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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
08/30/2006	00	New Site Profile for the Brookhaven National Laboratory. First approved issue. Incorporates internal and NIOSH formal review comments. Training required: As determined by the Task Manager. Initiated by Melton H. Chew.
04/26/2010	01	Revision to incorporate SEC language, BNL comments, and clarifies that background radiation is not included in occupational exposure. References were consolidated at the end of the document. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Melton H. Chew.
02/07/2013	02	Revision initiated to incorporate extension of SEC period through 1993, SC&A comments and DCAS concerns. Section 1 was revised to include a new Section 1.2 to address SEC issues. Section 2 was revised to include updated information on facilities including expansion of information on the LMFR. Section 3 was revised to correct PFG dose values (Section 3.7) and set medical X-ray uncertainty to 30% (Section 3.8). Section 4.4 was revised to update external ambient dose values (Table 4-1) and extend ambient intakes to 2010 (Tables 4-2 and 4-3). Section 5 was updated to provide additional information on minimum detection limits for bioassay sample analyses (Sections 5.4 and 5.5) and guidance on evaluation of internal dose to volunteers participating in human testing by the Medical Department (new Section 5.4.5). Section 6 was revised to update discussion of neutron dosimetry (Section 6.4), add annual average worker dose for unmonitored workers (new Section 6.5.2), expand guidance on neutron track fading and angular dependence of NTA film (Section 6.10), add guidance on evaluation of external dose to volunteers participating in human testing by the Medical Department (new Section 6.11), and revise uncertainty factor guidance (Section 6.12). Incorporates formal internal and NIOSH review comments. Information added on three facilities in response to SC&A issues and findings in Sections 2.4.16.3, 2.4.16.4, 2.4.16.5, 5.4 and Tables 2-1, 6-4, and 3-1. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Dennis L. Strenge.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AGS	Alternating Gradient Synchrotron
AMAD	activity median aerodynamic diameter
AP	anterior-posterior
ATR	AGS-To-RHIC
BGRR	Brookhaven Graphite Research Reactor
BLAF	Brookhaven Laboratory Animal Facility
BLIP	Brookhaven LINAC Isotope Producer
BMRR	Brookhaven Medical Research Reactor
BNL	Brookhaven National Laboratory
Bq	becquerel
BRAHMS	Broad Range Hadron Magnetic Spectrometers
CERN	European Organization for Nuclear Research (formerly Conseil Européen pour la Recherche Nucléaire)
CFR	Code of Federal Regulations
cGy	centigray
Ci	curie
CLIF	Chemistry LINAC Irradiation Facility (also CLIP in some documents)
cm	centimeter
CNF	Cold Neutron Facility
CR-39	Columbia Resin Number 39
d	day
D&D	decontamination and decommissioning
DØ	D-zero
DCF	dose conversion factor
DTL	Drift-Tube-LINAC
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EML	Environmental Measurement Laboratory
ENSD	entrance skin dose
ESE	entrance skin exposure
EXSD	exit skin dose
F	fast absorption type
FOV	field of view
ft	foot
g	gram
GeV	gigaelectron-volt, 1 billion electron-volts
GIF	Gamma Irradiation Facility
GSD	geometric standard deviation
HASL	Health and Safety Laboratory
HEBT	High-Energy Beam Transport

HEPA	high-efficiency particulate air (filter)
HFBR	High Flux Beam Reactor
HIRDL	High Intensity Radiation Development Laboratory
hr	hour
HVL	half-value layer
HWMF	Hazardous Waste Management Facility
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt (1,000 electron-volts)
kV	kilovolt
kVp	kilovolts-peak
L	liter
LAT	lateral
lb	pound
LEAF	Low Energy Accelerator Facility
LET	linear energy transfer
LINAC	linear accelerator
LMFR	Liquid Metal Fuel Reactor
M	moderate absorption type
m	meter
mA	milliampere
mAs	milliampere-second
mCi	millicurie
MDA	minimum detectable activity or amount
MDD	minimum detectable dose
MDL	minimum detectable level
MEL	Metallurgical Evaluation Laboratory
MeV	megaelectron-volt, 1 million electron-volts
MFP	mixed fission product
mL	milliliter
mm	millimeter
mR	milliroentgen
mrad	millirad
MRC	Medical Research Center
mrem	millirem
mrep	millirep
MW	megawatt
n	neutron
NaI(Tl)	Sodium iodide doped with thallium
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NSLS	National Synchrotron Light Source
NSRL	National Aeronautics and Space Administration Space Radiation Laboratory
NTA	nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program

NYOO	New York Operations Office
OEF	Office of Environmental Restoration
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
OW	open window
p	proton
PA	posterior-anterior
pCi	picocurie
PET	positron emission tomography
PFG	photofluorography
PHENIX	Pioneering High Energy Nuclear Interaction Experiment
POC	probability of causation
ppm	parts per million
QF	quality factor
R	roentgen
R&D	research and development
radwaste	radioactive waste
RARAF	Radiological Research Accelerator Facility
RBE	relative biological effectiveness
RF	radio frequency
RFQ	Radio-Frequency Quadrupole
RHIC	Relativistic Heavy Ion Collider
RSD	remote skin dose
RTF	Radiation Therapy Facility
S	slow absorption type
s	second
S&EP	Safety and Environmental Protection (Division)
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
STAR	Solenoidal Tracker (Detector)
STRL	Severn-Trent Richland Laboratory
t	tritium
T	tritium
TARL	TestAmerica Richland Laboratory
TLD	thermoluminescent dosimeter
TPL	Target Processing Laboratory
TRISTAN	Terrific Reactor Isotope Separator to Analyze Nuclides
U.S.C.	United States Code
VUV	vacuum ultraviolet
WB	whole body
WBC	whole-body counter
WCF	Waste Concentration Facility
wk	week
WMF	Waste Management Facility

yr	year
µA	microampere
µCi	microcurie
µg	microgram
µm	micrometer
§	section or sections

1.0 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy [DOE] facility" as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as "any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)" [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …" [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) restrict the "performance of duty" referred to in 42 U S. C. § 7384n(b) to nuclear weapons work (NIOSH 2010).

The statute also includes a definition of a DOE facility that excludes "buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program" [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled "Exposure in the Performance of Duty"] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2010):

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

[1] The U.S. Department of Labor (DOL) is ultimately responsible under the EEOICPA for determining the POC.

1.1 PURPOSE

The purpose of this document is to provide a site profile that contains technical basis information for the evaluation of the total occupational dose for EEOICPA claimants who were employed at the Brookhaven National Laboratory (BNL). Section 1.2 details the BNL classes in the Special Exposure Cohort (SEC), and Section 1.3 describes the document content.

1.2 SPECIAL EXPOSURE COHORT PETITION INFORMATION

Classes Added to the SEC

Brookhaven National Laboratory (1947 to 1979)

> *All employees of the Department of Energy, its predecessor agencies, and its contractors and subcontractors who worked in any area at Brookhaven National Laboratory in Upton, New York, from January 1, 1947, through December 31, 1979, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

Based on the findings and recommendations of NIOSH and the Advisory Board on Radiation and Worker Health, the Secretary of the U.S. Department of Health and Human Services has concurred with the finding that NIOSH does not have access to sufficient personnel or area monitoring data, or sufficient source or source term information, about BNL operations to bound potential internal exposures for the period from January 1, 1947, through December 31, 1979 (other than tritium after December 31, 1964) (Sebelius 2009).

Brookhaven National Laboratory (1980 to 1993)

> *All employees of the Department of Energy, its predecessor agencies, and its contractors and subcontractors who worked in any area at Brookhaven National Laboratory in Upton, New York, from January 1, 1980 through December 31, 1993, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

Through the course of ongoing dose reconstruction and research, NIOSH has determined that the issues identified in the first Brookhaven National Laboratory evaluation (NIOSH 2009) carry over into later periods at the site. Specifically, NIOSH has determined that, due to ongoing difficulty of consistently obtaining all requested personnel monitoring records for individual BNL claims, NIOSH is unable to estimate with sufficient accuracy internal exposures for workers at Brookhaven National Laboratory during the period from January 1, 1980, through December 31, 1993. NIOSH has found that there is a potential for undocumented worker movements across the site, and that there is limited claimant-specific information about work locations. NIOSH has found that a determination cannot always be made as to whether or not an employee worked in technical areas with a history of radioactive material use, or whether an employee should have been monitored for radiological exposures. Considering this information, NIOSH is unable to eliminate any specific worker from potential exposure scenarios based on assigned work location (Sebelius 2012).

Although NIOSH has found that it is not possible to completely reconstruct radiation doses for the SEC classes discussed above, it intends to use internal and external monitoring data that might become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at

the Brookhaven National Laboratory facility during the SEC periods stated above, but who do not qualify for inclusion in the SEC, can be performed using these data as appropriate to support a partial dose. NIOSH has found that it does have access to sufficient personnel and workplace monitoring data to bound potential external exposures and occupational medical dose for workers at BNL during the SEC periods.

1.3 SCOPE

This site profile consists of the following sections: 1.0, Introduction; 2.0, Site Description; 3.0, Occupational Medical Dose; 4.0, Occupational Environmental Dose; 5.0 Occupational Internal Dose; 6.0, Occupational External Dosimetry; and 7.0, Attributions and Annotations.

1.3.1 Site Description

Section 2.0 of this site profile describes the features and the history of BNL, including site areas or buildings, site processes, periods of operation, radionuclides of concern, and other information pertinent to dose reconstruction.

This section provides a description and general information for the facilities that operated or are operating at BNL. These include the Brookhaven Graphite Research Reactor (BGRR) and its replacement, the High Flux Beam Reactor (HFBR); the operations of the Medical Research Center (MRC), the positron emission tomography (PET) facility, and the medical research reactor; and the operation of the accelerators and production of radiopharmaceuticals. The section also addresses some of the high-energy physics facilities such as the relativistic heavy-ion collider complex where ion beams traveling in opposite directions in a circular orbit collide.

1.3.2 Occupational Medical Dose

Section 3.0 of this site profile provides information about the historical BNL occupational X-ray program. Beginning in 1947 and continuing through at least 1964, workers, including visiting scientists employed for more than 3 months, were given an entry and annual physical examination including a posterior-anterior (PA) chest X-ray. Later, the annual frequency might have actually been closer to about once every 2.5 years. The section presents organ doses for both PA and lateral (LAT) chest radiographs assuming both properly collimated and poorly collimated exposures. Skin doses inside and outside the primary beam are determined for PA and LAT chest projections.

1.3.3 Occupational Environmental Dose

Section 4.0 discusses the radiation doses received by workers at BNL while on the site but outside the facilities. From 1967 to 1984, ambient gamma radiation dose rates are available from four perimeter stations on the northwest, southwest, southeast, and northeast. Beginning in 1985, the onsite monitoring program was expanded to approximately 20 stations. The ambient external gamma radiation outside the radiological controlled areas is the result of background radiation or gamma emissions from stack effluent such as ^{41}Ar or skyshine due to air scatter from an otherwise well-shielded radiation source. The radionuclides emitted from the BNL stacks are identified and their annual inhalation in becquerels per year are listed. Neither potable groundwater nor soil ingestion has been found to be a pathway of exposure. No ingestion dose is indicated.

1.3.4 Occupational Internal Dose

Section 5.0 discusses the internal dosimetry systems and practices at BNL. The section provides supporting technical data to evaluate internal occupational doses that can reasonably be associated with worker radiation exposures that are covered by EEOICPA. It lists the types of routine monitoring

that constitute the *in vitro* and the *in vivo* dose control programs as well as the minimum detectable activities (MDAs), counting methods, and reporting practices. The section addresses the uncertainty for BNL exposure and dose records.

1.3.5 Occupational External Dosimetry

Section 6.0 discusses external dosimetry systems and practices at BNL. It presents historical and current practices in relation to the evaluation of external exposure data for monitored and unmonitored workers. It contains supporting documentation to assist in the evaluation of occupational external exposures from processes that occurred at BNL. In addition, it addresses the evaluation of worker exposure, missed dose, and the bias and uncertainty that are associated with the monitoring of external dose. The missed doses were basically those from dosimeter minimum detectable levels (MDLs) and exchange periods, as well as the inability to determine true doses from high-energy particles that were generated by the accelerators. Neutron doses were measured with nuclear track emulsion, type A (NTA) film until 1995, after which Columbia Resin Number 39 (CR-39) has been used.

1.3.6 Attributions and Annotations

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.0.

2.0 SITE DESCRIPTION

2.1 INTRODUCTION

BNL was founded in 1947, and it has been in operation since then at Upton, Long Island, New York. The site was formerly Camp Upton, and it was used by the Army during World Wars I and II.

BNL's early research focused on advanced physics but expanded into its current suite of research in the fields of medicine, biology, chemistry, physics, materials science, nuclear engineering, and environmental research. BNL was established to provide facilities for scientific research and was organized into departments that provide research nuclear reactors, particle accelerators, and engineering facilities, and that support the Biology, Chemistry, Physics, Medicine, Applied Science, Accelerator, and Applied Mathematics Departments.

2.2 PURPOSE

The purpose of this section is to describe the BNL facilities, the periods of the process operations, and major site incidents that resulted in significant internal or external exposure to one or more persons.

2.3 SCOPE

This section describes the features and history of BNL, including site areas, buildings, processes, periods of operation, radionuclides of concern, and other information pertinent to dose reconstruction. Much of the information is taken from the book "Making Physics, A Biography of Brookhaven National Laboratory, 1946 – 1972" by Crease (1999).

2.4 SITE DESCRIPTION AND GENERAL INFORMATION

Figure 2-1 shows an aerial view of BNL.

Figure 2-1. Aerial view of Brookhaven National Laboratory (BNL 2009a).

Research Reactors

The Laboratory's scientific history began in 1950 with the operation of the BGRR, a research reactor used for peaceful scientific exploration in the fields of medicine, biology, chemistry, physics, and nuclear engineering. The BGRR operated until 1968. In 1965, its capacity was replaced and surpassed by the HFBR, which provided neutrons to researchers of all disciplines, from solid-state physics to art history. The HFBR ceased operations in December 1996.

Medical Research Center

Medical research at BNL began in 1950 with the opening of one of the first hospitals devoted to nuclear medicine. It was followed by the MRC in 1958 and the Brookhaven Medical Research Reactor (BMRR) in 1959. The BMRR is the first nuclear reactor built exclusively for medical and biological research. It came on line on March 15, 1959, and operated until October 2000.

The Radiation Therapy Facility (RTF) was operated jointly by the BNL Medical Department and the State University of New York at Stony Brook. The RTF was a high-energy, dual X-ray mode linear accelerator (LINAC) for radiation therapy of cancer patients. The RTF has been dismantled and is not operational.

Chemists and physicians teamed to view the inner workings of the brain in 1977 with the advent of PET cameras. Two more imaging techniques were added to the PET research efforts to form the Center for Imaging and Neuroscience in 1996. Except for the BMRR, all of these facilities are currently operating.

Particle Accelerators

High-energy particle physics research began in 1952 with the Cosmotron, the first particle physics accelerator to achieve billion electron-volt energies. The Cosmotron operated from 1953 to 1966. In 1960, the Alternating Gradient Synchrotron (AGS), a large accelerator, was built to surpass the Cosmotron's capabilities. The AGS is capable of accelerating protons to energies up to 30 GeV and heavy ion beams to 15 GeV/amu. The AGS achieved full energy in 1960 and is still in use.

Between 1967 and 1970, the Tandem Van De Graaff, 60-Inch Cyclotron, and Vertical Accelerator were used for medium-energy physics investigations and for special isotope production. The heavy ions from the Tandem Van De Graaff can be injected into the AGS for physics experiments. The Tandem Van De Graaff began operating in 1970 and continues operating to the present.

The Heavy Ion Transfer tunnel connects the coupled Tandem Van de Graaff and the AGS. The interconnection of these two facilities permits intermediate-mass ions to be injected into the AGS where they can be accelerated to an energy of 15 GeV/amu. These ions then are extracted and sent to the AGS experimental area for physics research. The AGS Booster is a circular accelerator with a circumference of 200 m that receives either a proton beam from LINAC or heavy ions from the Tandem Van De Graaff. The Booster accelerates proton particles and heavy ions before injecting them into the AGS ring. The Booster receives protons and heavy ions from the LINAC and Tandem Van De Graaff facilities to increase their intensity for delivery to the AGS.

The Brookhaven LINAC Isotope Producer (BLIP) became operational in 1973. Protons from the LINAC are sent through an underground beam tunnel to the BLIP facility where they strike target metals. These metals, which become activated by the proton beam, are then processed at the Target Processing Laboratory (TPL) for use in radiopharmaceutical development and production. The targets are cooled by a continuously recirculating water system. The BLIP facility underwent significant upgrades in 1994 in support of the Brookhaven Isotope Research Center program. The 200-MeV Proton Linear Accelerator serves as a proton injector for the AGS and supplies a continuous beam of protons for radionuclide production by spallation reactions in the BLIP.

In 1982, the National Synchrotron Light Source (NSLS) began operation. The NSLS guides charged particles in an orbit. As the electrons spin inside a hollow donut-shaped tube called an electron storage ring, they emit light called synchrotron light. This light, which can be detected by specialized instruments, is used to study the properties of matter. The NSLS uses a LINAC and booster synchrotron as an injection system for two electron storage rings that operate at energies of 750 MeV vacuum ultraviolet (VUV), and 2.5 GeV X-ray. The synchrotron radiation produced by their stored electrons is used for VUV spectroscopy and for X-ray diffraction studies.

Brookhaven's newest accelerator facility is the Relativistic Heavy Ion Collider (RHIC), which was completed in 1999. The RHIC is designed to recreate a state of matter that scientists believe existed moments after the universe was formed.

Department of Applied Science

The TPL (also called the Hot Laboratory in 1993) officially opened on January 15, 1951. The original purpose of the central facility was to provide appropriately shielded areas for research with large amounts of radioactive material. The "hot" area of the Hot Laboratory included five hot cells, three chemical-processing hot cells, and three high-level hot cells for handling and processing of radioisotopes in gaseous, liquid, or solid form.

In 1971 and 1972, the High Intensity Radiation Development Laboratory (HIRDL), which contained million-curie-range (in the early 1970s) ^{60}Co and ^{137}Cs sources, was used for source development and experimental process irradiations. A ^{60}Co pool in the HIRDL facility operated at lower activity levels into the 1990s.

Waste Management
The new Waste Management Facility (WMF) replaced the original Hazardous Waste Management Facility (HWMF) in its entirety and consolidated several waste management operations into functional buildings. The WMF also provides for significant expansion in a dedicated site suitable for handling and storing hazardous and radioactive wastes generated at BNL.

Figure 2-2 shows the location of the major sites where nuclear research activities were carried out at BNL. Table 2-1 lists the major nuclear research facilities and their associated summary information; it also lists radionuclides of concern (see Section 2.5). The following subsections comprise further descriptions of the research facilities of significance on the BNL site and provide background information relevant to dose reconstruction.

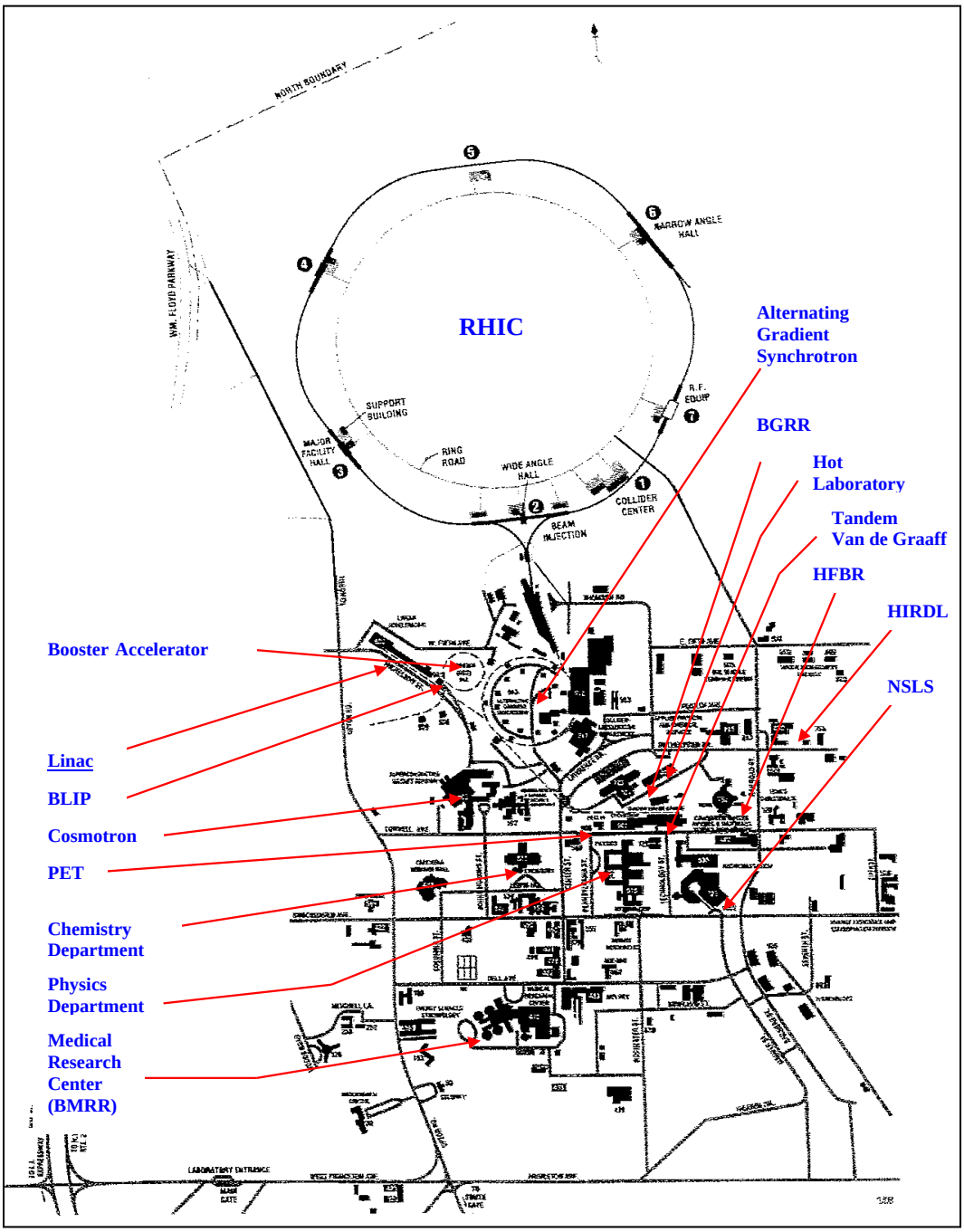

Figure 2-2. Nuclear and radioactivity sites at BNL [1].

