dose, a statistical t-test between the measured on-site and off-site external dose averages was conducted. The t-test showed that there was no significant difference between the off-site dose (62 ± 10 mrem) and on-site dose (66 ± 11 mrem). From these measured doses, it can be safely concluded that there was no external dose contribution to on- and off-site locations from BNL operations in 2004.

8.1.2 Facility Area Monitoring

Nine of the 56 on-site TLDs were designated as the Facility Area Monitors (FAM). These TLDs are deployed at locations known to have radiation contamination, possible radiation scatter, or are in proximity to radiological posted areas with a higher probability to potentially contribute to the external radiation doses. Table 8-3 shows the external doses measured with the FAM TLDs. Environmental TLDs are posted at the S-6 blockhouse location and on the fence of the former Hazardous Waste Management Facility (HWMF) (088-TLD1 through 088-TLD4). These TLDs measured much higher external dose than typical natural background

Table 8-1. On-Site Direct Radiation Measurements.

TLD#	Location	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Avg./Qtr. ± 2σ (95%)	Annual Dose ± 2σ (95%)
		mrem					
011-TLD1	North firebreak	14.5	13.8	15.0	16.0	15 ± 2	59 ± 7
013-TLD1	North firebreak	17.9	13.8	13.8	15.2	15 ± 4	61 ± 15
025-TLD1	Bldg. 1010 beam stop 1	17.9	13.4	14.3	16.7	16 ± 4	62 ± 16
025-TLD4	Bldg. 1010 beam stop 4	16.7	14.1	14.9	16.7	16 ± 3	62 ± 10
027-TLD1	Bldg. 1002A South	15.5	15.9	13.8	14.9	15 ± 2	60 ± 7
027-TLD2	Bldg. 1002D East	15.3	16.5	13.5	15.2	15 ± 2	61 ± 10
030-TLD1	NE Firebreak	16.5	17.7	17.3	18.5	17 ± 2	70 ± 7
034-TLD1	Bldg. 1008 collimator 2	18.0	15.9	15.6	18.4	17 ± 3	68 ± 11
034-TLD2	Bldg. 1008 collimator 4	18.2	14.9	15.4	17.7	17 ± 3	66 ± 13
036-TLD1	Bldg. 1004B East	16.7	13.3	14.3	15.3	15 ± 3	60 ± 11
036-TLD2	Bldg. 1004 East	20.4	19.3	17.0	20.4	19 ± 3	77 ± 12
037-TLD1	S-13	15.8	14.0	14.6	17.2	15 ± 3	62 ± 11
043-TLD1	North access road	17.5	15.9	15.6	18.0	17 ± 2	67 ± 9
043-TLD2	North of Met. Tower	16.6	15.7	16.6	17.6	17 ± 2	67 ± 6

(continued on next page)

Table 8-1. On-Site Direct Radiation Measurements *(concluded).*

TLD#	Location	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Avg./Qtr. ± 2σ (95%)	Annual Dose ± 2σ (95%)
		mrem					
044-TLD1	Bldg. 1006	15.2	14.9	14.5	16.4	15 ± 2	61 ± 6
044-TLD2	South of Bldg. 1000E	15.3	15.1	15.1	17.0	16 ± 2	63 ± 7
044-TLD3	South of Bldg. 1000P	16.0	13.7	15.7	16.3	15 ± 2	62 ± 9
044-TLD4	NE of Bldg. 1000P	21.3	16.4	17.2	20.4	19 ± 5	75 ± 19
044-TLD5	N of Bldg. 1000P	16.8	15.3	15.2	17.9	16 ± 3	65 ± 10
045-TLD1	Bldg. 1005S	18.7	17.3	16.1	16.1	17 ± 2	68 ± 9
045-TLD2	East of Bldg. 1005S	17.5	16.2	16.7	19.1	17 ± 2	70 ± 10
045-TLD3	SE of Bldg. 1005 S	16.2	14.6	15.4	17.6	16 ± 2	64 ± 10
045-TLD4	SW of Bldg. 1005 S	15.9	15.0	15.6	18.2	16 ± 3	65 ± 11
045-TLD5	WSW of Bldg. 1005 S	13.8	13.0	12.8	16.2	14 ± 3	56 ± 12
049-TLD1	East firebreak	16.9	14.3	14.4	16.6	16 ± 3	62 ± 11
053-TLD1	West firebreak	19.6	17.5	17.3	19.3	18 ± 2	74 ± 9
054- TLD1	Bldg. 914	18.1	13.4	13.8	16.5	15 ± 4	62 ± 18
063-TLD1	West firebreak	19.2	17.2	16.9	18.7	18 ± 2	72 ± 9
066-TLD1	New HWM Facility	14.4	13.2	12.9	15.5	14 ± 2	56 ± 9
073-TLD1	W Meterology Twr. /Bldg. 51	16.9	15.5	18.8	18.4	17 ± 3	70 ± 12
074-TLD1	Bldg. 197	17.4	17.1	17.8	20.0	18 ± 3	72 ± 10
074-TLD2	Bldg. 907	17.4	13.7	15.2	16.5	16 ± 3	63 ± 13
080-TDL1	East firebreak	18.9	17.7	17.5	19.7	18 ± 2	74 ± 8
082-TLD1	West firebreak	18.5	17.9	18.0	20.0	19 ± 2	74 ± 8
084-TLD1	Tennis courts	17.6	17.2	17.3	18.9	18 ± 2	71 ± 6
085-TDL2	Upton gas station	19.2	17.7	15.4	17.6	17 ± 3	70 ± 12
085-TLD1	TFCU (Credit Union)	17.7	15.9	16.9	16.7	17 ± 1	67 ± 6
086-TLD1	Baseball fields	19.9	18.8	20.8	20.0	20 ± 2	80 ± 6
105-TLD1	South firebreak	17.9	20.5	16.4	19.6	19 ± 4	74 ± 14
108-TLD1	Water tower	15.4	16.2	15.6	16.5	16 ± 1	64 ± 4
111-TLD1	Trailer park	18.7	19.1	16.2	18.1	18 ± 2	72 ± 10
122-TLD1	South firebreak	15.4	15.0	15.1	17.2	16 ± 2	63 ± 8
126-TLD1	South gate	17.6	17.0	16.6	19.3	18 ± 2	71 ± 9
P2	North Firebreak	13.9	14.2	12.8	13.8	14 ± 1	55 ± 5
P4	Apartment Laundry	15.5	17.0	15.7	15.9	16 ± 1	64 ± 5
P7	South Firebreak	17.3	14.9	15.5	16.6	16 ± 2	64 ± 8
S5	Sewage Treatment Plant	18.5	14.7	14.5	16.3	16 ± 4	64 ± 15
On-site average		**17.2**	**15.8**	**15.7**	**17.5**	**17 ± 3**	**66 ± 11**
Std. dev. (2 σ)		**3.3**	**3.6**	**3.2**	**3.2**		
075-TLD4	**Control TLD average**	**8.96**	**8.62**	**8.96**	**12**	**10 ± 3**	**39 ± 12**