Table 2-1. Site information of major nuclear research facilities at BNL.

Section	Name	Description	Period	Building No.	Note	Radionuclide
2.4.1	BGRR	U-fueled, graphite-moderated and -reflected reactor 28 MW	1950–1969	701		Ba-140
						Ce-144
						Cs-137
						La-140
						Nb-95
						Ru-103
						Ru-106
						Sr-89
						Sr-90
						Zr-95
						U-235
						U-238
						U-234
						Pu-239
2.4.1	BGRR fuel canal	BGRR fuel storage	1950–1972	709		Same as above
2.4.1	BGRR exhaust fanhouse	Highly contaminated ducting and fans	1950–1972	704		Cs-137
						Sr-90
2.4.1	BGRR exhaust filter house	Contaminated HEPA[a] filter elements	1950–unknown	708	Filter elements still in place in 1997	Same as above
2.4.2	HFBR	30- to 60-MW thermal, heavy-water-moderated research reactor	1965–1999	750	Ceased operation due to leak in spent fuel pool in 1997	Be-7
						Br-77
						Br-82
						Cs-137
						H-3
						I-126
						Mn-54
						Xe-133
						Xe-135
2.4.2	HFBR Stack	100-m stack and houses iodine filters	1965–1999	705		I-131
						I-132
						I-133
						I-134
						I-135
2.4.2	CNF	Produced extremely low-energy neutrons by moderating neutrons with liquid hydrogen	1980–1996	751		None
2.4.3	MRC	Medical Department	1959–present	490	Dispersible radioisotopes, PuBe sources (removed)	None

Section	Name	Description	Period	Building No.	Note	Radionuclide
2.4.4	BMRR	Enriched U-fueled, light-water-moderated and -cooled reactor	1959–2000	491		Ar-41
						Al-26
						As-76
						Ba-128
						Ba-140
						Br-82
						Ce-141
						Ce-144
						Co-60
						Fe-59
						Hg-203
						I-124
						I-131
						I-133
						La-140
						Mo-99
						Na-24
						Sb-122
						Sc-46
						Se-75
						Sr-91
						Tc-99m
						Ti-44
						Xe-133
						Xe-135
						Zn-65
						Zn-69m
2.4.5	PET Imaging Laboratory	PET scanners for nuclear medicine and neurosciences research	1977–present	906	Use of positron emitters (C-11, F-18, N-13, and O-18)	C-11
						N-13
						F-18
						O-15
						Ge-68
2.4.6	Cosmotron	Proton accelerator	1949–1968	902		C-11
						N-13
						O-15
2.4.7	AGS	A large accelerator capable of accelerating protons to energies up to 30 GeV and heavy ion beams to 15 GeV/amu	1960–present	913	AGS tunnel, short-lived activation products in tunnel	C-11
						N-13
						O-15
					Long-lived activation	C-14

Section	Name	Description	Period	Building No.	Note	Radionuclide
					products	Na-22, Na-24
						Cr-51
						Mn-54
						Fe-55, Fe-59
						Co-60
						Ni-59, Ni-63
						Zn-65
2.4.7	200-MeV LINAC	Preaccelerator and LINAC providing proton source	1967–present	930		None
2.4.7	LINAC cooling support service	Provides water-cooling services for the LINAC complex	1967–present	930 MER	Contains activated water	None
2.4.7	AGS Booster Tunnel	Accumulator booster ring	1967–present	942	Short-lived activation products in tunnel	C-11
						N-13
						O-15
					Long-lived activation products	C-14
						Na-22, Na-24
						Cr-51
						Mn-54
						Fe-55, Fe-59
						Co-60
						Ni-59, Ni-63
						Zn-65
2.4.7	AGS Experiment Area	Experimental target area for the AGS ring and later contains a proton beam switchyard system	1960–present	912	Activated components	C-14
						Na-22, Na-24
						Cr-51
						Mn-54
						Fe-55, Fe-59
						Co-60
						Ni-59, Ni-63
						Zn-65
2.4.7	AGS ventilation system fanhouses	Ventilation exhaust fans	1960–present	913A to 913E	High radiation area due to proximity to the tunnel; activated water	
2.4.7	AGS LINAC	Contains 50-MeV AGS LINAC	1960s	914		H-3
2.4.7	E10 Power Supply Building	Houses an assortment of power supplies	1971–present	920	Radiation hazard from activated cooling water and oil in transformer	
2.4.7	Siemens motor generator house	Electrical equipment and lubricating equipment for the motor generator	1960–present	928, 929	Radiation hazard from activated cooling water	

Section	Name	Description	Period	Building No.	Note	Radionuclide
2.4.7	AGS Department	Office, field testing shops, machine shops, high bays.	1960–present	911	Storage and repair of activated components	C-14
						Na-22, Na-24
						Cr-51
						Mn-54
						Fe-55, Fe-59
						Co-60
						Ni-59, Ni-63
						Zn-65
2.4.7	g-2/Bubble Chamber	30-in. and 80-in. Bubble Chamber	1962–1978	919	Tritium release in 1973	H-3
2.4.7	Proton Target	Proton target that produces secondaries for the g-2 ring	1962–present	949	Highly activated target	
2.4.7	7-ft Bubble Chamber	High-energy physics experimental facility	1972–1979	960	Activated components	
2.4.7	AGS Warehouses	Warehouses	1960–present	975, 918, 196, 209, 424, 178	Storage of activated components	C-14
						Na-22, Na-24
						Cr-51
						Mn-54
						Fe-55, Fe-59
						Co-60
						Ni-59, Ni-63
						Zn-65
2.4.8	BLIP	Production of radioisotopes by irradiation in the LINAC beam (200-MeV protons), C-11, F-18, N-13 and O-18	1972–present	931B	Buildings 931A and 931C have no nuclear hazard	As-72
						As-74
						Be-7
						Co-57
						Co-58
						Co-60
						Cs-132
						Cs-137
						H-3
						Mn-54
						Na-22
						O-15
						Sc-46
						Co-56
						Ga-68
						Ge-69
						I-126

Section	Name	Description	Period	Building No.	Note	Radionuclide
						Xe-127
						Zn-65
2.4.9	Tandem Van de Graaff	Van de Graaff accelerators,60-in. proton Cyclotron producing 10-MeV protons, 20-MeV deuterons, and 40-MeV alpha particles for medical radionuclide research	1970–present	901	Dynamitron (positron accelerator) operations ceased in 1989	Sr-90/Y-90
						Ca-47
						Cu-67
						Ar-38
						Mg-28
						I-124
						I-133
						Sc-47
						Po-210
						Kr-83m
						Ga-68
						Sm-151
						Te-132
						I-132
						Na-22
						H-3
						Mo-99
						Te-99m
2.4.10	NSLS	Produces intense sources of X-ray, ultraviolet, and infrared radiation	1981–present	725		
2.4.10	NSLS Development Laboratory	Houses a LINAC	1981–present	729	Has previous radioactive works	
2.4.11	RHIC Complex	High-energy particle accelerator and collider	1999–present	1002	Uranium plate machining	
		BRAHMS experimental hall	1981–present	1004A	Has previous radioactive works	
		RHIC service building	1981–present	1004		
		Open area (future experimental area)	1996–present	1004B		
		RHIC support building	1994–present	1005		
		Beam tunnel	1981–present	1005S		
		Offices and tech shops	1981–present	1006		
		STAR Detector	1985–present	1006A		
		Service building	1981–present	1007		
		Beam tunnel	1981–present	1008		
		PHENIX experiment	1988–present	1008A		
		Experimental support building	1985–present	1010		
		PHOBOS experiment	1994–present	1012		

Section	Name	Description	Period	Building No.	Note	Radionuclide
		Future experimental hall	1994–present			
2.4.12	Hot Laboratory	Radioisotope processing and production, TPL, and radiopharmaceutical research	1951–present	801	Explosion of UF_6/BrF_3 on 5/15/1957; a few people were injured	As-72
						As-74
						Be-7
						Co-57
						Co-58
						Co-60
						Cs-132
						Cs-137
						H-3
						Mn-54
						Na-22
						Au-199
						Bi-213
						Br-77
						Br-82
						Se-75
						V-48
						I-126
						Se-75
						Ge-69
						U-natural
2.4.13	HIRDL	Radioactive sources fabrication, irradiation cells and GIF	1951–present GIF 1970–present	830	GIF contained approximately 35,000 Ci of Co-60 (removed) and megacurie Co-60 and Cs-173 sources (removed)	Co-60
						Cs-137
2.4.14	Chemistry Department	Studies of chemical kinetics, hot-atom effects, nuclear chemical studies of nuclear decay schemes and nuclear reaction cross-sections, solar neutrinos, and radiocarbon dating	1966–present	555		

Section	Name	Description	Period	Building No.	Note	Radionuclide
2.4.15	Physics Department	Laboratories, office space, a machine shop, and basement area	1962–present	510	Radioactive material storage, counting laboratories	
2.4.16.1	LMFR Support Facility	R&D work for the LMFR	1957–1975	820		U-235
						U-238
						U-234
2.4.16.2	Hot Machine Shop	Hot Machine Shop	1947–1975	530		
2.4.16.3	Instrument Division	R&D of X-ray and neutron detectors and corresponding electronics, which are used at the NSLS, the HFBR, the AGS, and other worldwide scientific research facilities.	1964–present	535		
2.4.16.3	Instrument Division, Physics Department	Irradiation facility for material R&D	1960 – present	356	Sealed sources	Co-60
2.4.16.4	CERF	Gamma irradiation facility	1951 – present	463	Sealed sources	Co-60
						Cs-137
2.4.16.5	S&EP/348	Radiological Calibrations Facility	1959 - present	348	Sealed beta, gamma, and neutron sources	Co-60
						Cs-137
						PuBe
						Ra-226
						Sr-90
						U Nat
2.4.17	Radwaste Reclamation Building	Located in the WMF for processing of radwaste	1997–present	865		
2.4.17	Tritium Evaporator Facility	Reduce tritium release by evaporation	1995–present	802B		I-133
						Rb-83
						Rb-86
						Be-7
						Co-56
						Co-57
						Co-58
						Co-60
						Cs-137
						H-3
						Mn-54
						Na-22
						Se-75

Section	Name	Description	Period	Building No.	Note	Radionuclide
						Zn-65
2.4.17	Waste Management Incinerator	Low-level waste incinerator	Early 1970s to early 1990s	444	Incinerated animal carcasses with no radiological content	H-3
						Sr-85
						I-125
						Co-57
						Sc-47
						Zn-65
						C-14
						P-32
						S-35
						Fe-59
						Sn-117m
						Sn-113
						Cr-51
						Be-7
2.4.18	Radiological Waste Decontamination Facility	Decontamination of radiological waste		650		

a. HEPA = high-efficiency particulate air.

2.4.1 Brookhaven Graphite Research Reactor, Buildings 701, 704, 708, and 709

Building 701
The BGRR was a graphite-moderated and -reflected, thermal-neutron, air-cooled research facility. The original fuel loading was natural uranium, which was replaced in later years by highly enriched fuel elements. The 25-ft, 700-ton graphite cube was built in two halves separated by a narrow vertical gap running east-west. Filtered cooling air was drawn into this gap and flowed through north-south channels removing heat from the fuel elements and graphite; it then flowed out of the reactor through underground concrete ducts, passing through filters, coolers, and primary exhaust fans that discharged into a 100-m high stack (BNL 1997a, p. 3).

The BGRR achieved criticality on August 22, 1950, and operated at power levels up to 28 MW with natural uranium fuel. In April 1958, the reactor was reloaded with highly enriched uranium fuel elements and operated at power levels up to 20 MW. On June 10, 1968, the use of the reactor for experimental purposes ended and initial decommissioning operations began. The last fuel was shipped to the Savannah River Site in June 1972, and the canal was pumped dry (BNL 1997a, p. 4).

Building 704
The BGRR was shut down in 1968 and partially decommissioned in 1972. The exhaust ducts from the reactor and to the stack have been sealed from the fans. The fans remain in their cells and are contaminated. The intake duct, exhaust duct, and fans are grossly contaminated from fuel failures that occurred with the natural uranium fuel (BNL 1997b, p. 6).

Building 708
Radioactive hot air was filtered and discharged from the reactor through underground concrete ducts between the reactor (Building 701), the instrument house (Building 708), and the fanhouse (Building 704). The north and south exit air ducts were contaminated as a result of fuel ruptures in the original natural uranium fuel. Due to decay over some 30 years since use of uranium metal fuel, the major remaining fission products should be ^{137}Cs and ^{90}Sr. On August 18, 1988, a radiological survey was conducted on each exhaust duct upstream of the filters. Results indicated 20,000 to 30,000 dpm in the ducting. This indicates 40 to 70 times higher than the unrestricted limit of 500 dpm. Filter elements and exhaust coolers have been removed (BNL 1997b, p. 6).

Building 709
Building 709 was the fuel transfer and storage canal of the BGRR. During the 1950s, before the replacement of the natural uranium fuel slugs, there was a substantial problem with corrosion of fuel elements in the canal. This fuel was subject to deterioration and oxidation while in storage, making the slugs “dusty” when they were eventually prepared for shipment. The new type of fuel elements, which replaced the natural uranium elements in 1958, eliminated this problem. The new elements were aluminum-clad enriched uranium.

During the early years of operation of the BGRR, fuel failures occurred that resulted in radioactive materials being released to the air stream that cooled the reactor. There were 28 reported ruptures of BGRR fuel from 1952 to 1957. These all occurred with the natural uranium fuel. There was one rupture of a uranium oxide (U_3O_8) sample that was being irradiated for the radioiodine production program. Aside from ^{41}Ar, ^{131}I was the most important radionuclide that would contribute to a potential dose that was discharged to the atmosphere from the BGRR. Bromine-82 and ^{133}I were released in somewhat larger concentrations (Meinhold and Meinhold 2001).

More than 2,414 fuel elements, generated over a 12-year period, were shipped from the canal. The contaminated water, filter media, and backflushing from the ion exchange columns were pumped to the storage tanks at the Waste Concentration Facility, Building 811.

By 1972, the canal was finally drained, cleaned with detergents and water, and covered with concrete slabs for shielding (BNL 1997b, p. 8).

2.4.2 High Flux Beam Reactor Complex, Buildings 750, 705, and 751

The BGRR capacity was replaced and surpassed in 1965 by the HFBR (BNL 2001a). The HFBR is a 30-MW thermal, heavy-water-moderated nuclear research reactor. The reactor is designed to provide intense beams of neutron radiation to be used in the study of a variety of neutron scattering research projects and to allow for the radiating of materials close to the reactor core by insertion through special tubes near the reactor core. The HFBR is fueled with fully enriched uranium and aluminum alloy and is moderated and cooled by heavy water. Initial criticality was achieved on October 31, 1965. The HFBR was originally designed to operate at 40 MW. In 1979, a newer fuel design was employed that provided increased uranium loading in anticipation of an overall facility upgrade. In 1982, several major modifications were completed that allowed operation at power levels up to 60 MW.

In 1990, as a result of questions about the ability of natural circulation to cool the reactor during certain emergencies, reactor operations were limited to 30 MW. Heavy water flowing in the core is exposed to a dense neutron field that activates the deuterium atoms in the water to produce tritium. The primary mechanism by which tritium is transferred from the interior coolant system to the atmosphere is depressurization of the reactor vessel and evaporative losses during maintenance and refueling operations. During a scheduled maintenance shutdown in 1997, a leak in the HFBR's spent fuel storage pool was discovered. In November 1999, the Secretary of Energy made a decision to permanently close the HFBR.

Building 750
Building 750 is a domed cylindrical structure that contains the reactor and almost all associated process systems. The reactor, its beam lines, and laboratories are on the second main level known as the Experimental Level. The topmost level, the Operations Level, contains the Control Room and the Instrument Maintenance Shops.

Building 705
Building 705 is a 100-m stack that provides the path for ventilation exhaust from the HFBR. A 30-in. underground duct exhausts air from Building 750 ventilation blowers to particulate and iodine filters and then up through the stack. Stack effluent downstream of the filters is monitored from the Stack Monitoring Facility in Building 715 (BNL 1997c, pp. 6, 7).

Building 751
In 1980, the Cold Neutron Facility (CNF) was installed at beamline H9. This facility provided extremely low energy neutrons by moderating thermal neutrons with liquid hydrogen. Cooling of the liquid hydrogen was provided by liquid helium cooled by a large compressor in Building 751.

2.4.3 Medical Research Center, Building 490

The Medical Department (Building 490) was opened in 1956 and conducted research in the building. Active clinical research involving human volunteers is conducted in the Clinical Research Center and its satellite facilities. Animal research is conducted in the Brookhaven Laboratory Animal Facility (BLAF). Benchtop laboratory research is conducted in support of the various programs. Studies involving dispersible radioisotopes (used primarily as tracers) are carried out in the building.

There are several other facilities in the building, among them the Whole Body Neutron Irradiation Facility, the Prompt Gamma Neutron Facility, the Partial Body Neutron Activation Facility, the Inelastic Neutron Scattering Facility, the Whole Body Counter (WBC), and two single-photon emission

computed tomography cameras. Some of these facilities contain or contained radiation sources [e.g., PuBe sources (removed) for the Whole Body Neutron Irradiation Facility were stored in a vault below the facility and were raised into the facility when in use]. In addition, there are many fume hoods and other apparatus typically found in research laboratories (BNL 1997d, pp. 4, 5).

2.4.4 Brookhaven Medical Research Reactor, Building 491

The BMRR, Building 491, is the first nuclear reactor built exclusively for medical and biological research. It came on line on March 15, 1959, and operated until October 2000. It is an integral part of the MRC and is in a 60-ft-diameter steel and concrete structure adjacent to the MRC (Building 490). It is fueled with enriched uranium, moderated and cooled by light water, and operated intermittently at power levels up to 3 MW-thermal. The reactor is capable of operating for short periods at power levels up to 5 MW. Air from the interior of the containment building is used to cool the neutron reflector that surrounds the core of the BMRR vessel. When air is drawn through the reflector, it is exposed to a neutron field that causes the natural argon gas in the air to become radioactive (as ^{41}Ar). Due to a reduction of research funding, the BMRR conducted its last run on December 2000; transition and stabilization activities began in 2001.

The research function of the reactor was to provide radiation beams of known, controllable character and strength at the designed sample and treatment locations. In 1997, its primary use was to provide neutron beams for Boron Neutron Capture Therapy research and patient treatment. Testing of neutron capture compounds and techniques was ongoing (BNL 1997e, p. 4).

2.4.5 Positron Emission Tomography, Building 906

Building 906 was constructed in 1981 and houses two state-of-the-art PET scanners for use in basic research in the fields of neurology, nuclear medicine, and neurosciences.

Radiopharmaceuticals labeled with short-lived positron emitters (^{11}C, ^{18}F, ^{13}N, and ^{15}O) are used for physiological and biochemical research.

In 1987, the research group acquired a new PET scanner and another in 1997. The name changes for Building 906 have been evolutionary; today it is called the Pet Imaging Laboratory (BNL 1997f, p. 5).

2.4.6 Cosmotron, Building 902

The 902 Complex was originally built for the Cosmotron proton accelerator and experimental facilities. Some of the areas date back to Camp Upton. The Cosmotron was a very-low-intensity accelerator.

After the Cosmotron ceased operation, the Medical Department continued the operation of the Van de Graaff accelerator for a radiobiological program. The balance of the high bay area of Building 902 was used for research and development (R&D) of superconducting magnets. Eventually, the Medical Department moved its equipment out and the high bay was devoted entirely to the development and fabrication of a superconducting magnet.

The operational chronology of the Building 902 High Bay is:

1949–1968	Cosmotron
1968–1980	Radiation effects facility in the south portion of the high bay
1968–present	Superconducting magnet development and fabrication and other equipment assembly (BNL 1997g, p. 2)

2.4.7 Alternating Gradient Synchrotron, 900 Series Buildings

The AGS achieved full energy in 1960 and was capable of accelerating protons to energies up to 30 GeV and heavy ion beams to 15 GeV/amu. When such accelerated particles collide and interact with nuclei, part of their energy is transformed into new particles, which fly off from the target nuclei. These particles are then detected or their paths are made visible by devices such as scintillation counters, Cerenkov counters, bubble chambers, spark chambers, and photographs. The AGS has been continuously upgraded and it now provides the highest intensity of protons per pulse and the highest intensity of polarized protons per pulse than any other synchrotron in the world. The AGS also accelerates heavy ion beams for experimental use as well as for the RHIC.

The AGS consists of three different accelerating systems: the 200-MeV LINAC, which is the source of the protons housed in Building 930, the 200-m-circumference rapid cycling Booster synchrotron in Buildings 914 and 942, and the 800-m AGS main ring designated as Building 913. The Tandem Van de Graaff accelerator, which is a separate facility, provides heavy ions for the Booster ring through a beam transfer line. The particles that are accelerated by the AGS facility are extracted into and stopped in various experimental areas around the ring, or are sent through a transfer line to the RHIC. Building 912 was the original experimental target area. The asphalt pad inside the AGS ring adjacent to Building 912 was an experimental area in the 1960s with two secondary beam lines. Since that time, activated components have been stored and repaired in the area. Figure 2-3 shows the buildings associated with the AGS Complex.

The AGS LINAC, Building 930 and Associated Facilities
The LINAC was designed and built in the late 1960s as part of a major upgrade to the AGS facility. The LINAC tunnel is 140 m long and consists of nine large accelerators. From the end of the LINAC, a 17-m spur tunnel injects the beam into the booster accelerator. Another high-energy beam transport (HEBT) line extends 95 m to connect with the AGS. Another spur tunnel provides beam to the BLIP facility, the Chemistry LINAC Isotopes Facility (CLIF), the Radiation Effects Facility, the Neutral Beam Test Facility, and a beamstop. Only the BLIP facility is presently used, and it has recently undergone a significant upgrade to increase its intensity. The LINAC consistently produces a beam of 25 to 35 mA with a 0.5-ms pulse length or 1.0×10^{14} protons per pulse. Pulses are repeated at the rate of 7.5 per second, but at most only 1 in 7 pulses are transported to the Booster with the remainder transported to the BLIP facility.

Relevant buildings directly attached to Building 930 are:

- Building 930 MER provides water-cooling services for the LINAC complex and contains activated water. Associated with it is a large water-cooling tower and a compressor area.

- Building 930 Annex was added in the 1970s and contains shops and storage areas. Building 930A was added in the late 1980s as a service building for the Booster project. It contains power supplies and computer control equipment for the Booster (BNL 1997h).

The AGS Accelerator Rings, Buildings 913 and 914, and Associated Facilities
The AGS tunnel, Building 913, has a circumference of 800 m and is buried under a 7-m-thick berm of earth and concrete. There are 240 large alternating gradient dipole electromagnets, more than 100 supporting focus and beam tuning magnets as well as radiofrequency (RF) accelerating components, vacuum systems, water-cooling systems, etc., in the tunnel. Supporting the tunnel operation are the ventilation system fanhouses and multiple power supply houses. Some of these houses are designated 913A through R, while others have a separate plant engineering building number. The AGS tunnel (Building 913) is interrupted at the Experimental Hall, Building 912, for roughly 65 m. Here the AGS is under heavy concrete shielding blocks.

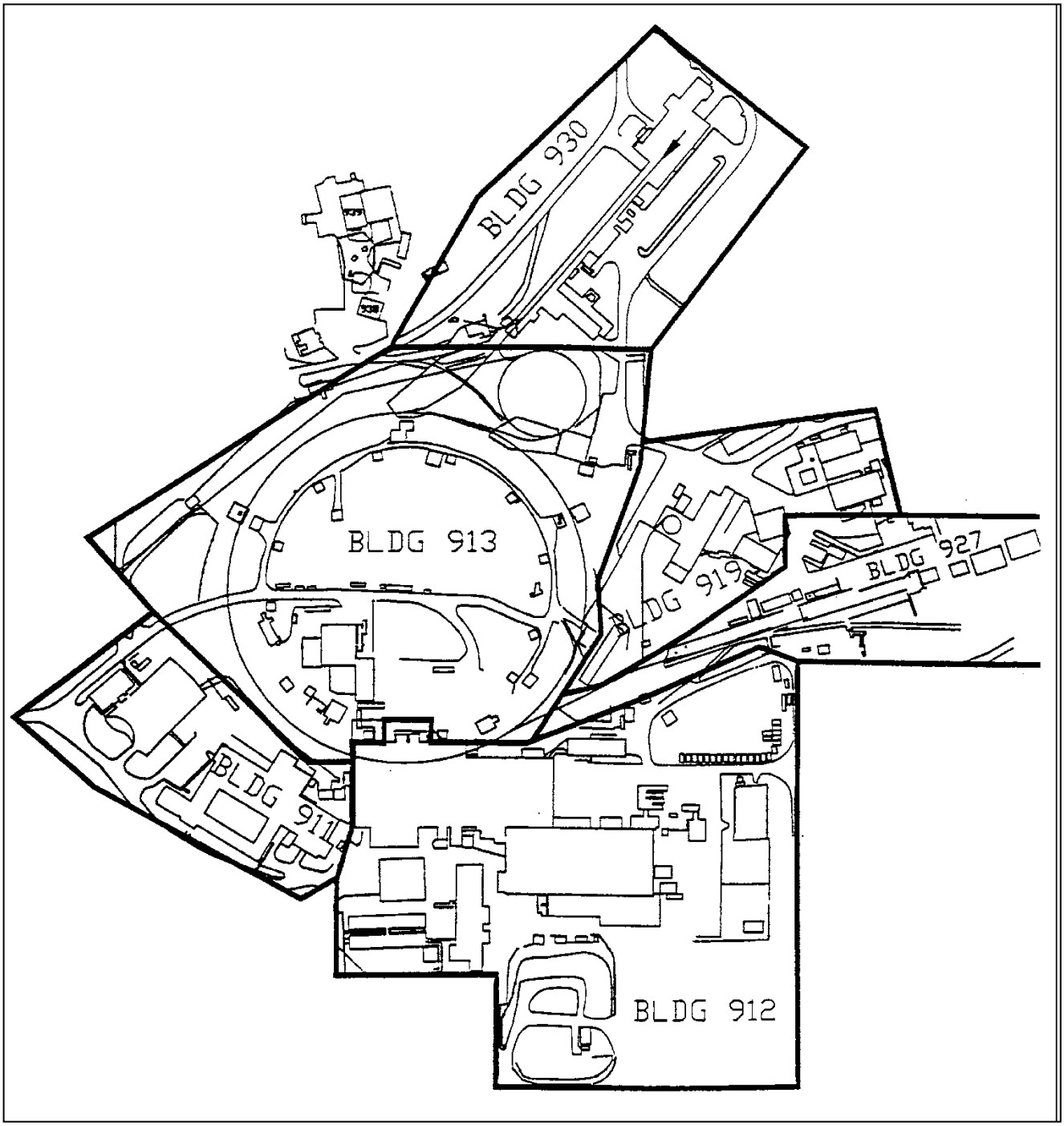

Figure 2-3. The AGS complex (BNL 1997h).

The Booster tunnel (Building 942) is directly adjacent to the AGS tunnel and the LINAC. It is connected to the LINAC and the Tandem Van de Graaff through the HEBT transfer tunnel. The original connecting tunnel is now used to transfer a beam from the Booster to the AGS. The Booster tunnel is 200 m in circumference and is buried under a 5-m thick berm.

The new section of the Booster tunnel was built in 1988. Some radioactive gases were created by activation of the ventilating air in the tunnel and experimental areas of the AGS. However, because the involved radionuclides are short-lived (^{11}C, half-life 20.5 minutes; ^{13}N, half-life 10.0 minutes; ^{15}O, 2.1 minutes), they did not create a measurable radiation level beyond the immediately adjacent onsite

area. All of the cooling water systems in the AGS and Booster tunnels are activated. This includes systems that use the AGS ring as a utility tunnel for piping that goes to the support buildings. In addition, there are activated components in the AGS and Booster rings, and activated spare parts are stored in the facility (BNL 1997h).

Other Related Buildings

Building 911. Building 911 is the main office for the AGS Department. It was one of the first structures built for the accelerator complex and has direct passageways to both the AGS tunnel (Building 913) and the AGS experimental area (Building 912). Building 911 also contains the main control room, the support building for the main magnet cooling system, the support building for the Westinghouse main magnet power supply (now the backup power supply for the ring magnets), a high bay assembly area with magnetic field testing shops, and other light electronics, vacuum, and machine shops. There are activated components from the AGS ring that have been stored and repaired in the high bay area and shop areas of Building 911 in posted radiation work areas. Because it also contains a system to cool water from the AGS ring, it does contain activated water (BNL 1997h).

Building 912. Building 912 is the designation of the approximately 5 indoor acres of AGS experimental floor. It is not one building, but five connected structures constructed over a 30-year period. The initial portion of Building 912 was started in 1958 to enclose both a portion of the AGS ring and the first experiments.

A majority of the AGS experiments, about 400, have occurred in this large area. The AGS experimental floor also contains a section of the AGS ring. Surrounding the beamlines are concrete and steel shielding needed to maintain radiation levels to within the required limits outside the lines. Shielding around the proton target stations that produce secondary beams is much thicker due to the high levels of radiation generated in the targets. This shielding can have significant activation in the body of the material. Cooling water at these target stations has higher levels of activation than the magnet cooling water and is handled by closed cooling systems. Radiation safety is provided by security systems that prevent access to these target stations and primary beamlines when protons are being extracted to these areas. There are activated components in the beamlines, including highly activated targets. Steel cutting, including activated shielding steel, has been performed inside and outside Building 912. Depleted uranium was used in experimental areas in the past, but its use and disposition were tightly controlled. Small radioactive sources (nanocuries) are used. They are inventoried and stored in locked shielding boxes (BNL 1997h).

Building 913A-E. AGS ventilation system fanhouses were an original part of the AGS ring and provided air conditioning and heating for the tunnel enclosure. Because of their proximity to the tunnel, they are designated High Radiation Areas when there is beam in the AGS ring and must be kept secured during operations. The cooling water for the air-conditioning system is activated and there is a potential for release in the fanhouse. In addition, air handled in the fanhouses is from the AGS tunnel air and can be activated (BNL 1997h).

Building 914. Building 914 was the location for the original 50-Mev AGS LINAC. This facility contained power supplies, water-cooling systems, pumps and compressors, and a motor generator set. During the 1970s and 1980s, after the LINAC was decommissioned, this building housed a mechanical maintenance group for the AGS division. Activated components were repaired and stored in this facility and new components were assembled and cleaned there (BNL 1997h).

Building 919 (g-2/Bubble Chamber and associated facilities). Building 919 was built in 1962 for an AGS experimental area and is still in use. The high bay area of Building 919 housed the 80–in. bubble chamber that was in operation until 1978. The bubble chamber had both a cryogenic component and a large mechanical component driven by a high-pressure oil hydraulic system. When

the bubble chamber reached the end of its run, the high bay area and control room were used as shops and assembly areas. The cryogenic shop continued to be used to support liquid H_2 target in Building 912 and the other buildings became storage areas. This continued to the late 1980s, when the area was modified for the g-2 experiment and the water-cooling systems were used to support Booster operations. A release of tritium from a gas chronograph detector occurred in 1973 in the technician machine shop. The detector held approximately 250 mCi of tritium (BNL 1997h).

Building 920. Building 920 is the EI0 Power Supply building, and is above the AGS ring. It houses an assortment of power supplies used under different beam configurations, but is presently running only a few supplies. The radiological hazard in Building 920 is radiation from activated cooling water and oil in transformers (BNL 1997h).

Building 928 (Siemens motor generator house). Building 928 is directly adjacent to and at one corner connected to Building 929. Both buildings share a basement area. The basement area contains support electrical equipment and lubricating equipment for the motor generator set. The basement also contains a mechanical equipment room with water-cooling equipment for power supplies in Buildings 928 and 929 and for systems in the AGS ring. There were potential radiation hazards from activated cooling water (BNL 1997h).

Building 949. This new building houses the proton target that produces secondaries for the g-2 ring. The water-cooled target is highly activated (>50 rem at contact immediately after the beam has been turned off). Building 949 is built into the side of the V-line and has extensive shielding. The beamline, which runs between the target building and Building 919, contains water-cooled magnets and vacuum pumps. The cooling water is shared with the target so it is activated (BNL 1997h).

Building 960. The building was the site of a large high-energy physics experimental facility, the Seven Foot Bubble Chamber, from 1972 to 1979. After the Bubble Chamber was removed from Building 960, the facility was used for storage and to test superconducting materials, in conjunction with a small cryogenic helium refrigerator, from 1979 to 1993. The testing of superconducting materials moved to Building 902 in 1993. From 1993 to 1997, the building was used for storage and then was abandoned. From a radiological standpoint, the particle beam delivered to the Bubble Chamber was very low intensity in relation to causing induced activity in the machine components (BNL 1997i, pp. 3, 6).

Buildings 912, 919, 975. The AGS Department has used warehouse space throughout the facility and the Laboratory. The AGS also stored materials outside, in the “Inner Mongolia” area of the AGS, the southwest area, and the ISABELLE experimental buildings. One of the major issues confronting AGS is the storage of activated materials. When a part fails in service in the accelerator tunnels or the experimental lines, it is often replaced with a spare. After a radiation cool-down period, the component is repaired and stored until needed (BNL 1997h).

Building 918 (Warehouse). This building is adjacent to Building 912. It was built in 1957 and 1958. It was a warehouse from the start and always used for that purpose. It was doubled in size in 1962. Activated components are stored in it (BNL 1997h).

Building 196 (Warehouse). This wood-frame building was built in 1942. It was formally abandoned and demolished in 1995. The building was used by the AGS Department for storage from 1975 until it was abandoned in 1991 to 1992. The building housed depleted uranium, lead, activated copper magnet coils, vacuum pumps with oil, and other activated components (BNL 1997h).

Building 209 (Warehouse). This warehouse belongs to the Safety and Environmental Protection (S&EP) Division. The AGS has been storing activated components (e.g., magnets, jacks and stands, and coils at <5 mR) in the building since 1995 (BNL 1997h).

Building 424 (Warehouse). Building 424 was built in 1942 for Camp Upton as its theater. The AGS began storing activated components and power supplies with polychlorinated biphenyl capacitors there from 1973 until it collapsed in 1996. After the building collapsed, it was demolished (BNL 1997h).

Building 178 (Warehouses). This building was built in 1922 as a theater and lecture hall. The AGS began storing activated components there in the early 1970s. Because the building was wood frame and in a "populated" area, the levels of activation were low and much of the equipment was new spares. The AGS equipment was removed and the building was demolished in 1992 (BNL 1997h).

2.4.8 Brookhaven LINAC Isotope Producer, Building 931

In 1972 the BLIP (Building 931B) and the CLIF (Building 931A) were constructed on a large earthen mound (BNL 1997j). The CLIF was operated independently by the Chemistry Department until 1977, at which time it was incorporated into the BLIP operation. Building 931C was constructed in 1996 as part of a substantial facility upgrade.

The BLIP was designed for production of radioisotopes by irradiation in the LINAC beam. It consists of a main building that houses a "hot cell" over a shaft that descends approximately 30 ft. The shaft allows targets containing various substances to be lowered along a track to an area into which a beam from the LINAC is conducted. Irradiated materials are withdrawn, transported to, and processed in the TPL (Building 801).

A smaller structure (Building 931A) houses the control panel for BLIP operations and monitoring, as well as a single fume hood and a second LINAC beam access system (currently not in use).

Building 931C acts primarily as a garage to house the forklift truck used to deliver irradiated targets to the TPL.

In the BLIP, targets are irradiated with 200-MeV protons. In this process, high-energy secondary neutrons are created. These penetrating neutrons are absorbed in the shielding soil around the main BLIP containment tank. This leads to some activation of the shielding soil (BNL 1997j, pp. 3, 4).

Building 946 houses the water-cooling equipment for the BLIP target area. This area was upgraded to provide better containment and shielding for this activated water system (BNL 1997h, p. 1).

2.4.9 Tandem Van de Graaff, Building 901

The Tandem Van de Graaff is housed in Building 901. It consists of two Van de Graaff machines, arranged so they can be used independently or in tandem, each capable of accelerating atomic particles to energies of 10 MeV. A special device that changes negative ions to positive ions makes it possible to achieve a maximum total acceleration of 30 MeV.

Building 901 is divided into laboratories, accelerator vaults, and office space. The building is the home of research that includes medical radionuclide research, accelerator target development, synthetic organic chemistry of radiotracers, and analysis of a variety of materials. Areas inside the building include the cyclotron area, the Radio-Frequency Quadrupole (RFQ)-Drift-Tube-LINAC (DTL) area, laboratory space associated with the Department of Applied Science, and a mechanical shop area that is currently occupied as a storage area. The west end of the building contains the Dynamitron, which has been shut down, and a robot room dedicated to the analysis of biological samples.

Area 1 – Cyclotrons. Two cyclotrons housed in this area are used for the production of radiopharmaceuticals. The radiopharmaceuticals are processed in the adjacent hot chemistry laboratory suite (Rooms 108 to 112). The cyclotrons are operated remotely from Room 118. These machines are used almost exclusively for producing the short-lived, positron-emitting isotopes used in the PET program. The radioactivity produced by these cyclotrons is remotely transferred, as liquids or gases, from the vaults to the adjacent hot laboratory. The hot laboratory contains preparation and purification equipment as well as shielded hoods for the processing of the radioisotopes from the cyclotrons into radiopharmaceuticals (BNL 1997k, p. 5).

Area 2 – RFQ-DTL Accelerator. The BNL RFQ-DTL facility consists of the AccSys RFQ accelerator, associated electronics, and other support systems. The areas associated with the accelerator include the vault (Room 130), where the accelerator is located, and the control room (Room 129), where the power supplies and control console are located. Since 1996, this facility has occupied space formally used for the 3.5-MeV Van de Graaff (BNL 1997k, pp. 4, 5).

Area 3 – Laboratory Space. This area is laboratory and office space devoted to DAS projects. Most of the space is used for data analysis and storage of materials to be analyzed at the NSLS. The former machine shop is used for storage of samples and unused electronic equipment associated with the 3.5-MeV Van de Graaff (BNL 1997k, p. 5).

Area 4 – 901W Dynamitron and Robot Room. This room in the southwestern corner of Building 901W is dedicated to the operation of a robot used by the BNL PET program to analyze plasma samples taken from subjects during the PET study. The rest of the area in Building 901W is occupied by the Dynamitron (BNL 1997k, p. 5).

Most of Building 901 was constructed in 1948 and 1949. It was originally constructed to house the cyclotron(s) and the 3.5-MeV Van de Graaff. The 3.5-MeV Van de Graaff was installed in the building in 1950.

In 1958, Building 901W was constructed to house the 18-in. cyclotron, which, in 1959, was moved from the test shack in the rear of the Cosmotron and reassembled in the new building. The 18-in. cyclotron was capable of accelerating protons to energies of about 3 MeV. From 1960 to 1963, this accelerator was used to study elastic and inelastic scattering of polarized neutrons. The target material for the production of the neutrons was a T(p,n) ^{3}He reaction on a target of zirconium tritide. The accelerator was capable of consistent beam currents of 100 µA and occasionally reached beam currents of 200 µA.

The 60-in. cyclotron was installed in 1950 and became operational in 1951. The 60-in. cyclotron was capable of producing 10-MeV protons, 20-MeV deuterons, and 40-MeV alpha particles. It was used to produce a variety of radioisotopes for use in medicine, biology, chemistry, and physics. Upgrading of the 60-in. cyclotron began in 1964 and was completed in 1968. The upgrade project resulted in the cyclotron becoming a variable-energy machine capable of producing 35-MeV protons, 23-MeV deuterons, 57-MeV ^{3}He and 46-MeV alpha particles. During the late 1950s and early 1960s, a number of radioisotopes were in development and/or production at what was known then as the Low Energy Accelerator Facility (LEAF), which included the cyclotrons and the Van de Graaff accelerator. The isotopes were ^{90}Y, ^{47}Ca, ^{67}Cu, ^{38}Ar, ^{28}Mg, ^{124}I, ^{133}I, ^{47}Sc, ^{210}Po, ^{83m}Kr, ^{68}Ga, ^{151}Sm, ^{132}Te-^{132}I generator, and ^{99}Mo-^{99m}Tc generator. The major purpose of these isotopes was medical research.

In 1965, the Dynamitron was installed in Building 901W. It was used as a positron accelerator until 1989 to study positron reactions. During this time, it used a large ^{22}Na source as the generator of positrons. This source was removed in 1996. At present, the facility is not in operation and there are no immediate plans for a restart. All activated items have been removed.

The Van de Graaff was installed in Building 901 in 1950. There were some problems during the early years and a reconstruction project was completed in 1954. In 1960 the first experiments with a ^{3}He beam were begun. The first test of the triton beam was carried out in June 1968. During 1968 and for a few years after, ^{28}Mg was being produced routinely on the Van de Graaff using the ^{26}Mg(t,p)^{28}Mg nuclear reaction. During the period from 1968 through the late 1970s, the 3.5-MeV Van de Graaff was being used to accelerate tritium for production of isotopes as well as basic physics studies. In 1985, the Van de Graaff was turned over to the DAS. Some experiments with tritium continued until the shutdown of the Van de Graaff in 1991 (BNL 1997k, pp. 6, 7).

2.4.10 National Synchrotron Light Source, Building 725

Building 725 houses the NSLS. It is one of the most intense sources of X-ray, ultraviolet, and infrared radiation in the world. The beams of radiation are created by electrons circulating in two storage rings. The X-ray ring energy is 2.5 GeV and the VUV ring energy is 800 MeV. The radiation produced by the circulating electrons travels down 85 independent beamlines to targets on a variety of experiments. There are also wet laboratories available for the experimenters, storage space, offices, and a user machine shop in the building.

The sole occupant of Building 725, which was constructed in 1981, has been NSLS. In 1987, a major Phase II expansion of the facility (office space, beamlines, laboratory, and setup space) was added, but the original function of the facility remained the same. Operations of the VUV ring commenced in 1982, and the X-ray ring began operations in 1984 (BNL 1997l, pp. 1, 2).

Building 729 is the NSLS Source Development Laboratory. The building houses a LINAC coupled to a magnetic “wiggler” that produces an intense X-ray beam (BNL 1997m, p. 1).

2.4.11 Relativistic Heavy Ion Collider Complex, Building 1000 Series

The RHIC Complex is shown in Figure 2-4. It is a high-energy particle accelerator that was commissioned in 1999 and achieved its first successful operation in the summer of 2000. The RHIC is designed to achieve much higher reaction energies by colliding two accelerated beams head on. When the ion beam is traveling at top speed in the AGS, it is taken down another beamline called the AGS-To-RHIC (ATR) transfer line. At the end of this line, there is a “fork in the road,” where a switching magnet sends the ion bunches down one of two beamlines. Bunches are directed either left to the clockwise RHIC ring or right to travel counter-clockwise in the second RHIC ring. From there, the counter-rotating beams are accelerated, as in the Booster and AGS, and then circulate in RHIC where they collide into one another at as many as six interaction points. These intersection points are described in Figure 2-5.