Note:
See Figure 8-1 for TLD locations.

Table 8-2. Off-Site Direct Radiation Measurements.

TLD#	Location	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Avg./Qtr. ± 2σ (95%)	Annual Dose ± 2σ (95%)
		mrem					
000-TLD4	Private property	14.8	13.2	13.4	15.6	14 ± 2	57 ± 9
000-TLD5	Smith Estate	16.5	14.8	15.3	15.1	15 ± 1	62 ± 6
000-TLD7	Mid-Island Game Farm	16.1	14.3	14.7	15.8	15 ± 2	61 ± 7
200-TLD2	Private property	18.2	17.8	19.0	18.0	18 ± 1	73 ± 4
300-TLD2	Private property	NP	NP	NP	NP		
300-TLD3	Private property	16.4	15.1	16.8	16.1	16 ± 1	64 ± 6
400-TLD1	Calverton National Cemetery	17.0	15.6	17.7	17.2	17 ± 2	68 ± 7
500-TLD1	Private property	13.8	12.2	12.3	13.7	13 ± 2	52 ± 7
500-TLD2	Private property	14.8	14.2	14.1	15.2	15 ± 1	58 ± 4
500-TLD4	Private property	16.5	15.2	NR	NR	16 ± 2	63 ± 7
600-TLD3	Sportsmen's Club	15.5	14.5	15.1	17.5	16 ± 3	63 ± 10
700-TLD2	Private property	15.7	14.1	14.5	NR	15 ± 2	59 ± 7
700-TLD3	Private property	16.6	15.2	16.0	18.2	17 ± 2	66 ± 10
700-TLD4	Private property	16.4	14.4	14.5	17.1	16 ± 3	62 ± 11
800-TLD1	Private property	12.6	14.6	14.6	14.1	14 ± 2	56 ± 8
800-TLD2	Private property	NP	NP	NP	NP		
800-TLD3	Suffolk County CD	17.3	15.6	15.7	15.2	16 ± 2	64 ± 7
999-TLD1	Private property	16.7	13.6	13.9	16.4	15 ± 3	61 ± 13
Off-site average		**15.9**	**14.7**	**15.2**	**16.1**	**15 ± 2**	**62 ± 10**
Std. dev. (2 σ)		**2.7**	**2.4**	**3.3**	**2.7**		
075-TLD4	**Control TLD average**	**8.96**	**8.62**	**8.96**	**12**	**10 ± 3**	**39 ± 12**

Notes:
NP = TLD not posted for the quarter
NR = TLD was not returned for measurements
See Figure 8-2 for TLD locations.

dose measured at other BNL areas. The high external dose measured at the former HWMF can be attributed to the presence of contaminated soil as the result of historical radioactive waste management practices at the facility. The entire former HWMF was demolished in 2003 and excavation of the contaminated soil commenced in 2004. Comparison of the 2004 dose rates with rates from previous years shows that the dose rates have begun to decline from the removal of the radioactive soils. The former HWMF is currently posted as a radiological area. Only radiation-trained personnel wearing personal dosimeters are allowed inside the facility until all the contaminated soil has been remediated.

Two TLDs (075-TLD3 and 075-TLD5) posted near Building 356 have shown higher quarterly averages, 26 ± 9 mrem (260 ± 90 μSv) and 28 ± 8 mrem (280 ± 80 μSv), respectively. The yearly doses were measured as high as 105 ± 36 mrem (1050 ± 360 μSv) for 075-TLD3, and 110 ± 33 mrem (1100 ± 330 μSv) for 075-TLD5. The direct doses are significantly higher than the on-site annual average. Building 356 houses a cobalt-60 (Co-60) source, which is used to irradiate materials, parts, and electronic circuit boards. The elevated measurements here can be attributed to the "sky-shine" phenomenon and shielding of Building 356. Although individuals who use the parking lot outside this building

Table 8-3. Facility Area Monitoring.

TLD#	Location	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Average ± 2σ (95%)	Annual Dose ± 2σ (95%)
		mrem					
054-TLD2	NE of Bldg. 913B	45.3	16.5	15.5	52.6	32 ± 38	130 ± 151
054-TLD3	NW of Bldg. 913B	27.2	15.8	14.1	16.5	18 ± 12	74 ± 47
S6	Outside Former HWMF	31.4	31.0	30.6	23.9	29 ± 7	117 ± 28
088-TLD1	Former HWMF-50' East of S-6	38.7	39.1	38.4	29.6	36 ± 9	146 ± 36
088-TLD2	Former HWMF-50' West of S-6	46.7	44.2	44.0	30.0	41 ± 15	165 ± 59
088-TLD3	Former HWMF-100' West of S-6	39.8	38.5	41.1	29.2	37 ± 11	149 ± 42
088-TLD4	Former HWMF-150' West of S-6	25.6	22.2	23.6	21.0	23 ± 4	92 ± 16
075-TLD3	Bldg. 356	28.5	31.6	23.5	21.4	26 ± 9	105 ± 36
075-TLD5	North Corner of Bldg. 356	31.2	30.3	26.9	22.0	28 ± 8	110 ± 33

Notes:
HWMF = Hazardous Waste Management Facility
See Figure 8-1 for TLD locations.

could receive a dose from this source, the dose would be minimal due to the limited time an individual spends in the parking lot.

Two TLDs placed on the fence northeast and northwest of Building 913-B (AGS Tunnel Access) also showed higher than normal ambient external dose. 054-TLD2, located on the northeast side of Building 913-B, showed higher dose in both the first quarter (45.3 mrem or 453 μSv) and the fourth quarter (52.6 mrem or 526 μSv). However, the northwest TLD (054-TLD3) showed a higher dose only in the first quarter (27.2 mrem or 272 μSv). The potential cause of the higher doses is under investigation for this area.

8.2 DOSE MODELING

EPA regulates radiological emissions from DOE facilities under the requirements set forth in 40 CFR 61, Subpart H, *National Emission Standard of Radionuclides Other than Radon* (NESHAPs). This regulation specifies the compliance monitoring and requirements for reporting the radiation doses received by members of the public from airborne radionuclides. The regulation mandates that no member of the public shall receive a dose from DOE operations that is greater than 10 mrem (100 μSv) in a year. The emission monitoring requirements are set forth in Subpart H, Section 61.93(b) and include the use of a reference method for continuous monitoring at major release points (defined as those with a potential to exceed 1 percent of the 10 mrem standard), and a periodic confirmatory measurement for all other release points. The regulations also require DOE facilities to submit an annual NESHAPs report to EPA that describes the major and minor emission sources and dose to the MEI. The dose estimates from various facilities are given in Table 8-4, and radiological emissions are discussed in more detail in Chapter 4.

As a part of the NESHAPs review process at BNL, any source that has the potential to emit radioactive materials is evaluated for regulatory compliance. Although the activities conducted under the Environmental Restoration (ER) Program are exempt under the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), these activities are monitored and assessed for any potential to release radioactive materials, and to determine dose contribution, if any, to the environment. In 2004, a number of ER, Waste Management, and other activities were evaluated for compliance with NESHAPs regulations. EPA's approved dose modeling software was used in dose calculations (see Section 8.2.1 for details). Because this model was designed to treat all radioactive emission sources as continuous chronic emissions over the course of a year, it is not well suited for estimating short-term or acute releases. Consequently, it overestimates poten-

Table 8-4. MEI Effective Dose Equivalent from Facilities or Routine Processes.

Building No.	Facility or Process	Construction Permit No.	MEI Dose mrem (a)	Notes
348	Radiation Protection	None	6.55E-10	(b)
463	Biology Facility	None	ND	(b)
490	Medical Research	BNL-489-01	3.88E-08	(b)
490A	Energy and Environment National Security	None	8.40E-08	
491	Brookhaven Medical Research Reactor	None	1.53E-06	(c), (e)
510	Calorimeter Enclosure	BNL-689-01	ND	(f)
510A	Physics	None	ND	(b)
535	Instrumentation	None	4.62E-14	(b)
555	Chemistry Facility	None	ND	(b)
725	National Synchrotron Light Source	None	8.95E-11	(b)
750	High Flux Beam Reactor	None	2.55E-05	(c)
801	Target Processing Lab	None	3.02E-05	(b), (c)
802B	Evaporator Facility	BNL-288-01	NO	(e)
820	Accelerator Test Facility	BNL-589-01	ND	(d)
830	Environmental Science Department	None	ND	(d)
865	Reclamation Building	None	ND	(c)
906	Medical-Chemistry	None	ND	
925	Accelerator Department	None	2.87E-10	(b)
931	Brookhaven Linac Isotope Producer	None	4.39E-02	(c)
938	REF/NBTF	BNL-789-01	ND	(g)
942	Alternate Gradient Syncrotron Booster	BNL-188-01	ND	(h)
---	Relativistic Heavy Ion Collider	BNL-389-01	ND	(d)
Total Potential Dose from BNL Operations			**4.402E-02**	
EPA Limit			**10.0 mrem**	

Notes:
Diffuse, fugitive, and other sources are not included in this table since they are short term emissions.
MEI = Maximally Exposed Individual
NBTF = Neutron Beam Test Facility
REF = Radiation Effects Facility
(a) "Dose" in this table means effective dose equivalent to MEI.
(b) Dose is based on emissions calculated using 40 CFR 61, Appendix D methodology.
(c) Emissions are monitored at the facility.
(d) ND = No dose from emission source in 2004.
(e) NO = Not operational in 2004.
(f) This has become a zero-release facility since original permit application.
(g) This facility is no longer in use; it produces no radioactive emissions.
(h) Booster ventilation system prevents air release through continuous air recirculation.

tial dose contributions from short-term projects and area sources. For that reason, the results are considered to be "conservative"—that is, erring on the side of caution.