The chronologies of the RHIC buildings (BNL 1997n, pp. 4–8) are:

Building 1002

1981–present	Experimental hall, assigned to the Broad Range Hadron Magnetic Spectrometers (BRAHMS) experiment
1982–1992	AGS storage, some activated components
1990–1992	AGS experiment 813

Building 1004A (Service Building)

1981–present	Utilities, no process usage before 1992
1992–present	RF system testing

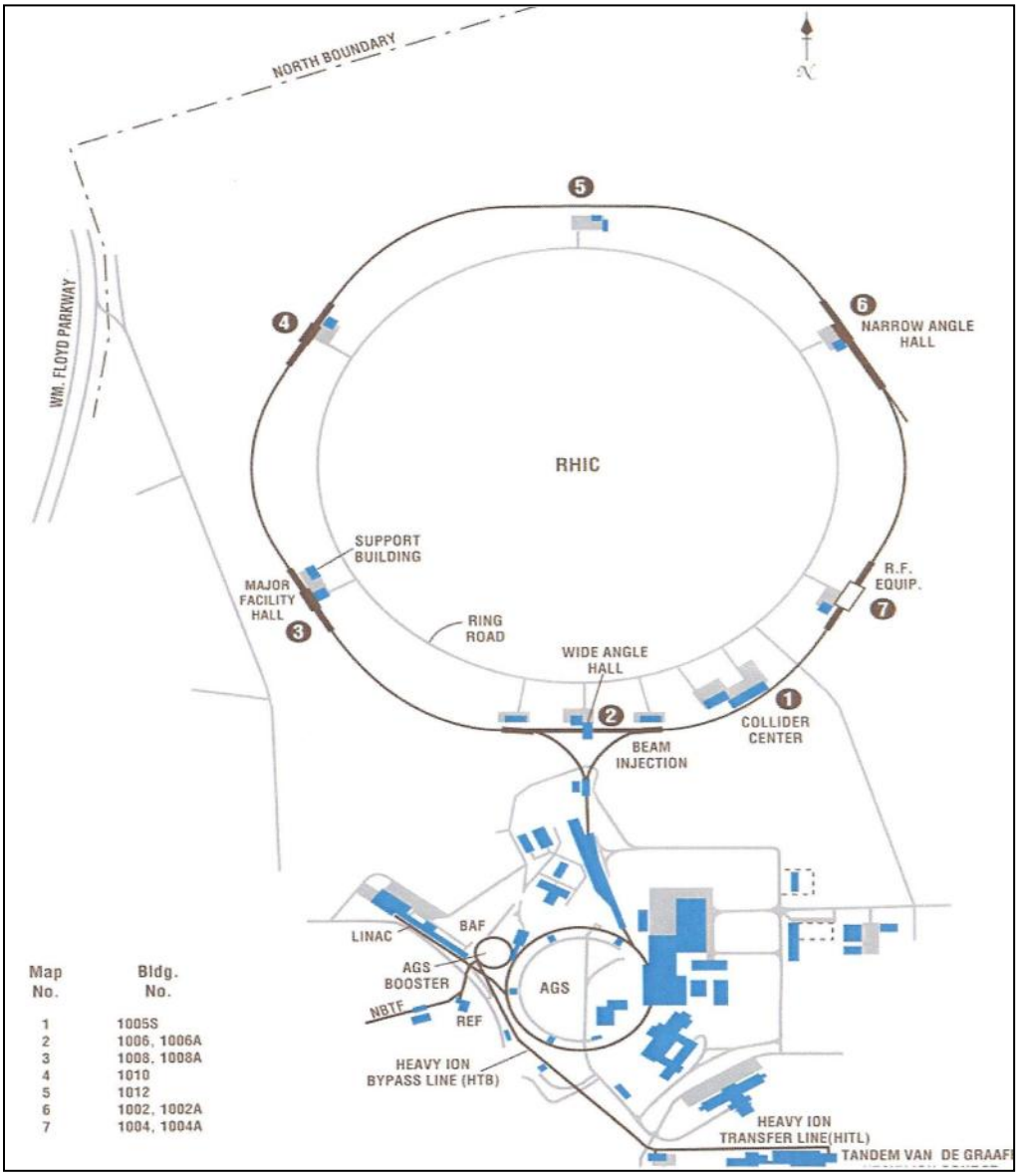

Figure 2-4. The Relativistic Heavy Ion Collider complex.

Building 1004 Open Area
1981–present Future experimental area
1994–present RF cavities testing for collider
1996 -present Temporary beamstop for Sextant Test completed during January 1997

Building 1004B (Support Building)
1994–present Power supplies, cryogenic valve boxes, other system electronics

Building 1005 Beam Tunnel
1981–present Commissioned in January 1997 as part of the Sextant Test

Building 1005S
1981-present Tech Shops on first floor, administrative offices on second, third, and fourth

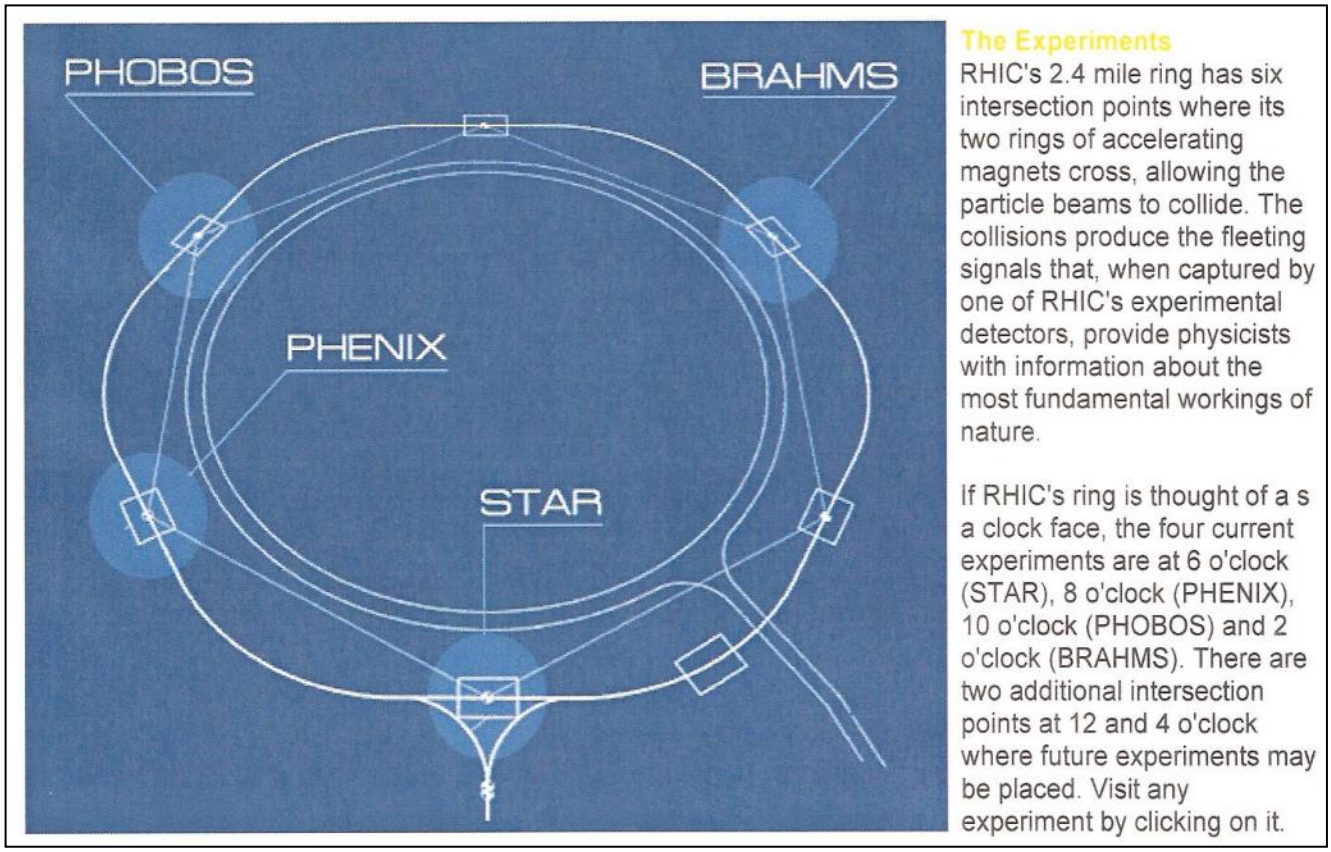

Figure 2-5. RHIC ring intersection points (BNL 2009b).

Building 1006

1991–present	Solenoidal Tracker (STAR) Detector
1985–1986	Machining of low-level radioactive steel for AGS Experiment 787
1981–1990	Other short-term uses, temporary storage

Building 1006A (Service Building)

1981–present	Experimental support, technical shop, utilities, and center room occasional storage

Building 1007

1981–present	Beam tunnel to be commissioned in 1999
1981–1995	Part of tunnel used as a calibration laboratory for Survey Group

Building 1008

1991–present	Pioneering High Energy Nuclear Interaction Experiment (PHENIX)
1992–1995	Injection magnet fabrication
1988–1991	Uranium plate machining for the DØ [D-zero] Experiment at the Fermi National Accelerator Laboratory

Building 1008A

1981–present	Experimental support building, utilities
1992–1995	Injection Magnet Group Shop/South Room
1990–1992	AGS/Physics Experiment Preparation/South Room
1986–1987	Assembly of wire chambers for AGS Experiment 755
1985–1986	Cleaning of depleted uranium with trichloroacetic acid for European Organization for Nuclear Research (CERN) Experiment NA-34

Building 1010
1994–present PHOBOS Experiment[2]

Building 1012
1994–present Future Experimental Hall

2.4.12 Hot Laboratory, Building 801

Building 801 has from its inception in 1949 been known as the Hot Lab. It consisted of a central laboratory, a fanhouse, a radioactive liquid waste tank farm, and a liquid radioactive waste concentration plant. The original purpose of the central facility was to provide appropriately shielded areas for research with large amounts of radioactive material. The building was designed as two separate but connected structures: a "Hot" area (the western portion) and a "Cold" area (the eastern side). The Hot area of the Hot Laboratory included five hot cells, three chemical-processing hot cells, and three high-level hot cells for handling and processing of relatively high levels of radioactivity in gaseous, liquid, or solid form. The cells were maintained at negative pressure in relation to their surroundings to minimize the possibility of radioactive releases to the building. Each has individual exhaust air filters as well as a backup filter preceding discharge to the BGRR stack.

Currently in the Cold area there are a number of chemistry laboratories that conduct basic research. On the Hot side, radioisotope processing and production (related to the BLIP), and radiopharmaceutical research is conducted by Medical Department staff.

There are Metallurgy Hot Cells [the Metallurgical Evaluation Laboratory (MEL)] in the Semi Works Area (Area 56) that were constructed in 1958. Metallurgical studies on highly radioactive reactor components were performed in this facility by Department of Applied Technology personnel.

There are "D" and "F" waste systems (storage tanks and piping) in the building basement. The D waste system holds radioactive nonhazardous liquid waste. The F waste system holds nonradioactive, nonhazardous liquids. The Hazardous Waste Management Group of S&EP manages that facility.

Some additional special facilities in the building are the TPL (a series of six hot boxes in a separate, radiation-monitored area (Room 66), the MEL, and three inactive Hot Cells. The 300-ft stack south of the building receives the ventilation from hoods on the hot side. The stack is monitored for radioactivity (BNL 1997o, p. 4).

There was a ^{60}Co irradiation pool in what is now the TPL (Laboratory 66). The pool, which was constructed of concrete, had depths of 4, 10, and 14 ft where large amounts of sealed ^{60}Co sources were stored under water. There was a drain to D waste in the deep portion of the pool. The pool was in operation from about 1952 until about 1960. The ^{60}Co pool was moved, water was drained, concrete was decontaminated, and the pool area was backfilled with sand (BNL 1997o, p. 7).

From March 1952 to June 1960, there was a program at the Hot Laboratory to produce ^{131}I by acid-dissolution of irradiated uranium samples. A separate exhaust system, which included NaOH scrubbers and backup charcoal filters, was installed in the hot cell and ducted to a stainless-steel pipe in the BGRR stack.

There are pneumatic tubes in the roadway connecting the BGRR to Building 801. These lines allowed quick transfer of short-lived isotopes from the reactor to Building 801 where they would be

[2] The name PHOBOS, which is a moon of the planet Mars, came about because the first proposed experiment was deemed too expensive and needed to be downscaled. It was called Modular Array for RHIC Spectra (MARS).

used. An interviewee reported that at times the radioactive sample carriers, known as rabbits, would break or get stuck in these tubes.

Incidents
On May 15, 1957, a serious incident occurred involving the reprocessing of uranium for the Volatility Project. An explosion involving UF_6 and BrF_3 injured a few people and one was hospitalized. There was a release of about 13 lb of unirradiated uranium to the local environment. Damage occurred to nearby trees, automobiles, and building equipment. Environmental sampling of the appropriate areas might identify residual uranium contamination (BNL 1997o, p. 5).

In August 1960, some highly radioactive waste (reactor fuel elements) was dissolved in aqua regia and stored in plastic containers in the storage vault (Room 51). These containers cracked and leaked onto a stainless-steel tray containing the plastic containers. The solution was sucked up using an aspirator connected to the house vacuum system. The vacuum system became highly radioactive. The main vacuum chamber removed all standing liquid in the lines. However, because of the high contamination level and long half-life of the activity, the pipes remain residually contaminated. The system was moved to a remote location, but some of the piping in the floor is still contaminated as a result of this incident. The floor is labeled as a radioactive material area (BNL 1997o, p. 7).

2.4.13 High Intensity Radiation Development Laboratory, Building 830

Building 830 is an office and laboratory complex presently occupied by Department of Applied Technology and RHIC personnel. All of the RHIC personnel are housed in a modular structure, which is all office space. The laboratory complex has always been used as an extensive experimental area, especially involving the use of radioactive and hazardous chemicals. This facility houses the Gamma Irradiation Facility (GIF), two inactive hot cells, analytical laboratories, electron microscopy, and associated offices.

In 1963, Building 830 commenced operations as the HIRDL. At that time, it consisted of the high bay area (which houses the two hot cells), laboratories, and offices. Cell 1 (Preparation Cell) was used to fabricate high-intensity ^{60}Co sources for food irradiation programs. Some of the sources were then moved to Cell 2 (Irradiation Cell) by means of a transfer canal. In the Irradiation Cell, the sources were stored in a 21-ft-deep water-filled pit in the cell until needed. Items to be irradiated were moved in and out of the Irradiation Cell by means of two transfer tunnels under the cell. Sources were also kept in the two bays for storage and irradiation purposes.

In 1970, the Low Dosimetry Facility (currently known as the GIF) was added to the northeast end of the main building. This facility included the gamma irradiation pool and a machine shop. Sources were stored at the bottom of the pool, and samples to be irradiated were lowered through air-filled tubes. In 1974, the last of the sources was removed from the pit in the Irradiation Cell. Since that time, the cells have been used for experiments with lower activity level materials and storage of radioactive materials and contaminated equipment.

At present, the two hot cells in Building 830 have no mission-essential work. The Laboratory spaces are being used for experiments, and the GIF currently contains approximately 35,000 Ci of ^{60}Co (BNL 1997p, pp. 2–3).

2.4.14 Chemistry Department, Building 555

Building 555 is the home of BNL's Chemistry Department. It was built between 1963 and 1966. The only major modification was made in 1996 when the underground Center for Radiation Chemistry Research was added. The building contains three floors plus a partial basement and loft. There are many extra-large laboratories with at least two fume hoods per laboratory. There are pipe chases

between rows of laboratories to deliver all services to the laboratories. Transparent glass piping in the basement carries wastewater from the sinks in the laboratories to the sewer system. Counting rooms constructed of specially selected low-activity materials are used for low background counting. There is a large chemical stockroom with separate storage rooms for acids, bases, and organic solvents, as well as a room devoted to the washing of laboratory glassware.

A large variety of research programs has been conducted during the past 31 years. Many different chemicals have been used, including inorganic acids and bases, inorganic compounds and metals, as well as a plethora of organic and organometallic compounds. A variety of specialized equipment and instrumentation is in use in different laboratories (e.g., there are several laser-based systems in operation).

In addition, research with radioactive isotopes has been done since the earliest days. Studies include chemical kinetics, hot-atom effects, nuclear chemical studies of nuclear decay schemes, nuclear reaction cross-sections, solar neutrino studies, and radiocarbon dating. Over the years, the mix of research projects in the Department has changed considerably, but virtually every chemical element in the Periodic Table has been used at one time or another in the building (BNL 1997q, p. 5).

Some radiological problems associated with the building are:

- Disposition of sealed ^{60}Co irradiation sources in Laboratories 171 to 173. These sources are in stainless-steel containers that penetrate the laboratory floors and extend into the ground below. There is evidence that one 40-Ci source was leaking small amounts of ^{60}Co into the water that surrounds it to provide radiological shielding. This source has been removed and disposed of. Some of this contaminated water has gotten into the soil surrounding the buried containment vessel. The amount of ^{60}Co in the water and the soil is small. (The estimated ^{60}Co activity remaining in this soil is less than 35 nCi.)

- Radioactive materials and counting samples are used and/or stored in many rooms: Laboratories 225, 227, 229, 233, 235, 236, 253, 367, 369, and 374 and counting rooms 103, 203, 204, and 205. Periodically, chemical manipulations are done with these materials in vented hoods that are posted as radioactive work areas in accordance with BNL procedures. Occurrence reports from 1990 to present were reviewed for Building 555. They indicated three separate reportable incidents involving small amounts of radioactivity. Two of these, one involving ^{32}P and the other ^{238}U, were confined to the chemistry laboratories in which they occurred. The third involved a Kimwipe contaminated with ^{124}I that was inadvertently thrown into the trash and was subsequently detected in the garbage truck before it left the BNL site (BNL 1997q, p. 9).

2.4.15 Physics Department, Building 510

Building 510 was built in 1962 to house Physics Department personnel and laboratories. It consists of approximately 400 offices and 100 laboratories in an area of about 200,000 ft^2. There is a large machine shop to support the research programs as well as a number of smaller machine shops. A high bay area is available for assembling equipment for experiments. The basement is used predominantly for storage but also houses the building support facilities such as heating and air-conditioning units and power distribution systems. The northeast parking lot contains 11 trailers permanently placed for storage of solid materials. These are not climate controlled and no chemical or radioactive materials are stored there (BNL 1997r, p. 6).

The northeast corner of the building houses the Health Physics Section (Rooms 1 to 136) where there are 35 vaults for storage of radioactive materials. There are nine Type A vaults that are 12 in. in diameter and penetrate 10 ft into the ground, six Type B vaults that are 8 in. in diameter and penetrate

10 ft into the ground, and 20 Type C vaults that are 8 in. in diameter and penetrate 3 ft into the ground. Vault 17 is contaminated because it contained a leaking radium source. The source has been placed in a sealed polyvinyl chloride pipe and is now stored in Vault 4 (BNL 1997r, pp. 4, 5).

2.4.16 Other Miscellaneous Radiological Research Facilities

2.4.16.1 Liquid Metal Fuel Reactor Support, Building 820

R&D on Liquid Metal Fuel Reactor (LMFR) technology took place in Building 820 from 1957 to 1975. A simulated reactor with a core of uranium dissolved in bismuth was studied (BNL 1997s, pp. 5, 55–64). The simulated reactor was a utility test loop in a figure-eight design with molten bismuth-uranium metal in one loop and molten bismuth in the other loop. The bismuth-uranium metal included uranium at 1,000 ppm. The test loop was made of a 4-in.-diameter stainless-steel tube. The simulated reactor was heated by a 5-MW oil-fired boiler. Small amounts of fission products were generated in the experiments from the ^{235}U in the simulated reactor. The support systems were used to study thermodynamic properties and removal of fission products from molten mixtures using nonradioactive salts of fission product elements. The facility also used X-ray machines to test welds in the experimental structures.

2.4.16.2 Hot Machine Shop, Building 530

Building 530 functioned as a Hot Machine Shop for the Central Shops Division. In 1975, Central Shops evacuated the building and relocated its equipment and function to Building 462. The building was then used as a storage area by Plant Engineering until 1988, at which time it was demolished (BNL 1997t, p. 3).

2.4.16.3 Health Physics and Safety Instrumentation, Buildings 535 and 356

Building 535 was constructed in 1962 and 1963 and was operational in 1964. The building is a multitenant, multifunctional facility consisting of a ground floor and a basement. The Instrumentation Division, which is responsible for the building, occupies a major part of the ground floor for R&D of X-ray and neutron detectors and corresponding electronics. These detectors and associated electronics are used at the NSLS, the HFBR, the AGS, and other worldwide scientific research facilities. A state-of-the-art electronic printed circuit design and fabrication facility and a microelectronics clean room facility are unique to the Laboratory. The S&EP Division's Analytical Services Laboratory was in the basement of Building 535 until the 1990s, when it relocated to Building 490. Personnel Monitoring was also in this building until 2002, when it relocated to Building 490. Personnel Monitoring processed film badges and thermoluminescent dosimeters (TLDs) in the building (first floor). The building also housed the equipment needed to operate the external monitoring program (X-ray machine, darkroom, densitometer, microscopes, etc.).

In 1978, the NSLS Department moved into the rear section of the basement and has occupied that space since. An underground walkway (tunnel) was constructed in 1986 between the basement of Building 535 and the main floor of Building 725 (NSLS Department).

In 1985, Instrumentation Division established the Optical Metrology Laboratory in the southeast section of the basement to conduct research for synchrotron radiation experiments. In 1987, an office module was added to the southwest corner of the building (BNL 1997u, pp. 5, 6).

Another building operated by the instrument division was Building 356 (BNL 1997y, BNL 2006a). From 1960, an irradiation facility using ^{60}Co sources was used to study the irradiation effects on semiconductor materials and devices, electronic systems, and optical materials. In 1986, ownership

of the facility was transferred from Physics to the Applied Sciences Department, and in 1991 a 20,000 Ci ^{60}Co source was installed.

2.4.16.4 Biology Department, Building 463

Building 463 was constructed in 1942 with several later additions to house the Biology Department (BNL 1997z). The building contains offices, wet laboratories, instrumentation rooms, and rooms for ancillary functions such as seminars, conferences, and a library. A Controlled Environmental Radiation Facility (CERF) is located in the sub-basement that houses strong gamma sources used for irradiation experiments. High activity ^{60}Co and ^{137}Cs sources have been in use in Building 463 since 1951 (BNL 1997z, BNL 2006b).

2.4.16.5 Radiation Calibrations Facility, Building 348

Building 348 has been used by the S&EP Division since 1959 as a facility to repair and calibrate radiation detection instruments. Standard encapsulated radiation sources are used to provide beta, gamma, and neutron radiation fields for instrument calibration. In about 1997, a replacement facility was constructed next to the original building, also designated as Building 348 (BNL 1997aa, BNL 2006c).

2.4.17 Waste Management Facility, Radwaste Reclamation Building, Building 865

The WMF was opened in December 1997 for managing the wastes from BNL's research and operation activities. The new WMF replaced the original HWMF in its entirety, and its design ensures that all storage and transfer activities are accomplished inside buildings and on paved and curbed areas. A security fence around the cleared area and a berm 8 ft high around the perimeter of the radioactive waste portion of the facility ensures that incidental exposures meet the BNL design criteria of 25 mrem/yr to nonradiation workers.

The WMF includes the Operations Building (Building 860), Resource Conservation and Recovery Act Waste Building (Building 855), Radwaste (Radioactive Waste) Reclamation Building (Building 865), and Mixed Waste Building (Building 870).

The Radwaste Reclamation Building (Building 865) is the primary facility for radioactive waste handling, size reduction, and repackaging for subsequent offsite disposal for BNL. The goal is to size reduce the radioactive waste stream generated at BNL. This building receives bulk radioactive waste of various sizes and configurations to be disassembled, decontaminated, size-reduced, and packaged for temporary storage before shipment off site. To achieve the goal of waste reduction, Building 865 houses a 500-lb/d lead smelter, a trash compactor, and a shredder (BNL 1997v, pp. 2, 3).

Additional radioactive waste treatment facilities include:

- Tritium Evaporator Facility (Building 802B). Wastewater processing began in 1995. This facility was constructed to reduce the total amount of tritiated water released to the Peconic River. Another room contains fans that exhaust Building 801 air to the stack.

- Waste Concentration Facility (WCF, Building 811. This facility was designed in the late 1940s and constructed in 1950 to support the radiological waste stream from the Nuclear Engineering Department and Medical Department in the 801 and 701 complexes. Liquid waste, which contains residual radioactive material that is generated on the site, is processed at the Waste Concentration Facility (WCF), Building 811. At the WCF, suspended solids are removed from the liquid along with a high percentage of radionuclides using a reverse osmosis process. The only radionuclide that is not removed during this process is tritium. The tritiated

water that remains following the waste concentration process is delivered to the Evaporator, where it is converted to steam and released as an airborne effluent.

- Waste Management Incinerator. The Laboratory incinerates certain low-level radioactive wastes at the HWMF incinerator, Building 444.

2.4.18 Radiological Waste Decontamination Facility, Building 650

The Waste Decontamination Facility was constructed in 1957 and occupies the western half of Building 650. It replaced the yard decontamination operations at Building 446. Decontamination of large pieces continued until the 1980s. The process slowed and is now idle.

The facility consists of an office, showers, decontamination showers, airlock, and high bay. The decontamination showers were once used for decontamination of equipment. They drain to the basement then to the D waste tanks (removed). The high bay houses a lead melter (contaminated vent pipe; out of service), two vapor blaster talc/water abrasive blasters (internally contaminated; out of service), one walk-in shot blaster (internally contaminated; out of service), sample hood, and limited storage (one B-25 container holding film badge records at the time of the walkthrough). Already removed were two wet decontamination tanks (a 20- by 4- by 4-ft top tank used for acid, base, and detergent wash).

Associated with the Waste Decontamination Facility is an outside decontamination pad. From 1957 to 1981, the 20- by 20-ft pad was used for washing surface contamination from large pieces of equipment. Drainage from the pad used to go to the underground storage tanks (now removed). The surface of the pad has low-level radiological contamination. About 1994, an asphalt cap was laid by the Office of Environmental Restoration (OER) to contain the radiological materials. Soil in the vicinity of the pad was characterized by OER in 1993 as containing ^{137}Cs, ^{152}Eu, ^{154}Eu to a depth of 0.5 ft exceeding OER cleanup goals, and the soil is in the OER remedial plan. The low-level contaminants are fixed in the soil and are unlikely to migrate.

From 1959 to 1996, the Plant Engineering portion of the building was used as a laundry facility for contaminated and clean laundry. In 1996, laundry services were subcontracted and the facility is now being used for storage of custodial supplies. Building 650T has been used as offices since it was first acquired in 1994 (BNL 1997w, pp. 4-7).

2.5 ISOTOPES OF CONCERN

Table 2-1 identifies the potential radiological contaminants associated with the specific nuclear operations at BNL in terms of isotopes of concern. Due to the wide variety of research activities in a single multiple-laboratory building and a significant number of programs that handle or create practically all of the isotopes of the elements in the Periodic Table, the isotopes of concern are identified forensically from the airborne radionuclide releases by the facility. This method is considered adequate for the purpose of this section of the BNL site profile—that is, to provide background information only.

In the absence of measurements or studies, NIOSH guidance requires the use of default solubility classes and particle size values from the International Commission on Radiological Protection (ICRP; NIOSH 2002, pp. 15, 16).

2.6 MAGNITUDE OF SITE ACTIVITY

Table 2-1 also provides a perspective of the magnitude of the nuclear operations at BNL. ORAUT-PROC-0031, *Site Profile and Technical Basis Document Development* (ORAUT 2011a), requires the

magnitude of the operations be expressed in radioactivity (Ci) of the isotopes of concern. The same information is also required in Section 5.0, "Occupational Internal Dose." Therefore, radioactivity values in Section 5.0 should be used for dose reconstruction. In cases where available information does not allow the estimation of the radioactivity at the time of the operations, the activity fraction of each isotope is listed. Activity fractions in the Section 5.0 table are the ratios of the activity of the individual released isotope and the total released activity of all the isotopes. In cases where numeric data are not available, the table provides descriptive text that gives a perspective on the magnitude of the radiological impact on personnel.

2.7 MAJOR INCIDENTS

Table 2-2 describes major site incidents that might have significant potential for internal or external exposure to personnel. The records include many incident investigations with estimates of dose to individuals. Most of these investigations involved localized contamination without release to the environment.

2.8 RADIOLOGICAL ACCESS CONTROLS

Radiological access controls have been exercised locally at the site where activities involving radioactive materials have been performed. Radiological Control Areas have been established for each specific laboratory, facility, or building in which nuclear operations have been conducted at the BNL site. Entry to those areas has been controlled (i.e., hand-and-shoe counters, dosimeters, and training requirements).

Table 2-2. Major incidents.

Facility/ building	Year	Description
BGRR, Building 701	1952–1957	During the early years of operation of the BGRR, fuel failures occurred that resulted in radioactive materials being released to the air stream that cooled the reactor. There were 28 reported ruptures of BGRR fuel from 1952 to 1957. These all occurred with natural uranium. There was one rupture of a uranium oxide (U_3O_8) sample that was being irradiated for the radioiodine production program. Aside from Ar-41, I-131 was the most important radionuclide that would contribute to a potential dose that was discharged to the atmosphere from the BGRR. Bromine-82 and I-133 were released in somewhat larger concentrations (Meinhold and Meinhold 2001).
BGRR, Building 703	12/20/1975	Contamination of Laboratory W-9 in Building 703 occurred during loading of a cerium sample. The contamination spread was identified and contaminated areas were decontaminated. Contaminated workers were provided bioassay monitoring. No release to the environment was identified (Roesler 1976).
Hot Laboratory, Building 801	5/15/1957	There was a serious incident involving the Volatility Project where uranium reprocessing was the objective (Eisenbud 1957). There was an explosion of UF_6/BrF_3 and a few people were injured, one hospitalized. There was a release of about 13 lb of unirradiated uranium to the local environment. The UF_6 combined with moisture to form an oxide. The BrF_3 was very corrosive and damaged nearby trees and automobiles and equipment in the building (BNL 1997o).
	8/1960	There were some highly radioactive waste reactor fuel elements dissolved in aqua regia in plastic containers in the storage vault (Room 51). These containers cracked and leaked onto a stainless-steel tray containing the plastic containers. It was decided to suck up the solution with an aspirator connected to the house vacuum system. The vacuum system became very highly radioactive. The main vacuum chamber removed all standing liquid in the lines. However, because of the high level and long half-life of the activity, the pipes remain residually contaminated. Thereafter, the system was moved to a remote location but some of the piping in the floor is still contaminated; the concrete is

Facility/ building	Year	Description
		labeled as a radioactive area (BNL 1997o).
BLIP, Building 931B	5/16/1978	A target at the BLIP facility vaporized causing radioactive material to be vented to the work environment and the general public. Bioassay monitoring was provided to two exposed individuals (Miltenberger 1978a). The estimated dose to the two workers was 0.275 rem and 1.350 rem to the thyroid, and the dose to the public was estimated to be 0.060 rem to the thyroid at the site boundary for full occupancy (Miltenberger 1978b).
HFBR, Building 750	3/18/1983	A rupture of the moderator chamber in the CNF occurred due to a rapid reaction of hydrogen and oxygen in the chamber. The incident resulted in loss of experimental time for researchers using the HFBR beam port H-9. No one was injured, and there was no impact to the public (BNL 1983).
	3/31/1994	On March 31, 1994, smoke was detected emanating from the TRISTAN Experimental facility. The TRISTAN (Terrific Reactor Isotope To Analyze Nuclides) was an on-line isotope separator located on the experimental level at beam hole H-2 of the HFBR. Radioactive contamination in the air on the experimental level was detected. The reactor was shut down and the reactor building was evacuated. The portal monitors detected contamination on 7 persons as they left the reactor building. The fire was most likely caused by electrical failure in the high current filament leads. The fire resulted in contamination of the reactor containment building with a small release to the outside (BNL 1994).
	1997	During a scheduled maintenance shutdown in 1997, a leak in HFBR's spent fuel storage pool was discovered. In November 1999, the Secretary of Energy made a decision to close the HFBR permanently (BNL 2001a).
HFBR, Building 750 (continued)	9/27/1999	Two gallons of D_2O drained into a sump on the equipment level, partially filling it with D_2O. The portable tritium monitor spiked at 1000 microcuries/m3. The off-gas system evaporated the liquid. The three workers involved in the incident received estimated thyroid doses of from 71 to 265 rem from radioiodine isotopes, and the maximum estimate thyroid dose to an individual at the site perimeter was 60 mrem (BNL undated a)

3.0 OCCUPATIONAL MEDICAL DOSE

3.1 PURPOSE

Work funded by the U.S. Atomic Energy Commission (AEC) began at BNL in 1948. BNL occupational medical physicians prescribed chest X-rays as part of the preemployment and periodic physical examination of employees from the beginning of operations (Sunderman 1947a; Cowan 1948). These radiographs caused exposure of the lungs and other organs and tissues of the body. Exposure of the radiographed workers came mostly from the primary X-ray beam and from scattered radiation.

The purpose of this section is to describe the occupational medical X-ray screening program over time at BNL and to provide organ and tissue doses associated with these exposures.

3.2 SCOPE

Substantial information about the historical BNL occupational medical X-ray program has been found. The current medical director and two BNL X-ray technicians were interviewed and some older X-ray films were reviewed to gain insight into the occupational medical program and collimation practices (Morris 2006a,b,c,d). A peer-reviewed publication (Handloser and Love 1951) provided exposure data specific to the BNL radiographic equipment and techniques in use in 1950. A transcript of interviews conducted in preparation for an epidemiology study (Brodsky 1964) included discussions with BNL's first medical director about the early radiographic examination practices, equipment, and exposure rates.

3.3 EXAMINATION FREQUENCY

The occupational medical program at BNL began in 1947. Sunderman (1947a) stated that the recommended X-ray screening protocol included a preemployment fluororoentgenogram (otherwise known as photofluorography or PFG), a preemployment anterior-posterior (AP) and LAT lumbar spine, and an X-ray of one forearm. The recommended annual physical included a PFG; and the recommended termination examination included a PFG, AP and LAT lumbar spine, and the same forearm radiographed in the preemployment examination. It was also recommended that beryllium workers under contract with the AEC receive a chest X-ray at preemployment, 6-month intervals, and termination. These recommendations specifically referred to chest X-rays on 14-in. by 17-in. film (Wolf 1948).

The X-ray records in the claim files from BNL do not bear out the recommendations by Sunderman (1947a). While there are some PFGs in the claim files, there are very few lumbar spine examinations, and no X-rays of forearms were found in the reviewed sample of cases [2]. Because no lumbar spine or forearm X-rays were found in claim files records in this early period, it is assumed that BNL occupational medical physicians did not implement the protocol recommended by Sunderman [3].

It is also clear from the claim file records that PFG was not used exclusively in the early years of BNL. Beginning in 1947 and continuing until at least 1964, all workers, including pile (reactor) operators, had a preemployment and an annual physical examination, including a PA chest X-ray on 14- by 17-in. film, but no X-ray on termination (Love 1957, 1959, 1963, 1964; Bond 1963). No PFG records were found in the claim files after 1955 [4].

Dose reconstructors should assign dose according to the number of PFG and 14- by 17-in. PA chest X-rays in the claim file records if present. If these records are not present, the dose reconstructor should assign dose from a preemployment PFG for start dates between 1947 and 1955 and from annual 14- by 17-in. PA chests during this period. The default X-ray examination frequencies are in Table 3-1.

Table 3-1. Default frequency of chest X-rays at BNL.

Period	Type of chest X-ray	Applicability	Frequency
1947–1955	PFG	All workers	Preemployment, if PFG in the records, or no records exist Periodic per the records No termination examination
1947–1955	14- by 17-in. PA	All workers	Preemployment if no PFG in the records Annual No termination examination
1956–1978	14- by 17-in. PA	All workers	Preemployment Annual No termination examination
1979–present	14- by 17-in. PA and LAT	All workers	Preemployment Every 7 years, unless records indicate otherwise No termination examination

The screening protocol policy seems to have stayed fairly constant (Flood 1976), although the actual interval between “annual” X-ray examinations seems to have increased in the mid-1960s, to an average of every 2.5 years (Brodsky 1964, pp. 38–39, 72). It is assumed this examination frequency pattern continued through 1978. Dose reconstructors should assign dose according the number of 14- by 17-in. PA chests in the claim file records if present. If these records are not present, the dose reconstructor should assign dose from a preemployment 14- by 17-in. PA chest and from annual 14- by 17-in. PA chests during this period (1956 to 1978). The default X-ray examination frequencies are in Table 3-1.

In 1978, a Presidential Recommendation to Federal agencies advised the discontinuation of routine or screening radiographic examinations, including chest X-rays, on patients for whom no previous clinical evaluation was made (Carter, J. 1978). This recommendation, at least in part, was incorporated into BNL practice, and the average time between radiographic examinations increased. Starting in 1979, the chest X-ray reexamination cycle for all employees might have decreased to once every 7 years (Morris 2006b). Preemployment radiographic examination is assumed to have continued as a condition of employment. Sometime in the 1980s (probably around 1981), the typical radiographic examination expanded to include both PA and LAT chest films (Morris 2006a). For expedience, this is assumed to have started in 1979 coinciding with the extended reexamination frequency. The current (as of 2009) preemployment X-ray examination practice is to offer PA and LAT chest X-ray to all employees, and it appears that most new employees choose to have that examination. For new employees who test positive for tuberculosis, preemployment PA and LAT chest X-ray are required. The time between screening chest X-rays has been extended over the years and now occurs on a 7- to 10-year cycle (Morris 2006b).

Dose reconstructors should use the number of X-rays in the Energy Employee's X-ray records if provided by DOE for the period from 1979 to the present. If records have not been provided, the default X-ray examination frequency from Table 3-1 can be used by dose reconstructors in assigning dose.

3.4 EQUIPMENT AND TECHNIQUES

Sunderman (1947b) states, "arrangements have been made with the Powers X-ray Service for chest roentgenograms for each employee attached to our Laboratory." Powers X-ray Service was a well-known provider of PFG services on Long Island. Conrad (1948) mentions the continuation of the use of Powers X-ray Service for 1948, and further mentions the exposure as a constant 3,000 mrep. This is consistent with the information in a retrospective interview, during which the medical director stated each examination caused a 4,000-mR exposure (Brodsky 1964, p. 36). Handloser and Love (1951) state that the approximate exposure range from the PFG at BNL was 0.7 to 1.2 R. It is assumed that this value is for a single PFG exposure, while the other exposure values reported by Conrad and Brodsky are for stereo exposures. Because these values are consistent with the dose values in ORAUT-OTIB-0006 (ORAUT 2011b), the dose values from that document should be used for dose reconstruction of PFG for BNL workers.

Sunderman (1947c) mentions that a decision was made to purchase Westinghouse X-ray equipment. It is assumed this equipment was purchased, because Westinghouse X-ray equipment is referred to in the published article by Handloser and Love (1951).

By 1951, the occupational X-ray program was well established and data about the equipment and exposure had been published in the *Journal of Radiology* (Handloser and Love 1951). That paper listed X-ray tube data for a PFG unit, a fluoroscopic unit, and a radiographic unit. However, it is concluded that only the radiographic unit was used for occupational medical screening examinations (Brodsky 1964). The 14- by 17-in. PA chest radiographic technique used for employee screening was 200 mA and 0.10 second at 72 in. with 0.5-mm Al added filtration in the beam. The nominal tube potential was 72 kV, and only the tube potential was adjusted to compensate for patient size (Handloser and Love 1951). The half-value layer (HVL) of the X-ray beam is assumed to be 1.5 mm Al for an X-ray tube with 0.5-mm Al equivalent inherent filtration, 0.5-mm Al added filtration, and peak kilovoltage of 72 kVp (NCRP 1989, Table B.2).

An interview (Morris 2006a) with a registered X-ray technician who worked at BNL from 1960 to 1993 revealed that a well-maintained Picker machine was in use for chest exposures in 1960. In 1973, phantom measurements were made on the Picker machines for various technique factors (Nelson

1973). In 1978, more measurements were made on the Picker machines, which provide information on the HVL and entrance exposures for this period (Carter, N. 1978).

An interview (Morris 2006b) with a registered X-ray technician who worked at BNL from 1993 to March 2006 revealed that the equipment currently in service is a single-phase Picker model BCX with a measured HVL of 3.2 mm Al. The unit is automatically collimated. A grid cassette film holder is routinely used. This unit has been in service since 1991. The unit is tested routinely by a medical physicist in accordance with New York State Department of Health Guidelines. Test records estimate tube filtration as 4.06 mm Al and the HVL as 3.3 mm Al at 80 kVp (Astarita Associates 1998–2004). For a PA chest film, the nominal technique in current use is 110 kVp and 3.2 mAs at 72 in. No measurements were found for the LAT chest film technique but the technique is similar to PA, except it is performed at 6 mAs.

A sample of 26 PA chest films spanning 1949 through 1971 was reviewed to determine past beam collimation practices [5]. The equipment in use appears to have allowed the technician to set the collimation manually. In some films before the early 1970s, collimation is evident. In other films from the same period, no evidence of collimation is visible. That does not necessarily mean collimation was not used. It might indicate that collimation was established just at the edges of the film cassette and was, therefore, not visible on the film, which would be consistent with the practice described by the technician employed in that timeframe. The earliest film in this sample with obvious collimation was made in 1959 (Morris 2006b). Based on this inspection and the statement of the retired technician, it is reasonable to assume that all exposures after 1960 were collimated [6].

3.5 INCIDENT AIR KERMA

BNL medical personnel adopted the unusual (possibly unique) practice of recording the entrance skin exposure (ESE) for some examinations. Such information is found on the radiologist's interpretation form "Report of X-ray Examination" that is included in the employee's medical chart. These ESE data were based on measurements made during the annual calibration of the X-ray machine. This practice of recording ESE was still occurring in 1964 (Brodksy 1964, p. 67). These contemporary data reflect the exposure for a specific machine and a nominal, not individual-specific, technique. It is not known if this exposure included backscatter, and it does not appear to adjust for the worker's chest thickness. Therefore, it is not likely to improve the accuracy of the dose reconstruction significantly.

1947 to 1955, PFG

PFG incident air kerma and organ doses are from ORAUT-OTIB-0006 (ORAUT 2011b).

1947 to 1959, 14- by 17-in. PA Chest

The incident air kerma in air for the 14- by 17-in. radiographs is based on the techniques in Handloser and Love (1951), information in National Council on Radiation Protection and Measurements (NCRP) Report 102 (NCRP 1989), and ORAUT-OTIB-0006 (ORAUT 2011b). The calculation follows:

1. From NCRP Report 102 (NCRP 1989), Table B-3, the average air kerma rate for a single-phase X-ray machine with 2.5-mm Al equivalent total filtration and operated at 75 kVp is 0.45 cGy/100 mAs at 100 cm.

2. Correcting for total filtration of 1.5 mm Al equivalent using the information in ORAUT-OTIB-0006 (ORAUT 2011b, pg. 9), the air kerma rate is 0.67 cGy/100 mAs at 100 cm.

3. Correcting for the actual mAs (20 mAs) used at BNL according to Handloser and Love (1951), the air kerma is 0.134 cGy at 100 cm.

4. Correcting for a source-to-skin distance of 155 cm, the incident air kerma is 0.06 cGy. The measurement of 0.055-R reported by Handloser and Love (1951) supports this value.

1960 to 1978
The incident air kerma for this period is based on the measurements on X-ray equipment reported by Carter, N. (1978). Carter reports a measured ESE of 7 mR (on the "old" Picker machine) and 11 mR (on the "new" Picker machine) for the PA chest at 115 kV, 100 mA, 1/30 second, and a measured HVL of approximately 3.0 mm Al equivalent (Carter, N. 1978). Both the ESE and the unit of Roentgen are now considered obsolete. Therefore, if ESE is converted to incident air kerma (as described in equation 1 of ORAUT-OTIB-0006), then the resulting value (9.6 cGy) is slightly less than the 11 mR reported by Carter (and used in the previous revision of this TBD). Since the values are so close, Carter's value was used as an incident air kerma for dose reconstruction.

1979 to 1986
In 1983, measurements of the X-ray machine in Room 2 were made at 120 kVp and 300 mA at various exposure times (Zukas 1983). It is clear from these measurements, in comparison with those made in 1978, that a grid was probably introduced because the entrance exposures are higher. The entrance exposure measurements range from 25.6 mR to 73.0 mR, without distinction between PA and LAT. It is reasonable to assume that the first three of five measurements, ranging from 15.6 to 37.6 mR are for the PA chest projection, and the latter two, ranging from 50.4 to 73.0 mR are for the LAT chest projection. Both the ESE and the unit of Roentgen are now considered obsolete. Therefore, if ESE is converted to incident air kerma (as described in equation 1 of ORAUT-OTIB-0006), then the resulting values are slightly less than the exposure values reported by Zukas (and used in the previous revision of this TBD). Since the values are so close, Zukas's values were used as incident air kerma values. For dose reconstruction, the incident air kerma for the PA chest for this period is assumed to be 0.030 cGy, and for the LAT chest it is assumed to be 0.060 cGy.

1987 to 1995
In 1987, measurements of the X-ray machine in Room 2 were made at 100 kVp and 200 mA and 300 mA at various exposure times (Zukas 1987). The listed entrance exposures are lower than those determined in 1983, so it is possible there was a change in technique factors and/or screen-and-film combination to produce the lower exposures. The entrance exposures range from 8 mR to 35.6 mR, with no indication of PA or LAT. It is reasonable to assume that the first three of four measurements, ranging from 8 to 16.2 mR are for the PA chest projection, and the last one, at 35.6 mR is for the LAT chest projection. Both the ESE and the unit of Roentgen are now considered obsolete. Therefore, if ESE is converted to incident air kerma (as described in equation 1 of ORAUT-OTIB-0006), then the resulting values are slightly less than the exposure values reported by Zukas (and used in the previous revision of this TBD). Since the values are so close, Zukas's values were used as incident air kerma values. For dose reconstruction, the incident air kerma for the PA chest for this period is assumed to be 0.015 cGy, and for the LAT chest it is assumed to be 0.030 cGy.