8.2.1 Dose Modeling Program

Compliance with NESHAPs regulations is demonstrated through the use of EPA software, the Clean Air Act Assessment Package-1988 (CAP88-PC), Version 2.1.1. This computer program uses a Gaussian plume model to estimate the average dispersion of radionuclides released from elevated stacks or diffuse sources. It calculates a final value of the projected dose at the specified distance from the release point by computing dispersed radionuclide concentrations in air, rate of deposition on ground surfaces, and intake via the food pathway (where

applicable). CAP88-PC calculates both the effective dose equivalent (EDE) to the MEI and the collective population dose within a 50-mile radius of the emission source. The CAP88-PC model provides conservative doses, in most cases. For purposes of modeling the dose to the MEI, all emission points are located at the center of the developed portion of the BNL site. The dose calculations are based on very low concentrations of the environmental releases and are based on chronic, continuous intakes in a year. The input parameters used in the model include radionuclide type, emission rate in curies (Ci) per year, stack parameters such as height and diameter, and emission exhaust velocity. Site-specific weather and population data are factored into the dose assessment. Weather data are supplied by measurements from BNL's meteorological tower, which includes wind speed, direction, frequency, and air temperature (see Chapter 1, Section 1.7). Population data used in the model are based on the Long Island Power Authority population survey (LIPA 1999). Because visiting researchers and their families may reside at the BNL on-site apartment area for extended periods of time, these residents are also included in the population file used for dose assessment.

8.2.2 Dose Calculation Methods and Pathways

8.2.2.1 Maximally Exposed Individual

The MEI is defined as a hypothetical person who resides at the site boundary and has a lifestyle such that no other member of the public could receive a higher dose. This person is assumed to reside 24 hours a day, 365 days a year at the BNL site boundary in the downwind direction, and to consume significant amounts of contaminated fish and deer based on projections from the New York State Department of Health (NYSDOH). In reality, it is highly unlikely that such a combination of "maximized dose" to any single individual would occur, but the concept is useful for evaluating maximum potential risk and dose.

8.2.2.2 Effective Dose Equivalent

The EDE to the MEI for low levels of radioactive materials dispersed into the environment was calculated using the CAP88-PC, Version 2.1.1 dose model program. Site meteorology data were used to calculate annual dispersions for the midpoint of a given wind sector and distance. Facility-specific radionuclide release rates (Ci/year) were used for continuously monitored facilities. For small sources, the emissions were calculated using the method set forth in 40 CFR 61, Appendix D. The Gaussian dispersion model calculated the EDE at the site boundary and collective population dose values from immersion, inhalation, and ingestion pathways. These dose and risk calculations to the MEI are based on low emissions and chronic intakes.

8.2.2.3 Dose Calculation: Fish Ingestion

To calculate the EDE from the fish consumption pathway, the intake is estimated**.** Intake is the average amount of fish consumed by a person engaged in recreational fishing in the Peconic River, based on a NYSDOH study that estimated the consumption rate at approximately 15 pounds per year (NYSDOH 1996). For each radionuclide of concern for fish samples, the dry weight activity concentration was converted to picocuries per gram (pCi/g) wet weight, since "wet weight" is the form in which fish are caught and consumed. A dose conversion factor was used for each radionuclide to convert the activity concentration into the EDE. For example, the committed dose equivalent factor for cesium-137 (Cs-137) is 5.0E-02 rem/μCi, as set forth in DOE/EH-0071. The dose was calculated as: *dose* (rem/year) = *intake* (kg/year) x *activity in flesh* (μCi/kg) x *dose factor* (rem/μCi).

8.2.2.4 Dose Calculation: Deer Meat Ingestion

The dose calculation for the deer meat ingestion pathway is similar to that for fish consumption. The Cs-137 radionuclide dose conversion factor was used to estimate dose for each radionuclide. The total pounds of deer meat ingested during the course of a year was 64 pounds (NYSDOH 1999), based on the *U.S. Environmental Protection Agency Exposure Factors Handbook* (EPA 1996).

8.3 DOSES FROM DIFFUSE, FUGITIVE, AND OTHER SOURCES

Diffuse sources are described as emissions of radioactive contaminants to the atmosphere that do not have a well-defined emission point such as a stack. Such sources are also known as non-point or area sources. Fugitive sources include releases to the air that are not released through an actively ventilated air stream (i.e., leaks from vents). The following potential radiological remediation/diffuse sources were evaluated in 2004 for their potential contribution to the overall BNL site dose.

8.3.1 Brookhaven Graphite Research Reactor Remediation

The Brookhaven Graphite Research Reactor (BGRR) decommissioning project continued through 2004. The Below Ground Duct (BGD), a part of the BGRR reactor's primary air cooling system, was constructed from steel-reinforced concrete. The BGD consists of two separate north and south ducts, which are connected to the BGRR exhaust air plenums. The duct sections upstream of the air coolers have a three-part thermal liner. Between the primary and secondary liners is a 3-inch space that forms the secondary air duct; the inner duct is the primary duct. The primary liner was constructed of 4 inches of crimped aluminum sheets sandwiched between two carbon-steel (inner and outer) binding plates. The BGD exhaust air filters were successfully removed in 2003 and the BGD primary liners were removed in 2004. Staff had previously collected and analyzed samples from the primary liner to identify and characterize the level of radiological contamination. The following radionuclides were identified in the primary liner: Am-241, Co-60, Cs-137, Pu-238, Pu-239, Pu-240, Sr-90, U-234, U-235, and U-238. These radionuclides have half-lives ranging from about 5 years to nearly 4.5 billion years (see Appendix C for half-life definition). Removal of the liners involved cutting the primary ducts using a remotely controlled robotic manipulator. All cut pieces were then containerized for off-site disposal.

In addition to the liner removal, the DOE Proposed Remedial Action Plan for 2004 included the cleanup of contaminated soils adjacent to the BGRR and beneath the fuel storage canal, south secondary air system bustle, belowground expansion joint No. 4, cooler drain sump, and Building 701 north trench areas. The BGRR canal excavation and remediation plan included removing a portion of the canal, starting at the construction joint that is about 4 feet from column 7 of Building 701, and removing the east wall of the canal. The soils in the area were characterized and contamination was determined to be present from 10 feet below grade to about 24 feet below grade. The contaminated soils on the north, south, and east side of the canal were excavated to 4 feet below the canal. The volume of potentially contaminated soil from the canal was expected to be 443 yd^3, with the highest concentrations of Cs-137 at 1,500 pCi/g and strontium-90 (Sr-90) at 572 pCi/g. Trace amounts of uranium-238 (U-238) and plutonium-239 (Pu-239) were also found during the soil characterization activities. Other radionuclide concentrations were very low and statistically close to the analytical minimum detection limits and therefore not included in the NESHAPs evaluation. About 40 yd^3 of soil adjacent to the south secondary air system bustle were also removed, together with a 12-foot diameter caisson. As expected, the Cs-137 and Sr-90 soil concentrations near the air system bustle were the highest, at 89,000 pCi/g and 11,200 pCi/g, respectively. Other locations that were remediated are expansion joint No. 4 (6 yd^3 of soil), the cooler drain sump (2 yd^3 of soil), and the Building 701 north trench (2 yd^3 of soil). The amount of soil estimated for removal was conservative; excavation activities continued until the cleanup goals were met or the slumping of soil prevented any further excavation. Clean soil was used to fill the excavated areas. The contaminated soil in these areas was removed to the extent possible based on field conditions, and to a maximum depth possible until the established cleanup concentration levels were reached: 67 pCi/g for Cs-137 and 15 pCi/g for Sr-90. These cleanup levels are based on EPA's 15-mrem (150 μSv) dose free-release criteria for unrestricted land use. The dose to the MEI from the BGRR canal and

deep soil removal operation was estimated to be 2.5E-08 mrem (2.5E-07 μSv) in a year. An ambient air-sampling unit was set up in the downwind direction to ascertain if unplanned releases occurred and to quantify the airborne particulate activity during the soil removal process. The analytical results for particulate air filter samples were below the detection limits.

8.3.2 Alternating Gradient Synchrotron Snake Magnet Tritium Production

A NESHAPs compliance review was completed for possible tritium production in the liquid helium used to cool ("quench") the Alternating Gradient Synchrotron (AGS) "snake magnets." The potential for tritium production in the li-quid helium is due to secondary and tertiary hadrons scattering and absorption interactions. For all three snake magnet cooling systems, only a small quantity (19.5 mCi) of tritium was estimated to be produced, even with the most conservative assumptions—such as using the highest value of helium cross section, independent spallation cross sections above 100 MeV energy, and similar interaction cross sections for all types of particles. The tritium saturation concentration in the AGS Ring was well below the derived air concentration (DAC). Consequently, any fugitive losses to the environment were insignificant.

Based on the conservative assumption that all the tritium produced in the magnet quench was released to the environment, a NESHAPs compliance dose estimate was done using EPA's CAP88-PC dose modeling program. The dose modeling program calculated the EDE to the MEI in the southwest direction as 3.39E-8 mrem (3.4E-07 μSv) in a year. This is well below the 0.1 mrem/year (1 μSv) limit that would require a NESHAPs permit and continuous monitoring of the emission source. A confirmatory air sample taken and analyzed for gamma-emitting radionuclides, in accordance with standard operating procedures, also showed that there were no emissions of any other radionuclide from the AGS Ring. The AGS facility was found to be in compliance with the NESHAPs regulations even when any potential fugitive loss of tritium from the snake magnet cryogenic cooling system was factored in.

8.3.3 NESHAPs Evaluations for Other Remediation Projects

8.3.3.1 Remediation of Building 650 Dust Collectors

Building 650 (Hot Laundry) was used from the late 1950s through the late 1960s to clean radioactive contaminants from clothes and equipment. The disposal system included a hopper with bag filters to collect sandblast grit and small particles that were emitted during the mechanical cleaning process. The decontamination and decommissioning work objective was to demolish the baghouse, stabilize or remove the facility equipment, and dispose of the waste at a licensed, off-site disposal facility. The planned operational sequence was to remove the highly contaminated components, remove or affix any loose surface contamination, survey and decontaminate the other components (as necessary), remove the filter bags, and remove the hopper. The radiological source term was developed through surveys and sampling. The radiological dose and risk assessment to the MEI was estimated to be 6.85E-05 mrem (6.9E-04 μSv) in a year and would not require monitoring. Due to the presence of asbestos in the filter media and baghouse structures, work on this project was temporarily postponed in 2004 and will likely resume in 2005.

8.3.3.2 Sr-90 Remediation Systems

To remediate groundwater contaminated with Sr-90, a groundwater treatment system with five low-flow extraction wells was constucted in 2004. Two of the wells are located south and southeast of the Waste Concentration Facility (WCF), and three are located south of the BGRR area. Three drywells near Building 855 will be used to recharge the treated groundwater back to the aquifer. The Sr-90 treatment system includes a tray aeration system for removal of volatile organic compounds (VOCs), to be operated when necessary, with an oversized demister section to minimize the possibility of vapor releases, and an ion exchange, zeolite filtration system for Sr-90 removal. The well locations, screen intervals, and pumping rates were designed to avoid moving or capturing any existing tritium plumes from the g-2 and HFBR

tritium plume areas (see Chapter 7 for plume locations). A NESHAPs evaluation was completed to assess the potential dose to members of the public, because low concentrations of tritium were detected in the Geoprobe© characterization data. A conservative EDE estimate of 4.07E-09 mrem (4.1E-08 μSv) in a year was calculated for the Sr-90 remediation system when it becomes operational in 2005.