1996 to present
Between 1998 and 2004, the PA chest ESE from the Picker unit in Room 1 was measured at least four times by American Board of Radiology/American Board of Medical Physics-certified radiation physics consultants during the course of routine inspection (Astarita Associates 1998–2004). The technique factors were similar to those for nominal PA and LAT chest examination in current use, namely 110 kVp and 3.2 mAs on the large focal spot with a 72-in. source-to-image distance and 23-cm phantom. The measured ESEs were approximately 7 mR. For 1996 to 2010, the incident air kerma for PA chest is rounded up to 0.010 cGy to account for the possibility that the actual technique used was set to a slightly higher tube potential. No measured ESE is available for the LAT projection. Therefore, the incident air kerma for the LAT chest was calculated to be 0.025 cGy based on the rule of thumb that the incident air kerma for LAT chest is conservatively estimated to be 2.5 times that of the PA chest (ORAUT 2011b, p. 22).

3.6 ORGAN DOSE CALCULATIONS

Organ doses for PFG, PA, and LAT chest X-rays were determined using the method described in ORAUT-OTIB-0006 (ORAUT 2011b) with dose conversion factors (DCFs) from ICRP Publication 34 (1982). ICRP (1982) provides tables of average absorbed dose (in milligray) in selected organs for selected X-ray projections at 1-Gy entrance kerma (i.e., air kerma without backscatter) and selected beam qualities (i.e., various HVLs). The Publication 34 tables list the basic DCFs for converting air kerma to organ dose. Substitute DCFs for organs that are listed in the Interactive RadioEpidemiological Program (IREP) but without unique DCFs in ICRP Publication 34 (1982) were selected as described in ORAUT-OTIB-0006 or are footnoted in the organ dose tables. Incident air kerma was obtained from Table 3-2.

Table 3-2. X-ray equipment used at BNL and incident air kerma values.

Period	X-ray machine	kVp (kV)	Assumed HVL (mm Al)	Image size	Incident air kerma (cGy)	
					PA chest	LAT chest
1947–1955	PFG	--	2.5	4 by 5 in. or 70 mm	2.27[a]	--
1947–1959	Westinghouse[b]	72[b]	1.5	14 by 17 in.	0.06	--
1960–1978	Picker[c]	115[c]	3.0[c]	14 by 17 in.	0.011[c]	--
1979–1986	--	120[d]	3.0	14 by 17 in.	0.030	0.060
1987–1995	Picker[f]	110[e]	3.0[f]	14 by 17 in.	0.015	0.030
1996–present	Picker[f]	110[f]	3.5	14 by 17 in.	0.010[f]	0.025

a. From ORAUT-OTIB-0006 (ORAUT 2011b). The reported incident air kerma is for stereo PFG views.
b. Handloser and Love (1951).
c. Carter, N. (1978).
d. Zukas (1983).
e. Zukas (1987).
f. Astarita Associates (1998–2004).

Table 3-3 lists the organ doses for all periods. Skin doses for all skin areas were determined according to the method described in ORAUT-OTIB-0006 (ORAUT 2011b) and listed in Table 3-5.

Table 3-3. Organ doses (rem) for PFG, PA, and LAT 14- by 17-in. chest radiography.

Organ	Projection	1947–1955 PFG[a]	1947–1959 14- by 17-in.	1960–1978	1979–1986	1987–1995	1996 to present
Thyroid	PA	3.94E-01	7.26E-03	5.06E-04	1.38E-03	6.90E-04	6.20E-04
	LAT				7.98E-03	3.99E-03	3.78E-03
Eye/brain	PA	7.25E-02	7.26E-03	5.06E-04	1.38E-03	6.90E-04	6.20E-04
	LAT				7.98E-03	3.99E-03	3.78E-03
Ovaries	PA	2.50E-02	4.14E-03	1.98E-05	5.40E-05	2.70E-05	3.20E-05
	LAT				5.40E-05	2.70E-05	4.00E-05
Urinary bladder/prostate	PA	2.50E-02	4.14E-03	1.98E-05	5.40E-05	2.70E-05	3.20E-05
	LAT				5.40E-05	2.70E-05	4.00E-05
Colon/rectum	PA	2.50E-02	4.14E-03	1.98E-05	5.40E-05	2.70E-05	3.20E-05
	LAT				5.40E-05	2.70E-05	4.00E-05
Testes	PA	5.00E-03	1.86E-04	1.10E-07	3.00E-07	1.50E-07	1.00E-07
	LAT				6.00E-06	3.00E-06	2.50E-06
Lungs (male)	PA	9.50E-01	1.46E-02	5.46E-03	1.49E-02	7.44E-03	5.65E-03
	LAT				1.42E-02	7.08E-03	6.90E-03
Lungs (female)	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Thymus	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Esophagus	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Stomach	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Bone surface	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Liver/gall bladder/spleen/ pancreas	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Remainder organs	PA	1.02E+00	1.50E-02	5.89E-03	1.61E-02	8.03E-03	6.10E-03
	LAT				1.60E-02	8.01E-03	7.75E-03
Breast	PA	1.11E-01	1.08E-03	7.59E-04	2.07E-03	1.04E-03	9.10E-04
	LAT				1.72E-02	8.61E-03	7.90E-03
Uterus	PA	2.50E-02	3.84E-03	2.53E-05	6.90E-05	3.45E-05	3.00E-05
	LAT				5.40E-05	2.70E-05	3.50E-05
Bone marrow (male)	PA	2.09E-01	2.94E-03	1.29E-03	3.51E-03	1.76E-03	1.46E-03
	LAT				2.88E-03	1.44E-03	1.53E-03
Bone marrow (female)	PA	1.95E-01	2.58E-03	1.23E-03	3.36E-03	1.68E-03	1.41E-03
	LAT				2.28E-03	1.14E-03	1.20E-03
Entrance skin[b]	PA	3.06E+00	7.80E-02	1.54E-02	4.20E-02	2.10E-02	1.40E-02
	LAT				8.40E-02	4.20E-02	3.50E-02

a. All organ doses from ORAUT-OTIB-0006 (ORAUT 2011b). The doses are for stereo PFG (i.e., two exposures).
b. Entrance skin dose is the incident air kerma in air multiplied by the backscatter factor of 1.30, 1.35, or 1.4 for HVLs of 1.5, 2.5, or 3.0 mm Al, respectively, from NCRP Report 102 (NCRP 1989).

3.7 DOSE RECONSTRUCTION

The medical records provided by DOE might include adequate information to define the date, type, and count of X-ray examinations that were administered to the energy employee for screening and as a condition of employment. Only radiographs made for screening and as a condition of employment are eligible to be included in dose reconstructions. Such exposures are limited to PA and LAT chest radiographs and were required or recommended as part of a preemployment examination or a routine reexamination.

Table 3-4. Skin dose guidance for various chest projections and periods.[a]

Area of skin	PFG	Poorly collimated PA chest	Properly collimated PA chest	Properly collimated LAT chest
Right front shoulder	EXSD	EXSD	EXSD	ENSD
Right back shoulder	ENSD	ENSD	ENSD	ENSD
Left front shoulder	EXSD	EXSD	EXSD	EXSD
Left back shoulder	ENSD	ENSD	ENSD	EXSD
Right upper arm to elbow	10% ENSD	ENSD	10% ENSD	ENSD
Left upper arm to elbow	10% ENSD	ENSD	10% ENSD	EXSD
Left hand	ENSD	ENSD	10% ENSD	10% ENSD
Right hand	ENSD	ENSD	10% ENSD	10% ENSD
Left elbow, forearm, wrist Right elbow, forearm, wrist	10% ENSD	ENSD	10% ENSD	10% ENSD
Right elbow, forearm, wrist Right elbow, forearm, wrist	10% ENSD	ENSD	10% ENSD	10% ENSD
Right side of head (including temple and ear)	10% ENSD	10% ENSD	10% ENSD	10% ENSD
Left side of head (including temple and ear)	10% ENSD	10% ENSD	10% ENSD	10% ENSD
Front left thigh	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)
Back left thigh	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)
Front right thigh	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)
Back right thigh	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)	RSD (0.52m)
Left knee and below	RSD (0.86m)	RSD (0.86m)	RSD (0.86m)	RSD (0.86m)
Right knee and below	RSD (0.86m)	RSD (0.86m)	RSD (0.86m)	RSD (0.86m)
Left side of face	Eye/Brain	Eye/Brain	Eye/Brain	10% ENSD
Right side of face	Eye/Brain	Eye/Brain	Eye/Brain	10% ENSD
Left side of neck	10% ENSD	ENSD	10% ENSD	10% ENSD
Right side of neck	10% ENSD	ENSD	10% ENSD	10% ENSD
Back of head	10% ENSD	10% ENSD	10% ENSD	10% ENSD
Front of neck	Eye/Brain	Eye/Brain	Thyroid	10% ENSD
Back of neck	10% ENSD	ENSD	10% ENSD	10% ENSD
Front torso: base of neck to end of sternum	EXSD	EXSD	EXSD	Lung
Front torso: end of sternum to lowest rib	EXSD	EXSD	EXSD	Lung
Front torso: lowest rib to iliac crest	EXSD	EXSD	10% EXSD	10% Lung
Front torso: Iliac crest to pubis	10% EXSD	10% EXSD	10% EXSD	10% Lung
Back torso: base of neck to midback	ENSD	ENSD	ENSD	Lung
Back torso: midback to lowest rib	ENSD	ENSD	ENSD	Lung
Back torso: lowest rib to iliac crest	ENSD	ENSD	10% ENSD	10% Lung
Back torso: buttocks (Iliac crest and below)	10% ENSD	10% ENSD	10% ENSD	10% Lung
Right torso: base of neck to end of sternum	ENSD	ENSD	ENSD	ENSD
Right torso: end of sternum to lowest rib	ENSD	ENSD	ENSD	ENSD
Right torso: lowest rib to iliac crest	ENSD	ENSD	10% ENSD	10% ENSD
Right torso: Iliac crest to pubis (right hip)	10% ENSD	10% ENSD	10% ENSD	10% ENSD
Left torso: base of neck to end of sternum	ENSD	ENSD	ENSD	EXSD
Left torso: end of sternum to lowest rib	ENSD	ENSD	ENSD	EXSD
Left torso: lowest rib to iliac crest	ENSD	ENSD	10% ENSD	10% EXSD
Left torso: iliac crest to pubis (left hip)	10% ENSD	10% ENSD	10% ENSD	10% EXSD

a. ENSD = entrance skin dose; EXSD = exit skin dose; RSD = remote skin dose.

Medical X-rays were recorded on a form called “X-ray Examination Request and Report” that was used for all X-rays including screening examinations, injury investigations, and for diagnosing conditions such as shortness of breath. While the forms changed over time, they typically included a section for the date of the request, the reason for the request, and the part of the body to be examined. The examinations that are to be considered in the dose reconstruction are those that indicate “Pre-Emp. Physical,” “Physical Recheck,” or “Termination” as the reason for the examination.

Table 3-5. Skin dose (rem) from various chest projections.[a]

Area of skin	PFG 1947–1955	PA chest 1947–1959	PA chest 1960–1978	PA chest 1979–1986	LAT chest 1979–1986	PA chest 1987–1995	LAT chest 1987–1995	PA chest 1996–present	LAT chest 1996–present
Right front shoulder	6.67E-02	6.E-04	4.E-04	1.1E-03	8.40E-02	5.E-04	4.20E-02	4.E-04	3.50E-02
Right back shoulder	3.06E+00	7.80E-02	1.54E-02	4.20E-02	8.40E-02	2.10E-02	4.20E-02	1.40E-02	3.50E-02
Left front shoulder	6.67E-02	6.E-04	4.E-04	1.1E-03	4.E-04	5.E-04	2.E-04	4.E-04	2.E-04
Left back shoulder	3.06E+00	7.80E-02	1.54E-02	4.20E-02	4.E-04	2.10E-02	2.E-04	1.40E-02	2.E-04
Right upper arm to elbow	3.06E-01	7.80E-02	1.5E-03	4.2E-03	8.40E-02	2.1E-03	4.20E-02	1.4E-03	3.50E-02
Left upper arm to elbow	3.06E-01	7.80E-02	1.5E-03	4.2E-03	4.E-04	2.1E-03	2.E-04	1.4E-03	2.E-04
Left hand	3.06E+00	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Right hand	3.06E+00	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Left elbow, forearm, wrist Right elbow, forearm, wrist	3.06E-01	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Right elbow, forearm, wrist Right elbow, forearm, wrist	3.06E-01	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Right side of head (including temple and ear)	3.06E-01	7.8E-03	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Left side of head (including temple and ear)	3.06E-01	7.8E-03	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Front left thigh	9.E-04	1.E-05	5.E-06	1.E-05	1.E-05	7.E-06	6.E-06	5.E-06	6.E-06
Back left thigh	9.E-04	1.E-05	5.E-06	1.E-05	1.E-05	7.E-06	6.E-06	5.E-06	6.E-06
Front right thigh	9.E-04	1.E-05	5.E-06	1.E-05	1.E-05	7.E-06	6.E-06	5.E-06	6.E-06
Back right thigh	9.E-04	1.E-05	5.E-06	1.E-05	1.E-05	7.E-06	6.E-06	5.E-06	6.E-06
Left knee and below	3.E-04	4.E-06	2.E-06	5.E-06	5.E-06	2.E-06	2.E-06	2.E-06	2.E-06
Right knee and below	3.E-04	4.E-06	2.E-06	5.E-06	5.E-06	2.E-06	2.E-06	2.E-06	2.E-06
Left side of face	7.25E-02	7.3E-03	5.E-04	1.4E-03	8.4E-03	7.E-04	4.2E-03	6.E-04	3.5E-03
Right side of face	7.25E-02	7.3E-03	5.E-04	1.4E-03	8.4E-03	7.E-04	4.2E-03	6.E-04	3.5E-03
Left side of neck	3.06E-01	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Right side of neck	3.06E-01	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Back of head	3.06E-01	7.8E-03	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Front of neck	7.25E-02	7.3E-03	5.E-04	1.4E-03	8.4E-03	7.E-04	4.2E-03	6.E-04	3.5E-03
Back of neck	3.06E-01	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Front torso: base of neck to end of sternum	6.67E-02	6.E-04	4.E-04	1.1E-03	1.60E-02	5.E-04	8.0E-03	4.E-04	7.8E-03
Front torso: end of sternum to lowest rib	6.67E-02	6.E-04	4.E-04	1.1E-03	1.60E-02	5.E-04	8.0E-03	4.E-04	7.8E-03
Front torso: lowest rib to iliac crest	6.67E-02	6.E-04	4.E-05	1.E-04	1.6E-03	5.E-05	8.0E-04	4.E-05	8.E-04
Front torso: iliac crest to pubis	6.7E-03	6.E-05	4.E-05	1.E-04	1.6E-03	5.E-05	8.0E-04	4.E-05	8.E-04
Back torso: base of neck to midback	3.06E+00	7.80E-02	1.54E-02	4.20E-02	1.60E-02	2.10E-02	8.0E-03	1.40E-02	7.8E-03
Back torso: midback to lowest rib	3.06E+00	7.80E-02	1.54E-02	4.20E-02	1.60E-02	2.10E-02	8.0E-03	1.40E-02	7.8E-03
Back torso: lowest rib to iliac crest	3.06E+00	7.80E-02	1.5E-03	4.2E-03	1.6E-03	2.1E-03	8.0E-04	1.4E-03	8.E-04
Back torso: buttocks (Iliac crest and below)	3.06E-01	7.8E-03	1.5E-03	4.2E-03	1.6E-03	2.1E-03	8.0E-04	1.4E-03	8.E-04
Right torso: base of neck to end of sternum	3.06E+00	7.80E-02	1.54E-02	4.20E-02	8.40E-02	2.10E-02	4.20E-02	1.40E-02	3.50E-02
Right torso: end of sternum to lowest rib	3.06E+00	7.80E-02	1.54E-02	4.20E-02	8.40E-02	2.10E-02	4.20E-02	1.40E-02	3.50E-02

Area of skin	PFG 1947–1955	PA chest 1947–1959	PA chest 1960–1978	PA chest 1979–1986	LAT chest 1979–1986	PA chest 1987–1995	LAT chest 1987–1995	PA chest 1996–present	LAT chest 1996–present
Right torso: lowest rib to iliac crest	3.06E+00	7.80E-02	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Right torso: iliac crest to pubis (right hip)	3.06E-01	7.8E-03	1.5E-03	4.2E-03	8.4E-03	2.1E-03	4.2E-03	1.4E-03	3.5E-03
Left torso: base of neck to end of sternum	3.06E+00	7.80E-02	1.54E-02	4.20E-02	4.E-04	2.10E-02	2.E-04	1.40E-02	2.E-04
Left torso: end of sternum to lowest rib	3.06E+00	7.80E-02	1.54E-02	4.20E-02	4.E-04	2.10E-02	2.E-04	1.40E-02	2.E-04
Left torso: lowest rib to iliac crest	3.06E+00	7.80E-02	1.5E-03	4.2E-03	4.E-05	2.1E-03	2.E-05	1.4E-03	2.E-05
Left torso: Iliac crest to pubis (left hip)	3.06E-01	7.8E-03	1.5E-03	4.2E-03	4.E-05	2.1E-03	2.E-05	1.4E-03	2.E-05

a. Values shown to nearest 0.1 mrem

The part of the body examined will be given as “Chest” or “Routine Chest.” The presence of a request for X-ray examination does not mean the employee was actually given the examination. After about 1979, there may be handwritten entries of “Not Indicated” or “Declined” that indicate the X-ray examination was not given and no dose should be assigned. The form (usually the bottom part) indicates the date the examination was given and includes an interpretation (report) by a radiologist. If the form does not include a date given and an interpretation, then it is unlikely the examination was actually given; therefore, dose should not be assigned. In the early years (1947 to 1955), the record will likely indicate if the X-ray was a PFG (identified as “photofluoro”) or a “14 x 17” PA film.

If confusion about the radiographic exposure record exists, the dose reconstructor should consider requesting that the notes on the exterior of the envelope(s) containing the worker’s X-ray films be transcribed and provided. These notes should give insight to the reason the exposures were made, for example preemployment examination, routine surveillance, or diagnosis of injury. The dose reconstructor should assume that any radiograph that was not a PA or LAT chest was performed as a diagnostic test rather than screening and is therefore not to be included in dose reconstruction. If the X-ray envelope notes associate the annotation “LMD” or “industrial” with a particular exposure, that means the radiograph was a diagnostic exposure (i.e., associated with a workplace injury). The dose from these radiographs is not included in dose reconstruction.

The photon energy associated with all occupational medical X-ray dose is in the 30- to 250-keV energy range.

3.8 UNCERTAINTY

ORAUT-OTIB-0006 (ORAUT 2011b) lists the major sources of uncertainty in X-ray output intensity and subsequent effect on dose to the worker. The five sources of uncertainty are:

1. X-ray beam measurement error (±2%),
2. Variation in peak kilovoltage (±9%),
3. Variation in X-ray beam current (±5%),
4. Variation in exposure time (±25%), and
5. Variation in source-to-skin distance as a result of worker size (±10%).

The 10% uncertainty in output intensity as a result of worker size was based on an inverse square correction of output intensity changes from differences of standard chest thickness of ±7.5 cm.

These uncertainties are assumed to be random; therefore, the combined statistical uncertainty was calculated as the square root of the sum of the squares of all the uncertainties, which is ±28.9%. Rounding this up to ±30% provides an adequate and suitably conservative indication of uncertainty. Therefore, for a derived dose equivalent to an individual organ, a total combined standard uncertainty of ±30% can be assumed. Dose reconstructors should, therefore, input the organ dose equivalent as the mean of a normal distribution with a standard deviation of 30%.

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

4.1 INTRODUCTION

Ambient external gamma radiation outside radiologically controlled areas is the result of ubiquitous background radiation, or gamma emissions from stack effluent such as ^{41}Ar, or skyshine due to air scatter from an otherwise well-shielded radiation source.

During the early years (1950 to 1973), BNL used an ion chamber and dynamic capacitor electrometer assembly to measure ambient external radiation. The ion chamber/dynamic capacitor electrometer

assemblies and TLDs were used concurrently during 1973 and 1974. Information on the specific type of TLD used was not published until 1980 when the use of CaF_2:Dy TLDs was noted (Naidu and Olmer 1981). The site changed to LiF:Mg Tl technology in 2000 (BNL 2001a).

Monitoring results from as many as four onsite stations are available for some years between 1967 and 1984. If available, these data are included in the average dose estimates. Beginning in 1985, the perimeter and onsite monitoring program was expanded to approximately 20 onsite stations and steadily increased to more than 50 onsite stations as of 2004 [7].

4.2 PURPOSE

The purpose of this section is to identify the sources and quantify the magnitude of the radiation dose received by BNL employees due to ambient radiation on the BNL site.

4.3 SCOPE

This section evaluates the ambient external radiation based on the monitoring results from four perimeter stations around the BNL site at the northwest, southwest, southeast, and northeast.

Doses due to inhalation and ingestion are evaluated using stack emissions and the results of BNL environmental reports.

4.4 AMBIENT EXTERNAL RADIATION

4.4.1 Onsite Monitoring

Table 4-1 provides average and maximum external gamma radiation results that have been adjusted to reflect a 2,000-hr/yr occupancy. The table also provides annual dose rates at normally unoccupied locations near sources of radiation on the site. The basis for the values in this table are discussed in this section. The following references were used to develop the values in the table: BNL 1963a; Hull 1963, 1964, 1966; Hull and Gilmartin 1967, 1969; Hull 1970a,b,c; BNL 1972; Hull 1973; Gilmartin 1974a through 1974l, Hull and Ash 1974 through 1976; Naidu 1977 through 1980; Gilmartin 1980a through 1980l; BNL 1980; Naidu and Olmer 1981, 1982; Gilmartin 1983a through 1983j, Day and Naidu 1983, 1984, Day, Miltenberger, and Naidu, 1985, 1986, Gilmartin 1986a through 1986e, Miltenberger, Royce, and Naidu 1987 through 1990, 1992; Naidu, Royce, and Miltenberger 1992, 1993; Naidu and Royce 1994, 1995; Naidu et al. 1996; Meinhold and Hull 1998; Schroeder et al. 1998, Lee et al. 1999; BNL 1999a, 2000; Meinhold and Meinhold 2001; BNL 2001a, 2002 to 2008, 2009c, 2011.

Table 4-1. External gamma radiation dose (mrem/yr).