8.3.3.3 Building 811 Remediation and Former HWMF Soil Removal

The remediation work at the WCF, also known as Building 811, consisted of transferring and stockpiling soil and debris (i.e., concrete, geotextile, wood, and asphalt) generated by the removal activities for the underground storage tanks at the building. Excavated contaminated soil was transferred from Building 811 to the railroad unloading area at the northeast boundary of the former HWMF, which was designated as the soil stockpile location. This location was selected because the soil there is already contaminated and will require remedial excavation to a depth of 1 ft. below grade. The site provides convenient access to the railroad spur so the soil can be loaded onto railcars for off-site disposal.

Residual soil contamination in the former HWMF area was also removed during the remediation project. The former HWMF had a permit under the Resource Conservation and Recovery Act (RCRA) for processing and storing hazardous and mixed waste generated at BNL. Radioactive waste was managed under DOE guidelines. The former HWMF is a fenced 12-acre controlled compound in Operational Unit (OU) I located in the southeastern part of the BNL site. All the facility's structures were removed in 2003, and approximately 1,200 yd^3 of asphalt contaminated above the cleanup goal (67 pCi/g for Cs-137 and 15 pCi/g for Sr-90) were excavated and shipped off site for disposal.

The area contains a number of metal containment pits as well as concrete vaults and trenches that were used to store radioactive waste and sources. All stored waste materials have been removed and shipped off site for disposal. At present, only building slabs and foundations remain in the area. These were decontaminated to levels below the soil cleanup goals. However, soils, vaults, and trenches remain to be remediated and shipped to an off-site waste facility. The former HWMF also had a shallow seasonal pond, known as the former HWMF wetland, along the fence bordering the western side. Approximately 25 yd^3 of the sediment and surface soil in the wetland will be excavated and transported to an off-site disposal facility.

The remedial work radioactive source term for emission estimates was based on the concentration of radionuclides in the soil, concrete, external surface of the tanks, asphalt, and other structures. The EDE to the MEI from the remediation work at the former HWMF and Building 811 was estimated to be 3.31E-04 mrem (0.3E-03 μSv) in a year in the southeast direction.

8.3.4 NESHAPs Evaluations at the Waste Management Facility

8.3.4.1 Waste Characterization Activities

BNL's Waste Management Program is responsible for collecting, characterizing, consolidating, and repackaging hazardous, mixed, and radioactive materials for shipment to off-site licensed disposal facilities. Building 865 is the primary facility for the handling of radioactive waste.

Before being shipped to a disposal facility, a 55-gallon drum that had been packed and sealed at Building 865 required visual inspection to ascertain if the six small containers inside it were intact. A paint-type container (No. A0172) was of primary concern because it contained 13.7 Ci of U-238 and 15 Ci of tritium, in the form of uranium hydride. If this container had been breached, there was the potential to release tritium gas. A NESHAPs evaluation was completed for the drum, based on worst-case assumptions that radioactivity from the tritium and uranium source would become airborne during inspection. The EDE to the MEI at the southeast location was estimated to be 1.11E-05 mrem (1.1E-04 μSv) in a year. Nevertheless, the large drum was opened in Building 801's Room

67, the Hot Cell, which is specially designed for radiological work of this type.

8.3.4.2 Risk Analysis Scenario: Disposal of Neutron Source

The possibility of significant radiological dose from a mixed, compact, sealed neutron source was evaluated using an unlikely scenario, for assurance purposes. In this scenario, a 50-Ci americium-beryllium neutron source [^{241}Am–Be (α, n)] was acquired from Nuclear Materials and Equipment Corporation in 1965. The source activity after decay correction was calculated to be approximately 47 Ci. The sealed neutron source had a cylindrical shape and was doubly encapsulated in a tantalum inner container (0.050 in. thick) and stainless steel outer container (0.040 in. thick).

In the risk analysis, inhalation of the Am-241 radionuclide was considered most likely, so this pathway was used for the dose risk assessment. To estimate the potential airborne concentration, it was assumed that the neutron source's integrity was breached due to a strong mechanical force that subsequently spread the contamination in the building, leading to inhalation, with dose consequences to the worker and members of the public. Since the neutron source was solid, a conservative assumption was made that approximately 1E-06 fractions of activity would contaminate the surface area. Therefore, the spread source activity on the building surface inside would be 4.7E-05 Ci. It was also assumed that about 10 percent of the total surface area would be contaminated by the release of the Am-241 radionuclide. The total surface area that potentially could be contaminated was 13.72 x 28.04 m; therefore, the 10 percent surface area contaminated was estimated to be approximately 38.5 m^2.

Another parameter that is very important in dose estimation is the resuspension factor (RF), which is the ratio of the airborne concentration of a contaminant to the surface concentration of the contaminant. The RF is a highly sensitive parameter that would greatly influence the inhalation dose calculation. The RF would be affected by many physical factors, such as the type and intensity of physical (mechanical) action on the sealed source, nature of the surface contamination, particle size, particle distribution, surface chemistry, and any filtration mechanisms already in place. Another possible factor is the role of a ventilation system, which was assumed to be "off" for purposes of the risk analysis scenario. The effects of the entire list of possible variables were not evaluated in the dose risk assessment; a very simplistic, straightforward model was adopted.

It was assumed that 1 percent of the source radionuclide activity would be released into the environment through the stack in the building. The total volume of the release was estimated to be 5.38E3 m^3 (3,200 ft^3/min x 60 minutes x 0.028 m^3/ft^3). Based on these assumptions, the total activity that would be was released from the building was calculated to be 933 μCi. Therefore, the MEI residing 2,500 meters in the southeast direction would receive an EDE of < 0.1 mrem (1 μSv) in a year.