	Onsite locations			Normally unoccupied locations		
Year	Average	Maximum	Monitoring location P-9[a]	HWMF[b] S-6	Building 356[c]	Building 913B[d]
1947[e]	0					
1948	0					
1950[f]	0.18	0.27				
1951	6.4	9.1				
1952	8.0	11				
1953	8.4	12				
1954	8.9	13				
1955	8.9	13				
1956	8.4	12				
1957	7.3	11				
1958	21	30				

Year	Onsite locations			Normally unoccupied locations		
	Average	Maximum	Monitoring location P-9[a]	HWMF[b] S-6	Building 356[c]	Building 913B[d]
1959	27	39	10			
1960	31	51	11			
1961	28	38	8.8			
1962	28	37	31			
1963	28	42	30			
1964	34	41	31			
1965	44	62	29			
1966	53	103	30			
1967	55	110	23			
1968	55	110	21			
1969	55	110	19			
1970	15	30	19			
1971	15	30	3.7			
1972	15	30	3.6			
1973	15	30	3.5	202		
1974	15	30	20	338		
1975	15	30	1.1			
1976	15	30	1.0			
1977	15	30	17	190		
1978	15	30	14	192		
1979	15	30	11	184		
1980	5	10		190		
1981	5	10				
1982	5	10				
1983	5	10		213		
1984	5	10				
1985	5	10				
1986	5	10		209		
1987	5	10				
1988	5	10				
1989	5	10				
1990	5	10				
1991	5	10				
1992	5	10		125		
1993	5	10		105		
1994	5	10		59		
1995	5	10		73		
1996	5	10		31		
1997	5	10		37		
1998	5	10		44		
1999	5	10		33	5.0	
2000	5	10		42	4.1	
2001	5	10		25	5.9	
2002	5	10		23	7.3	
2003	5	10		24	16	2
2004	5	10		24	11	16
2005	5	10		5.3	6.4	16
2006	5	10		4.3	8.7	16
2007	5	10		4.6	12	11
2008	5	10		4.1	11	3

	Onsite locations			Normally unoccupied locations		
Year	Average	Maximum	Monitoring location P-9[a]	HWMF[b] S-6	Building 356[c]	Building 913B[d]
2009	5	10		4.6	8.2	4
2010	5	10		4.6	53	4

a. Monitoring location P-9 was originally known as E-9. After 1979, the P-9 values are included in the average values.
b. The HWMF is represented by dosimeter location S-6. Monitoring data for this site are sporadic.
c. Building 356 contains a Co-60 source. The only monitoring data available starts in 1999.
d. Building 913B dose values represent areas near the AGS tunnel access. The only monitoring data available starts in 2003.
e. Data before 1950 are assumed based on site activities in those years.
f. Data from 1950 to 1961 are from Meinhold and Meinhold (2001).

The site performed environmental monitoring of ambient external radiation levels starting in 1949. Initially there were 16 monitoring stations to measure the ambient levels before and after startup of the BGRR. The locations included onsite locations, perimeter locations, and offsite locations. Four onsite monitoring locations (S-10, S-11, S-12, and S-13) were use to measure external dose from ^{41}Ar releases from the BGRR and later the HFBR. Four perimeter locations were used to monitor for external dose to demonstrate the maximum potential dose for offsite locations. Both of the reactors released activity through the same stack (Meinhold and Meinhold 2001, p. 33). During the first decade of operation, the BGRR was fueled by natural uranium and released about 7,000 Ci/d of ^{41}Ar. In 1958, the fuel was changed to enriched uranium and the emission rate increased to about 20,000 Ci/d. The measured doses above background levels were found to be directly related to the ^{41}Ar emission rate. BGRR operations ended in 1969. The HFBR released much lower levels of ^{41}Ar with a corresponding smaller contribution to onsite and perimeter external dose.

Before 1962, average and maximum environmental external gamma doses are derived from summary data published in retrospect (Meinhold and Meinhold 2001). During the period from 1962 through 1966, doses are derived from measurements at three or four continuous onsite monitoring stations as published in annual environmental monitoring reports, in addition to four perimeter locations.

From 1961 through September of 1979, a large ^{137}Cs source was used to expose a forested area in the Northeast area of the site. The site annual reports during this period list the external dose contribution from this source to the northeast perimeter location and to one of the onsite locations (S-13).

In 1980, the environmental monitoring program was expanded by adding 40 additional TLD monitoring stations in rings at 1, 3, and 10 km from the HFBR stack (Gilmartin 1980a). However, results from the added onsite monitoring stations are not available until 1985. The 1981 annual environmental report (Naidu and Olmer 1982, p. 22) indicated that the Northeast perimeter location (P-9) lies on a bed of coal cinders that contain radium and thorium at larger concentrations than the foundation material used at other perimeter stations, which results in a background that is slightly different from the other perimeters stations. However, the annual external dose values for the perimeter locations in Table 4-1 are not adjusted for this effect.

The HWMF was used to store highly radioactive materials in shielded vaults, trenches, and vertical tubes in the ground. The facility has been in operation since the early 1950s. Monitoring station S-6 is located directly opposite the HWMF area, where low and intermediate level radioactive materials are stored before offsite disposal. The annual external doses for this monitoring station are available for some but not all years. This area includes the igloo storage structures. In 1992 an effort was made to reduce the skyshine from these sources to nearby facilities (Van Der Karr 1993). The HWMF includes several small buildings, only one of which is normally used by workers. This is Building 445, which was constructed in 1956 and housed a triple distillation still for mercury until 1980 (BNL 1997x).

From 1981 to 1991 the building housed the waste management incinerator. Other buildings in the area were used only for storage and were not occupied on a regular basis. As part of the 1992 effort to reduce external dose in the HWMF area, surveys were taken before and after the relocation of vaults and the addition of shielding to radiation sources. The highest annual dose rate (for 2,000-hr/yr occupancy) in the vicinity of Building 445 was found to be 125 mrem/yr before the source relocation and 105 mrem/yr afterward. These values are based on monitoring for only a 2-month period, before and after the relocation. Most of the radioactive materials were removed from the HWMF by the end of 2004, and the doses decreased to less than 10 mrem/yr.

Another potential source of onsite exposure is Building 356 (BNL 1997y, 2009a). The building housed a ^{60}Co irradiation facility under the Physics Department from 1960 to 1966, which was decommissioned in 1965 and 1966. During 1964 to 1966 an addition to the building was designed and built to house the present irradiation facility. The facility was upgraded in 1991 with a new 20,000-curie ^{60}Co source. The areas outside the building were added to the environmental monitoring program, and doses were reported starting in 1999. The annual doses for this location were generally well under 15 mrem/yr, except for 2010 when the dosimeter at the North corner of the building indicated 53 mrem/yr (for 2,000-hr/yr occupancy). Although it is conceivable that individuals using the parking lot at Building 356 could receive a dose from this source, the annual dose would be small because of the short occupancy period for the location (BNL 2009a).

In about 1990, dosimeter locations were added to monitor potential external exposure points near operating facilities with the potential to deliver radiation dose in occupied areas and unoccupied areas. By 1999 these locations included several near the AGS, BLIP, and by 2001 locations near the new RHIC. The recorded doses for all of these locations are included in the summary information in Table 4-1 and indicate that average annual dose (based on 2,000-hr/yr occupancy) is well under 5 mrem/yr from 1984 through 2010 and the maximum dose for occupied areas is under 10 mrem/yr.

Starting in 2003, two dosimeter locations near the AGS tunnel access were added to monitor skyshine from the AGS (BNL 2006d). The maximum of the two dosimeters was always less than 20 mrem/yr. These locations would normally be unoccupied.

In 2010, neutron monitors were placed around the RHIC, AGS, and BLIP facilities. The highest monitored neutron dose was 1 mrem/quarter near these facilities, which indicates an annual dose of less than 1 mrem/yr based on 2,000-hr/yr occupancy. Therefore, the assignment of ambient neutron dose is not necessary.

The available information covers most of the site operating period except for 1967 through 1983, although some information is available for some of these years. For 1967 through 1969 (when the BGRR was still running), the values for 1966 are used to estimate the average and maximum dose. This likely represents an overestimate of dose values because the 1966 values are the highest up to that point. For 1970 through 1983, the average dose is set to 15 mrem/yr based on the available perimeter and onsite station S-13 values. The maximum is set to twice the average value, based on the 1966 ratio of the maximum dose to average dose (the year of the highest ratio for all previous years). For 1984 through 2010, the average values were all under 5 mrem/yr and the maximum values were all under 10 mrem/yr. Therefore, these values have been used as the average and maximum values for all of these years.

4.4.2 Applicability for Dose Reconstruction

For dose reconstruction, the ambient external dose should be applied to all unmonitored workers. For monitored workers, external dose should be based on the dosimetry records without assignment of ambient external dose. The average dose values from Table 4-1 (based on exposure for 2,000 hr/yr) should be assigned to unmonitored workers as a normal distribution with a standard deviation of

100%. This assumes the maximum value (2 times the average value) represents 1 standard deviation. This is reasonable considering the relatively small number of onsite monitoring locations for most years. The values should be corrected for annual exposure time as necessary for each year of exposure.

The dose values in Table 4-1 for unoccupied locations should not normally be applied for unmonitored workers because workers in these locations would normally be assigned radiation dosimeters. If the available information indicates a worker was assigned to one of these areas, and no dosimetry records exist, then the assignment of unmonitored doses described in Section 6.5.2 should be considered.

4.5 INHALATION OF ONSITE AIRBORNE RADIONUCLIDES

BNL has monitored releases to the environment since 1950 when the first stack became operational (Meinhold and Meinhold 2001). Stack release points and principal radionuclides in the airborne effluent are contained in the site annual environmental reports.

4.5.1 Stack Monitoring

With the exception of tritium, the intake for each radionuclide is the product of the highest annual average concentration reported for that radionuclide at any stack or perimeter sampling location during the year and the assumed breathing rate of 1.2 m^3/hr (ICRP 1994) based on a 2,000-hr/yr exposure.

The individual stacks and their period of operation are:

Stack ID	Period of operation	Stack ID	Period of operation
BGRR	1950–1969	Chem	1973–1993
HFBR	1966–2010	BMRR	1996–1997
Hot Lab & HFBR	1986	Incinerator	1981–1996
Hot Lab	1987–2004	Evaporator	1995–present
BLIP/LINAC	1973–2010		

The following isotopes were identified:

H-3	Co-60	Cs-137
Na-22	Zn-65	Ba-140
Na-24	Rh-106	La-140
Ar-41	Ru-106	Ce-141
Mn-54	I-125	Ce-144
Co-58	I-131	Eu-152
Fe-59	Cs-134	Eu-155

4.5.2 Estimates of Potential Inhalation Intake

The intake of each radionuclide that substantially contributes to personnel dose is tabulated by year for 1950 through 2004. After 2004, the only radionuclide of significance identified in the airborne effluent was tritium. In Table 4-2, the maximum annual intakes are provided for 1950 through 2004. Intake values for tritium for 2005 to 2010 are given in Table 4-3. For other radionuclides after 2004, use the 2004 values from Table 4-2. These values are then assigned as a lognormal distribution with a geometric standard deviation (GSD) of 3.

As stated above, with the exception of tritium, the intake for each radionuclide is the product of the highest annual average concentration reported for that radionuclide at any stack or perimeter

Table 4-2. Maximum annual environmental occupational radionuclide inhalation (Bq/yr).[a,b,c]

Year	Radionuclide												
	H-3	I-131	1-125	Na-22	Na-24	Mn-54	Co-58	Fe-59	Co-60	Zn-65	Sr-90	Nb-95 Zr-95	Tc-99[d]
1950	3.01E+04	2.70E-01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1951	3.01E+04	4.73E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1952	3.01E+04	6.09E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1953	3.01E+04	6.42E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1954	3.01E+04	6.76E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1955	3.01E+04	6.76E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1956	3.01E+04	6.42E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1957	3.01E+04	5.41E+01	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1958	3.01E+04	6.15E+02	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1959	3.01E+04	6.15E+02	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1960	3.01E+04	6.15E+02	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1961	3.01E+04	6.15E+02	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1962	3.01E+04	7.03E+02	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	2.11E+02	N/A
1963	3.01E+04	6.15E+02	4.84E+04	5.51E-02	1.19E+03	3.78E+01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	8.30E+02	N/A
1964	3.01E+04	5.71E+02	4.84E+04	5.51E-02	1.19E+03	3.52E+00	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	5.27E+00	N/A
1965	3.01E+04	7.45E+02	4.84E+04	5.51E-02	1.19E+03	6.15E-01	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	5.27E+00	N/A
1966	3.01E+04	8.11E+02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	8.79E-01	N/A
1967	2.11E+05	3.16E+02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	4.84E+00	N/A
1968	1.71E+06	7.47E+05	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	1.05E-01	3.52E-01	4.84E+00	N/A
1969	1.71E+06	1.32E+01	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	2.64E-01	3.52E-01	6.68E+00	N/A
1970	3.96E+06	2.64E-01	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	7.91E-01	3.52E-01	8.79E+00	N/A
1971	1.05E+03	8.79E-02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	3.52E-01	1.76E-01	8.09E+00	N/A
1972	7.91E+02	8.79E-02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	8.79E-02	1.76E-01	1.58E+00	N/A
1973	3.41E+06	8.79E-02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	8.79E-02	1.76E-01	2.64E-01	N/A
1974	4.46E+06	1.76E-02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	1.26E+01	1.76E-01	3.08E+00	N/A
1975	3.42E+06	1.76E-02	4.84E+04	5.51E-02	1.19E+03	8.79E-02	3.22E-01	1.22E+04	1.49E-03	8.79E-02	1.76E-01	1.05E+00	N/A
1976	4.08E+06	1.76E-02	4.84E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.22E+04	1.49E-03	8.79E-02	1.76E-01	1.05E+00	N/A
1977	4.11E+03	8.79E-01	4.84E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.22E+04	1.49E-03	8.79E-02	1.76E-01	8.88E+00	N/A
1978	5.32E+05	8.79E-01	4.84E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.22E+04	1.49E-03	8.79E-02	1.76E-01	8.88E+00	N/A
1979	1.37E+06	8.79E+00	4.84E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.22E+04	1.49E-03	3.42E+04	1.76E-01	8.88E+00	N/A
1980	2.05E+06	8.79E+00	4.84E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.22E+04	1.49E-03	3.42E+04	1.76E-01	8.88E+00	N/A
1981	1.32E+07	4.88E+03	4.84E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.22E+04	1.49E-03	3.42E+04	1.76E-01	8.88E+00	4.88E+03
1982	2.05E+07	4.88E+03	6.35E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	4.88E+03	1.49E-03	3.42E+04	1.76E-01	8.88E+00	4.88E+03
1983	1.90E+07	4.88E+03	8.30E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	4.88E+03	1.49E-03	3.42E+04	1.76E-01	3.52E-03	4.88E+03
1984	6.91E+06	1.47E+04	4.88E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	4.88E+03	2.37E-03	3.42E+04	1.76E-01	3.52E-03	4.88E+03
1985	9.66E+05	1.37E+00	2.98E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.47E+02	2.64E-02	1.40E-02	1.76E-01	3.52E-03	4.88E+03
1986	6.92E+06	1.03E+03	2.54E+04	5.51E-02	1.19E+03	4.88E+02	3.22E-01	1.47E+02	1.15E+00	1.40E-02	1.76E-01	3.52E-03	4.88E+03
1987	1.15E+07	8.30E+03	4.35E+03	5.51E-02	1.19E+03	1.70E-03	3.22E-01	4.88E+01	1.15E+00	1.40E-02	1.76E-01	3.52E-03	2.05E+03
1988	1.71E+06	4.88E+02	9.52E+02	5.51E-02	1.19E+03	5.37E-01	3.22E-01	4.88E+01	4.36E-01	3.24E-01	1.76E-01	3.52E-03	2.44E-01

Year	Radionuclide												
	H-3	I-131	1-125	Na-22	Na-24	Mn-54	Co-58	Fe-59	Co-60	Zn-65	Sr-90	Nb-95 Zr-95	Tc-99[d]
1989	2.70E+06	8.88E-02	1.03E+04	1.03E+00	1.19E+03	5.37E-01	3.22E-01	4.88E+01	6.44E-01	8.16E+00	1.76E-01	3.52E-03	2.44E-01
1990	4.63E+05	6.94E+02	2.30E+03	1.03E+00	1.19E+03	3.81E-02	3.22E-01	4.88E+01	6.55E+00	5.37E-03	1.76E-01	3.52E-03	2.44E+02
1991	6.44E+05	6.94E+02	4.75E+03	5.11E+00	1.19E+03	4.75E-02	1.34E+00	4.88E+01	2.83E-01	2.22E+00	1.76E-01	3.52E-03	2.44E+02
1992	4.44E+05	6.94E+02	2.59E+03	5.11E+00	1.19E+03	2.44E+01	1.34E+00	4.88E+01	2.77E+00	2.22E+00	1.76E-01	3.52E-03	2.44E+02
1993	5.33E+05	3.61E-02	2.44E+04	3.11E-01	1.19E+03	7.89E+00	6.83E+00	1.99E+01	6.65E-01	2.22E+00	1.76E-01	3.52E-03	2.44E+02
1994	5.92E+05	3.61E-02	2.44E+04	3.11E-01	1.19E+03	2.19E-01	6.83E+00	1.99E+01	9.49E-04	1.17E-01	1.76E-01	3.52E-03	2.44E+02
1995	7.15E+05	3.21E+00	1.95E+03	2.52E-03	1.19E+03	2.78E-02	6.74E-01	1.99E+01	3.84E-02	1.17E-01	1.76E-01	3.52E-03	2.44E+02
1996	3.52E+05	3.21E+00	1.47E+03	1.12E-01	1.19E+03	4.30E-02	6.74E-01	1.99E+01	4.88E-03	1.01E+02	1.76E-01	3.52E-03	2.44E+02
1997	1.98E+05	1.72E+02	1.47E+03	4.55E-03	1.19E+03	7.42E-02	4.18E-01	1.99E+01	1.37E+01	1.01E+02	1.76E-01	3.52E-03	N/A
1998	2.73E+05	1.72E+02	1.47E+03	6.59E-03	1.19E+03	1.21E-02	1.99E-01	1.99E+01	7.03E-03	3.92E-01	1.76E-01	3.52E-03	N/A
1999	1.33E+05	1.72E+02	7.86E+00	6.59E-03	1.19E+03	9.33E-02	8.84E-02	1.99E+01	1.18E-01	2.17E+00	1.76E-01	3.52E-03	N/A
2000	3.52E+04	1.72E+02	7.86E+00	1.08E-03	1.19E+03	1.53E-02	1.45E-02	1.99E+01	1.93E-02	3.56E-01	1.76E-01	3.52E-03	N/A
2001	2.64E+04	3.83E-05	7.86E+00	1.08E-03	1.19E+03	1.53E-02	1.45E-02	1.99E+01	1.93E-02	3.56E-01	1.76E-01	3.52E-03	N/A
2002	2.37E+04	3.83E-05	7.86E+00	1.08E-03	1.19E+03	1.53E-02	1.45E-02	1.99E+01	1.93E-02	3.56E-01	1.76E-01	3.52E-03	N/A
2003	5.92E+05	3.83E-05	7.86E+00	1.08E-03	1.19E+03	1.53E-02	1.45E-02	1.99E+01	5.25E-08	3.56E-01	1.76E-01	3.52E-03	N/A
2004	5.92E+05	3.83E-05	7.86E+00	1.08E-03	1.19E+03	1.53E-02	1.45E-02	1.99E+01	5.25E-08	3.56E-01	1.76E-01	3.52E-03	N/A

Year	Radionuclide									
	Tc-99m[e]	Ru-103	Rh-106 Ru-106	Cs-137	Ba-140	La-140	Ce-141	Ce-144	Ra-226	Th-228
1950	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1951	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1952	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1953	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1954	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1955	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1956	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1957	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1958	N/A	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1959	9.82E+03	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1960	9.82E+03	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1961	9.82E+03	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1962	9.82E+03	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1963	9.82E+03	6.51E+01	5.27E+01	2.02E+01	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1964	9.82E+03	8.79E-01	1.23E+01	7.03E+00	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1965	9.82E+03	8.79E-01	2.20E+00	2.37E+00	1.23E+00	1.23E+00	8.79E-01	6.33E+01	2.55E-01	1.76E-01
1966	9.82E+03	6.15E-01	5.27E-01	7.03E-01	1.23E+00	1.23E+00	4.40E-01	6.33E+01	2.55E-01	1.76E-01
1967	9.82E+03	1.05E+00	5.27E-01	7.03E-01	1.23E+00	1.23E+00	4.40E-01	6.33E+01	2.55E-01	1.76E-01
1968	9.82E+03	1.05E+00	5.27E-01	6.15E-01	5.27E-01	5.27E-01	4.40E-01	6.33E+01	2.55E-01	1.76E-01
1969	9.82E+03	1.05E+00	2.11E+00	5.27E-01	1.76E-01	1.76E-01	4.40E-01	1.58E+01	2.55E-01	1.76E-01