8.3.4.3 Tritium-Containing Bottles

Because there are no economically viable and technically feasible treatment methods available for tritium gas, a NESHAPs compliance review was completed for two small tritium-containing bottles that were to be vented via stack No. 105 in the Building 865 hood. The total volume released from the two bottles was 6.25 ft^3. The tritium activity in the first bottle was 20 mCi in 1996. Tritium activity in the second bottle was 0.5 μCi, as of 1998. Although decay correction would yield a lower quantity of tritium, these original activity levels were used as the basis for the airborne source term. According to the radiological dose risk assessment calculations, the tritium-containing bottles would yield an EDE to the MEI of 1.49E-11 mrem (1.5E-10 μSv) in a year.

8.3.5 NESHAPs Evaluations for Activation Sources

8.3.5.1 Former Source Development Laboratory

A NESHAPs compliance review was completed for the Source Development Laboratory, now known as the Deep Ultra Violet–Free Electron Laser (DUV–FEL), to update the Safety Assessment Document. There was a potential for air activation in the 4-inch air gap

due to bremsstrahlung radiation. The saturation activities of short-lived radionuclides nitrogen-13 (N-13) and oxygen-15 (O-15) were 8.5 μCi and 0.9 μCi, respectively. Consequently, the dose contribution to members of the public would be insignificant. The production of ozone gas was also estimated. The result was 4.2E-5 ppm for a 12-hour irradiation, which is well below the threshold limit for ozone of 0.1 mg/L. Therefore, the potential risk from ozone production was also miniscule.

8.3.5.2 National Synchrotron Light Source

During operation of the National Synchrotron Light Source (NSLS, Building 725), induced activity in air can be created by beam losses occurring during injection or routine operations. The induced activity is primarily created from high-energy photon or neutron interactions with nitrogen or oxygen atoms in the air, creating short-lived positron emitters such as carbon-11 (C-11), N-13, or O-15. In high-intensity and high-energy machines, these radionuclides can reach significant levels and require engineering or administrative controls to reduce exposure to personnel and releases into the environment. The potential for air activation was considered for two NSLS occurrences: 1) the beam transported to the x-ray or VUV injection line beam dumps, and 2) loss of stored beam in the x-ray ring. The operations and enclosures described below are located on the first floor in Building 725.

The saturation concentrations within the booster enclosure have previously been determined at 0.025 percent of the DAC for occupational workers. Therefore, it can be reasonably concluded that the dose to members of the public would be negligible. However, an environmental risk assessment was necessary for any fugitive emissions that could contribute to dose when emissions are outside of the accelerator enclosure. The fugitive emissions were evaluated to demonstrate compliance with EPA's annual limit of 10 mrem (100 μSv) to the members of the public from DOE facilities' operations. The total dose to the MEI resulting from the activation of air due to NSLS operations was estimated to be 1.12E-08 mrem (1.2E-07 μSv) in a year. Therefore, the dose and risk to the members of the public would be minimal. An annual administrative review of the facility is sufficient to evaluate any changes in operations, process, beam intensity, or any other contributing factor that may increase emissions to the environment.

8.4 DOSE FROM POINT SOURCES

8.4.1 Brookhaven Linac Isotope Producer

Source term descriptions for point sources are given in Chapter 4; however, an update on emissions from the Brookhaven Linac Isotope Producer (BLIP) is appropriate here because the BLIP is the only emission source with any potential to contribute dose to members of the public greater than 1 percent of the DOE limit. The BLIP facility uses the excess beam capacity of the Linear Accelerator (Linac) to produce short-lived radioisotopes for medical diagnostic procedures, medical imaging, and scientific research. During the irradiation process, the targets are cooled continuously by recirculating water in a 16-inch-diameter shaft. The principal gaseous radionuclides produced as a result of activation of the cooling water are O-15 and C-11. Because the BLIP facility is considered a major emission source, the facility emissions are directly measured using a low-resolution gamma spectrometer with an in-line sampling system connected to the air exhaust. This measures the short-lived gaseous products that cannot be sampled and analyzed by conventional methods. Particulates and radioiodines are monitored with filter paper and granular activated charcoal, which are exchanged weekly for laboratory analysis. The tritium sampler also operates continuously, with weekly sample collection and analyses.

In 2004, the BLIP facility operated over a period of 11 weeks. During the year, 680 Ci of C-11 and 2,027 Ci of O-15 were released from the BLIP facility. Tritiated water vapor from activation of the targets' cooling water was also released, 7.32E-02 Ci. The EDE to the MEI was calculated to be 4.39E-02 mrem (0.44 μSv) in a year from BLIP operations.

In 2004, the total emissions from the facility decreased in comparison to the previous two years, and an objective to further reduce emissions is being pursued for the BLIP facility. Since moisture is the primary source of emis-

sions (humidity from the Hot Cell's cooling water), a shroud enclosure was installed to enclose the cooling water surface (16-in.-diameter shaft), target holder transfer cases, chain drive assembly (including motor supports), and other associated accessories. This engineering control was expected to significantly reduce (by > 28 percent) the gaseous emissions from the BLIP facility. The efficiency test of the shroud seal was not satisfactorily completed in 2004 because of the short runtime (11 weeks) and lower beam energy irradiation of the targets. However, preliminary data show that the emissions reduction should be greater than 30 percent.

8.4.2 Brookhaven Medical Research Reactor

In January 2003, the fuel elements from the decommissioned Brookhaven Medical Research Reactor (BMRR) were shipped off site, greatly reducing any potential for radioactive emissions from the facility. The semiannual particulate and charcoal air sampling data showed no quantifiable radioactive emissions from the BMRR. In September 2003, BNL initiated a requested for approval from EPA to stop monitoring emissions at the BMRR. However, EPA requested clarification of the sampling data submitted and also required an additional round of sampling before making a final determination to eliminate BMRR emissions sampling. During 2004, the semiannual sampling of the BMRR continued and the results confirmed that there were no radionuclides present with the exception of naturally occurring potassium-40 (K-40), at 164 ± 35 pCi per filter. Although there was no airborne environmental pathway for tritium emissions, as all the vents from the reactor vessel are sealed, a dose estimate was completed for the evaporative loss of tritium. The dose risk to the MEI from tritium was estimated to be 7.42E-08 mrem (7.4E-07 μSv) in a year.

8.4.3 Unplanned Releases

There were no unplanned releases in 2004.

8.5 DOSE FROM INGESTION

Deer and fish bioaccumulate radionuclides in their tissues and organs, and therefore samples are analyzed to evaluate the dose contribution to humans from the ingestion pathway. As discussed in Chapter 6, deer meat samples collected off site and less than 1 mile from the BNL boundary were used to assess the potential dose impact to the MEI. Eleven samples of deer meat (flesh) were used to calculate the "off site and less than 1 mile" averages for the purpose of dose calculations. K-40 and Cs-137 were the two radionuclides detected in the tissue samples. K-40 is a naturally occurring radionuclide and is not related to BNL operations. The average K-40 concentrations were 3.6 ± 1.1 pCi/g (wet weight) in the flesh and 2.7 ± 0.8 pCi/g (wet weight) in the liver. The average Cs-137 concentrations were 0.9 ± 0.3 pCi/g (wet weight) in the flesh and 0.4 ± 0.1 pCi/g (wet weight) in the liver (average for the "off site and less than 1 mile" as shown in Table 6-2). The potential dose from consuming deer meat with the average Cs-137 concentration was estimated as 1.31 mrem (13.1 μSv) in a year. This is about 13 percent of the health advisory limit of 10 mrem (100 μSv) established by NYSDOH.

In collaboration with the New York State Department of Environmental Conservation (NYSDEC) Fisheries Division, BNL maintains an ongoing program of collecting and analyzing fish from the Peconic River and surrounding freshwater bodies. In 2004, the pumpkinseed samples collected by NYSDEC in the Peconic River on the BNL site had the highest concentration (1.05 pCi/g) of Cs-137, so this value was used to estimate the EDE to the MEI (assuming consumption of 15 pounds of fish). The potential dose from consuming fish was estimated at 0.37 mrem (3.7 μSv) in a year. It is highly unlikely that an individual would consume fish with the highest concentration from this location, but these data were used to estimate potential maximum dose as a worst-case scenario for the MEI.

8.6 DOSE TO AQUATIC AND TERRESTRIAL BIOTA

DOE-STD-1153-2002, *A Graded Approach for Evaluating Radiation Doses to Aquatic and Terrestrial Biota*, provides the guidelines for screening methods to estimate radiological doses to aquatic animals, terrestrial plants, and

Table 8-5. BNL Site Dose Summary.

Pathway	Dose to Maximally Exposed Individual	Percent of DOE 100 mrem/year Limit	Estimated Population Dose per year
Inhalation			
Air	0.044 mrem (0.44 μSv)	<1%	0.16 person-rem
Ingestion			
Drinking water	None	None	None
Fish	0.37 mrem (3.7 μSv)	<1%	Not tracked
Deer Meat	1.31 mrem (13.1 μSv)	<2%	Not tracked
All Pathways	1.72 mrem (17.2 μSv)	<2%	0.16 person-rem

terrestrial animals using environmental surveillance data. The RESRAD-BIOTA 1.0 biota dose screening program was used to evaluate compliance with the requirements for protection of biota specified in DOE Order 5400.5 (1990), *Radiation Protection of the Public and the Environment,* and proposed Rule 10 CFR 834, Subpart F (66 FR 25380). The terrestrial animal and plant doses were evaluated based on 0.56 pCi/L of Sr-90 in surface waters at the HE sampling location on the Peconic River (see Figure 5-8 for sample station locations). Soil samples collected at location P4 had concentrations of Sr-90 of 0.18 pCi/g. The dose to terrestrial animals was calculated to be 6.09E-06 Gy/day and 1.10E-06 Gy/day to terrestrial plants. The doses to terrestrial animals and plants were well below the biota dose limit of 1 mGy/day.

For calculating dose to aquatic animals, values from Swan Pond were used because both the surface water and sediment samples came from the same location. The Cs-137 sediment concentration was 460 pCi/kg, and Sr-90 concentration in surface water was 0.78 pCi/L. The aquatic animal dose was estimated to be 2.38E-07 Gy/day; to riparian animals, the estimated dose was 2.93E-06 Gy/day. Therefore, the dose to aquatic and riparian animals was also well below the 10-mGy/day limit specified by the regulations.

8.7 CUMULATIVE DOSE

Table 8-5 summarizes the potential cumulative dose from the BNL site. The total dose to the MEI from air and ingestion pathways was estimated to be 1.72 mrem (17.2 μSv), as shown in Table 8-5. In comparison, the EPA regulatory limit for the air pathway is 10 mrem (100 μSv) and the DOE limit from all pathways is 100 mrem (1,000 μSv). The effective dose was well below the DOE and EPA regulatory limits, and the ambient TLD dose was within normal background levels seen at the BNL site. The potential dose from drinking water was not estimated, because most of the residents adjacent to the BNL site get their drinking water from the Suffolk County Water Authority and not private wells.

To put the potential dose impact into perspective, a comparison was made with other sources of radiation. The annual dose from all natural background sources and radon is approximately 300 mrem (3000 μSv). A diagnostic chest x-ray would result in 5 to 20 mrem (50-200 μSv) per exposure. Using natural gas in homes yields about 9 mrem (90 μSv) per year, cosmic radiation yields 26 mrem (260 μSv), and natural potassium in the body yields approximately 39 mrem (390 μSv) of internal dose. Even with conservative estimates of dose from the air pathway and ingestion of local deer meat and fish, the cumulative dose from BNL operations in 2004 was well below the dose that could be received from a single chest x-ray.

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