Year	Radionuclide									
	Tc-99m[e]	Ru-103	Rh-106 Ru-106	Cs-137	Ba-140	La-140	Ce-141	Ce-144	Ra-226	Th-228
1970	9.82E+03	1.05E+00	2.11E+00	1.23E+00	1.76E-01	1.76E-01	4.40E-01	2.46E+00	2.55E-01	1.76E-01
1971	9.82E+03	1.05E+00	2.11E+00	1.05E+00	8.79E-02	8.79E-02	4.40E-01	6.15E-01	2.55E-01	1.76E-01
1972	9.82E+03	1.05E+00	2.11E+00	4.40E-01	2.64E-01	2.64E-01	4.40E-01	2.37E+00	2.55E-01	1.76E-01
1973	9.82E+03	1.05E+00	1.23E+00	8.79E-02	2.64E-01	2.64E-01	4.40E-01	2.37E+00	2.55E-01	1.76E-01
1974	9.82E+03	1.05E+00	1.23E+00	8.79E-02	2.64E-01	2.64E-01	4.40E-01	4.22E+00	2.55E-01	1.76E-01
1975	9.82E+03	1.05E+00	6.15E-01	8.79E-02	3.52E-01	2.64E-01	4.40E-01	6.68E+00	2.55E-01	1.76E-01
1976	9.82E+03	4.88E+03	6.15E-01	8.79E-02	3.52E-01	3.52E-01	4.40E-01	8.00E+00	2.55E-01	1.76E-01
1977	9.82E+03	4.88E+03	7.91E-01	8.79E-01	3.52E-01	3.52E-01	3.52E-02	1.32E+00	2.55E-01	1.76E-01
1978	9.82E+03	4.88E+03	7.91E-01	8.79E-01	3.52E-01	3.52E-01	3.52E-02	3.52E-01	2.55E-01	1.76E-01
1979	9.82E+03	4.88E+03	7.91E-01	8.79E-01	2.20E-01	2.20E-01	3.52E-02	2.02E+00	2.55E-01	1.76E-01
1980	9.82E+03	4.88E+03	7.91E-01	8.79E-01	2.20E-01	2.20E-01	3.52E-02	7.91E-01	2.55E-01	1.76E-01
1981	9.82E+03	4.88E+03	7.91E-01	3.96E-01	2.20E-01	2.20E-01	3.52E-02	7.91E-01	2.55E-01	1.76E-01
1982	9.82E+03	4.88E+03	9.67E-02	2.64E+00	2.20E-01	2.20E-01	3.52E-02	5.36E-03	2.55E-01	1.76E-01
1983	9.82E+03	4.88E+03	9.67E-02	2.99E-01	2.20E-01	2.20E-01	3.52E-02	5.36E-03	2.55E-01	1.76E-01
1984	9.82E+03	4.88E+03	9.67E-02	1.77E-01	3.70E-03	2.20E-01	3.52E-02	5.36E-03	2.55E-01	1.76E-01
1985	9.82E+03	5.86E+02	9.67E-02	2.81E-01	3.70E-03	2.20E-01	3.52E-02	5.36E-03	2.55E-01	1.76E-01
1986	9.82E+03	5.86E+02	7.03E-02	2.81E-01	3.70E-03	2.20E-01	3.52E-02	5.36E-03	2.55E-01	1.76E-01
1987	4.88E+02	5.86E+02	7.03E-02	2.17E-01	3.70E-03	2.20E-01	3.52E-02	5.36E-03	2.55E-01	5.27E-02
1988	4.88E+02	1.12E+00	7.03E-02	2.00E-01	3.70E-03	1.86E+02	3.52E-02	5.36E-03	8.44E-01	8.79E-02
1989	7.33E+03	1.12E+00	1.28E+02	4.48E-01	3.70E-03	1.86E+02	3.52E-02	5.36E-03	6.15E-01	8.79E-02
1990	7.33E+03	1.52E+01	1.28E+02	6.15E-01	3.70E-03	1.86E+02	3.52E-02	1.86E+00	1.03E+00	3.52E-02
1991	7.33E+03	1.52E+01	1.28E+02	7.77E-01	3.70E-03	1.86E+02	3.52E-02	1.86E+00	1.86E-01	3.52E-02
1992	7.33E+03	1.52E+01	1.28E+02	2.50E-01	7.76E+01	1.86E+02	9.41E-01	1.86E+00	1.86E-01	1.55E-01
1993	7.33E+03	1.52E+01	1.28E+02	4.55E-01	7.76E+01	1.86E+02	9.41E-01	1.86E+00	1.86E-01	1.55E-01
1994	3.03E+02	1.52E+01	1.28E+02	1.38E+00	7.76E+01	1.86E+02	9.41E-01	1.86E+00	5.07E+00	1.55E-01
1995	3.03E+02	1.52E+01	1.28E+02	2.76E+01	7.76E+01	1.86E+02	9.41E-01	1.86E+00	5.07E+00	1.55E-01
1996	3.03E+02	1.52E+01	1.28E+02	5.86E-01	7.76E+01	1.86E+02	9.41E-01	1.86E+00	5.07E+00	1.55E-01
1997	3.03E+02	1.52E+01	1.28E+02	1.47E-01	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
1998	3.03E+02	1.52E+01	1.28E+02	1.31E-02	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
1999	3.03E+02	1.52E+01	1.28E+02	2.59E-01	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
2000	3.03E+02	1.52E+01	1.28E+02	4.25E-02	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
2001	3.03E+02	1.52E+01	1.28E+02	4.25E-02	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
2002	3.03E+02	1.52E+01	1.28E+02	5.44E-04	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
2003	3.03E+02	1.52E+01	1.28E+02	5.44E-04	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01
2004	3.03E+02	1.52E+01	1.28E+02	1.81E-09	8.07E+02	4.27E+03	9.41E-01	7.34E+00	5.07E+00	1.55E-01

a. Highlighted values are bounded by available data. Missing years are filled in with nearest results (highest value if uneven interval).
b. Values not highlighted are from closest available result.
c. Other radionuclides were reported, but were not included because the maximum dose was less than 0.001 rem to the highest metabolic organ.
d. Tc-99 reported only for incinerator, which operated from 1981 to 1996.
e. Tc-99m is reported for the BMRR, which began operating in 1959; Tc-99m is also reported for incinerator.

Table 4-3. Maximum annual environmental occupational radionuclide inhalation, 2005 to 2010 (Bq/yr).

Year	H-3
2005	1.31E+05
2005	2.95E+04
2007	9.74E+03
2008	1.47E+05
2009	2.80E+05
2010	3.98E+04

sampling location during the year, the assumed breathing rate, and an exposure time of 2,000-hr/yr. The ^{3}H intake, assumed to be in the form of water vapor, is determined as described for other radionuclides and then increased by a factor of 1.5 to account for direct absorption through the skin [8].

In most years, stack effluent sampling shows that some radionuclides were released but not specifically monitored at ground-level site or perimeter sampling points. In those instances, the highest annual average stack concentration was reduced by a factor of 0.01 to account for the lessened overall intake due to contribution from multiple widely spaced facilities, atmospheric dispersion, and building wake, and the result was used as though it were a ground-level site or perimeter sample. Most radionuclides released from the stacks were not subsequently detected by ground-level monitoring stations. Therefore, a reduction factor could not be calculated from the data, so a reduction factor of 0.01 was applied to the stack air concentrations to provide a bounding ground-level concentration [9].

No measurements of particle size or solubility are available. Solubility and particle size for environmental intake calculations should be assumed to be the same as those used in Section 5.0 of this document. If no data are available, default solubility classes and particle size values from the ICRP should be used (NIOSH 2002, pp. 15–16).

4.6 INGESTION

Potable water, supplied from onsite wells, provides the most likely pathway for radionuclide ingestion at BNL. Environmental reports include the radionuclide concentration in potable water. Most of the observed radionuclides are naturally occurring and consistent with concentrations found in offsite regional water supplies.

Neither potable groundwater nor soil ingestion has been found to be a pathway of exposure. No ingestion dose is indicated [10].

4.7 UNCERTAINTY

A high degree of uncertainty is associated with environmental monitoring. Uncertainty is the result of many factors listed here in approximate order of importance: atmospheric dispersion models used to describe dilution of radionuclide concentration from the stack to the receptor, location of ground-level monitoring stations to represent the unmonitored worker adequately, efficiency of radioactive iodine sampling media before use of charcoal collectors, average water intake, occupancy factors for workers on the site, average breathing rate, measurement detection limits, and accuracy of measurement results. Additional factors could be listed. The error associated with each factor is unknown, but can be reasonably expected to range from more than a factor of 10 for atmospheric dilution, to about 20% for accuracy of measurement [11].

Although the historical uncertainty cannot be substantially reduced without a great deal of effort, importance of uncertainty in dose reconstruction is reduced by the fact that many of the doses are small. Both the maximum and average annualized concentrations are used to estimate intakes and other assumptions are biased to be favorable to the claimant. An example is the dilution factor used to adjust stack effluent concentrations for use as ground-level exposures. Actual data would suggest that roof-level stacks tend to create dilution factors of 0.001 or more (ORAUT 2004, p. 28), but in this instance a factor of 0.01 is assumed, which results in higher ground-level exposure to an unmonitored worker.

5.0 OCCUPATIONAL INTERNAL DOSE

5.1 INTRODUCTION

BNL has played an important role in the development of the U.S. nuclear program. It has conducted applied research in nuclear and high-energy physics, chemistry and physics of materials, environmental and energy research, nonproliferation, neurosciences and medical imaging, and structural biology. Methods and concepts of measuring occupational internal doses to workers have evolved since the beginning of BNL operations. This section describes BNL internal dosimetry systems and practices.

5.2 PURPOSE

The purpose of this section is to describe internal dosimetry systems and practices at BNL. This section provides supporting technical data to evaluate internal occupational doses that can reasonably be associated with worker radiation exposures covered by EEOICPA.

5.3 SCOPE

This section presents historical and current practices as they relate to the evaluation of internal exposure data for monitored and unmonitored workers. It describes plant facilities and processes, and historical information in relation to dose reconstruction for BNL workers. In addition, it contains supporting documentation to assist in the evaluation of occupational internal doses from these processes in accordance with OCAS-IG-002, *Internal Dose Reconstruction Implementation Guideline* (NIOSH 2002) and addresses the evaluation of dose for unmonitored and monitored workers.

The section presents the technical basis of methods used to prepare dose information for input to the NIOSH IREP computer program and evaluates the uncertainty for BNL exposure and dose records.

NIOSH has determined, and the Secretary of the U.S. Department of Health and Human Services has concurred, that it lacks sufficient personnel or area monitoring data, or sufficient source or source term information, associated with BNL operations to bound potential internal exposures for the period from January 1, 1947, through December 31, 1993 (other than tritium after December 31, 1964) (see Section 1.2).

NIOSH found that while it is not possible to completely reconstruct internal radiation doses for employees who worked at BNL from January 1, 1947, through December 31, 1993, NIOSH intends to use any reliable internal monitoring data that may be available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures) to support a partial dose reconstruction for nonpresumptive cancers and/or cases that have less than 250 work days of employment.

5.4 BIOASSAY RECORDS IN THE INDIVIDUAL FILES

The work performed at BNL included a large variety of operations with varying potential for internal exposure. The greatest potential for worker internal exposures at BNL were with the operating reactors (BGRR, HFBR, and BMRR) where exposure to airborne tritium, activation products, fission products, and fuel components was possible. Workers at the reactors were monitored for internal exposure with bioassay monitoring on a regular basis. Workers in some of the other facilities were also monitored if the there was a significant potential for internal exposure. The potential for worker internal exposure in accelerators and colliders was considered to be small because much of the activation isotopes were short lived or fixed in structural materials (Lessard, Schaefer, and Karol 2001). However, the need for monitoring was not applied uniformly across the site and the bioassay monitoring records are not always available in the employee files. Data in worker files may contain results for bioassay monitoring as described in this section.

5.4.1 Bioassay Record (Form 1720)

The BNL bioassay record, Form 1720, contains one line (record) per sample. The record includes the following fields:

1. NO. (number, i.e., the sequential number of the sample),
2. DATE RECV'D (date sample was received),
3. I or S (code for incident or survey; survey is assumed to be a routine sample),
4. COLLECTION DATE/TIME,
5. COUNT DATE,
6. TOTAL SAMPLE (volume in milliliters or X for a spot sample), and
7. SAMPLE TYPE (U for urine, F for feces, B for blood, H for hair, and WB for whole body).

The results portion of the form is filled in according to the analysis. Columns are provided for mixed fission products (MFPs) in dpm/d beta activity, ^{90}Sr in dpm/d beta activity, and ^{3}H in µCi/L beta activity. Additional columns are provided for NUCLIDE, AMOUNT, and UNITS for alpha activity. Results for ^{90}Sr appeared only with MFP results, indicating a sequential analysis. This form appears to have been used from 1967 to the 1980s. Data from as far back as 1952 appear to have been transferred to these forms in 1967.

5.4.2 Tritium Exposure Evaluation Form

This form was used to calculate the WB dose from tritium bioassay results. Results for the bioassay sample under evaluation that were above some threshold (probably around 1 µCi/L) were transferred to this form. Each row was used to calculate the dose (if any) from one urine result. The column headings are:

1. No. of Intervening Days (since last sample),
2. Gross µCi/L Current Date,
3. Gross µCi/L Previous Date,
4. Fraction Present of Previous Concentration,
5. µCi/L Remaining from Previous Date,
6. Net µCi/L Current Date,
7. Tritium Body Dose for Current Date MREM, and
8. Cumulative Tritium Body Dose MREM.

Doses were not calculated if the Net µCi/L Current Date (Gross µCi/L Current Date minus µCi/L Remaining from Previous Date) was less than some threshold. The threshold might have been

0.3 µCi/L. The form was used from the fourth quarter of 1968 until at least 1986. The doses were totaled for each quarter for comparison to the limits in place at the time. This form could be a secondary source of information if the BNL Bioassay Record (Form 1720) discussed above is missing or illegible. Based on the review of site records and personnel interviews, tritium dose was recorded on the external dosimetry cards (in the employee records) until the committed dose era starting in 1989.

5.4.3 Summary of Whole-Body Counting Results

Summaries of the WB counting results might be included in some employees' records for the 1970s and possibly earlier. The summaries currently list several employees on a single page. Cover letters included with the results provided an interpretation of positive results. For example, the letters discuss cesium results believed to be from fallout, results attributed to clothing contamination, results due to an intake with a body burden calculated, etc. [12].

5.4.4 Individual Whole-Body Counting Results

By the 1980s, WB counting results for each count were included in the records. The earliest results were titled "NAI Results" and consisted of a one- or two-page computer printout. The counts appeared to be done with a single detector and were 15 minutes long. The printout contains:

1. ACQUISITION DATE (date and time of the count),
2. ELAPSED LIVE TIME (length of the count),
3. NUCLIDE (a table of results listing the radionuclide),
4. CONCENTRATION (UNITS) (activity), and
5. ERROR (error).

By 1993, the site had changed to the Canberra ABACOS-plus software package. The ABACOS printout contained:

1. COUNT STARTED (the date and time of the count),
2. ELAPSED LIVE TIME (length of the count),
3. NUCLIDE (a table of results listing the radionuclide),
4. ACTIVITY (nCi) (activity), and
5. %ERROR (2 SD) (error).

Initially the ABACOS printouts displayed "<" and the MDA in the results column for results below the MDA. Due to software limitations, an MDA and decision level could not both be displayed. In preparation for DOE Laboratory Accreditation Program (DOELAP) accreditation, in 1999 the number displayed was changed to the decision level, but the printout incorrectly stated that the number displayed was the MDA (i.e., "MDA activity reported"). A list of MDAs was published separately (BNL 2001b). This was corrected when the software was upgraded and subsequently both a decision level and an MDA were reported [13].

5.4.5 Voluntary Intakes of Radionuclides

The Medical Department conducted research involving internal exposures to radionuclides by human volunteers (DOE 1995). The first experiment started in 1950 and the last started in 1976. Some experiments continued beyond this date as indicated by employee records that show internal exposures as late as 1985. While many of the experiments involved patients who were not employees at the laboratory, some involved volunteer employees. The experiments included a variety of radionuclides administered through injection, inhalation, and ingestion.

The medical records for the energy employee should be reviewed for indications of participation in the voluntary exposure experiments. If information in the telephone interview record, DOL records, or DOE records indicates the employee participated in these experiments but the actual intakes are not included in the records, the dose reconstructor should request the voluntary study medical records from the MRC at BNL. The voluntary intakes are provided in a summary form that indicates the date, radionuclide administered, the mode of administration, and the quantity involved.

5.5 *IN VITRO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PROTOCOLS

5.5.1 *History of In Vitro* Urine Analysis

In January 1949, a chemistry laboratory was organized under the Health and Safety Branch, Medical Division, New York Operations Office (NYOO) of the AEC to provide analytical and spectrographic facilities for dealing with hazards arising from NYOO facilities (e.g., uranium, beryllium, fluorine, thorium, and radium). In April 1949, the NYOO chemistry laboratory began an investigation into the feasibility of using the levels of uranium excretion in urine as a control measure in production plants and processing facilities (AEC 1949). In 1953, this organization became the Health and Safety Laboratory (HASL) and in 1977 the Environmental Measurement Laboratory (EML). In 2002, it became part of the U.S. Department of Homeland Security.

Initially, support was also obtained from Oak Ridge National Laboratory to perform urinalysis for plutonium, polonium, MFPs, barium-lanthanum, yttrium, and strontium. By 1951, BNL was doing its own R&D on urine sample preparation (BNL 1951). In 1952, an analysis procedure for fission products in urine was successful using an adaptation of the Chalk River method (BNL 1952). However, in 1955, some uranium in urine at BNL was still analyzed fluorometrically by EML (AEC 1955). In 1958, a vibrating reed electrometer and chambers suitable for tritium urinalysis were set up in Building T-145 (Bishop 1958). In 1965, enriched uranium in urine was analyzed by EML. The analysis appears to have used both liquid scintillation counting (perhaps as a screening measurement) and electroplating for counting or autoradiography (Steimers 1965). The uranium fusion photofluorimetry urinalyses performed by the NYOO were similar to those performed at other AEC facilities. The default detection threshold for uranium urinalysis is assumed to be 10 µg/L based on a reported sensitivity of 5 to 10 µg/L for uranium fluorometric urinalysis in the early years (ORAUT 2005). During the period from 1956 to 1964, the reporting level for uranium in urine by fluorometric analysis by HASL appears to have been 1.0 µg/L (see for example AEC 1957). Determination of ^{233}U and ^{235}U in urine was by fluorometry and alpha counting (AEC 1958, 1961, 1962). It is uncertain whether the analyses were for direct support, verification, or only for specific analyses needed. In May 1963, 112 urine samples were processed by the in-house bioassay laboratory. Gross beta and ^{90}Sr activity was measured in one individual who had been contaminated in April during reloading of spent fuel elements for shipment (BNL 1963b).

In 1973, bioassay analyses were performed by an Analytical Chemistry section (BNL 1973). In 1977, the Health Physics and Safety Division housed an analytical chemistry and bioassay laboratory (ERDA 1977). In 1999, Eberline was contacted for bioassay analyses, with the exception of ^{3}H, which remained in house until approximately April 2004. Due to some problems with meeting the contractual MDAs, Eberline was replaced by the Severn Trent Services Richland Laboratory (STRL) in December 2001. STRL also did the ^{3}H analyses when they were discontinued at the site. About August of 2007, STRL became TestAmerica Richland Laboratory (TARL). The current BNL bioassay program is accredited by DOELAP.

During the operation of the BGRR, some workers apparently were routinely sampled for MFPs and ^{90}Sr. No program documentation was recovered, but this seems to be the case from a limited review of the individual files. Sampling frequency varied from as short as monthly to as long as annually.

After 1965, when the HFBR started operation, the most prevalent bioassay sampling was for tritium. During the operation of the HFBR, the concentrations of radionuclides other than tritium were not considered to be significant. The rationale was that concentration of tritium in the primary cooling water was several million times its "tolerance limit," where fission products from fuel uranium surface contamination and activations products, such as ^{24}Na, were present at less than 100 times tolerance (BNL undated b, p. 14).

During most of the site's existence, it appears that department or division safety representatives or managers designated personnel for participation in the internal dosimetry program [whole-body counts (WBCs) and/or urinalysis] using guidance provided by Personnel Monitoring. This could have included the frequency of monitoring (BNL 1995; Holeman 1999). Therefore, it might be difficult to determine if monitoring should have occurred by job title. For example, in the 1950s, several references mentioned that personnel described as "janitors" were involved in the cleanup of radioactive contamination. It was not clear if this was a specially designated crew or random site janitors being supervised by health physics personnel.

As some of the site's older facilities have undergone decontamination and decommissioning (D&D) in recent years, D&D workers were sampled for ^{90}Sr, ^{241}Am, ^{238}Pu, $^{239/240}$Pu, ^{234}U, ^{235}U, and ^{238}U.

5.5.2 Fecal Sample Analysis

Although fecal sampling might have been used in response to specific events, no evidence of a comprehensive fecal sampling program was located.

Table 5-1 lists the frequencies for *in vitro* monitoring.

Table 5-1. Internal dose control program (*in vitro*).

Routine monitoring type	Period	Frequency
Spot urine sample for H-3	Before 1993	No comprehensive documentation was found. It appeared to vary from weekly to monthly over time. The majority of samples after 1965 were from workers at the HFBR.
Urine samples for H-3[a]	1993–2000	Weekly for reactor operators, supervisors, maintenance technicians, and S&EP building safety services technicians.
24-hr urine samples for fission, activation, uranium, and transuranic elements		No comprehensive documentation was found. In 1984, an annual program of urine sampling was in place for reactor workers. The analysis included Cd-109, Sr-85, Ga-68, Na-22, and Co-58[b].

a. BNL (1993).
b. Miltenberger and Steimers (1984).

5.6 *IN VIVO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PRACTICES

5.6.1 Whole-Body Counting

WB counting was established in 1960. It was first used as a research tool, but it became evident that it was also of use when someone was potentially involved in an incident. After confirmed intakes, WB counting was repeated daily and the results were correlated with daily urine excretion results. Pile operators were the first group to be routinely counted. Some of the pile operators had been working before WB counting started. Later, all new pile operators received a WB count before they started working and, starting in about 1962, annually thereafter. Other groups were apparently counted in the WBC as time permitted. Background counts were taken on unexposed individuals as a check on fallout radionuclides (e.g., cesium) in the general population (Brodsky 1964, p. 40). The counting facility, which is still in existence in the Medical Department, includes a counting bed with an array of sodium iodide thallium-activated [NaI(Tl)] scintillation crystals.

By 1963, a portable "shadow shield" WBC was developed to measure the fission product body burdens of the Marshallese people who were exposed to the fallout from nuclear test Bravo on March 1, 1954, and to the subsequent uptake of radioactive material in crops. The detector was a 10-cm-thick by 28-cm-diameter NaI(Tl) crystal. The detector was stationary and was fixed in a position over the thorax of the subject. For more than 25 years, BNL had a contract for the long-term medical surveillance of the Marshallese. In 1977, the responsibility for providing body burden measurements was transferred from the Medical Department to the S&EP Division. In 1980, the MDL of the system was 1 nCi of ^{137}Cs or ^{60}Co for a 15-minute count (Miltenberger, Greenhouse, and Lessard 1980). Between these measurement trips, the counter was used at BNL as part of the health and safety program and eventually performed the routine counts (Miltenberger 1981a). For positive results, additional counts could be done at the medical WB counting facility to localize the contamination (Miltenberger 1981a). The MDL for the shadow shield arrangement was calculated in microcuries by:

$$MDL = \frac{4.65\sqrt{Ct_b}}{900 \times 37 \times K} \quad (5\text{-}1)$$

where:

Ct_b = blank counts (background for region of interest)
900 = count time (seconds)
37 = unit conversion factor
K = product of gamma abundance and counting efficiency for the radionuclide

By 1981, ^{137}Cs body burdens of 2 nCi or less were considered to be the result of fallout from atmospheric weapons testing environmental sources (Miltenberger 1981b).

In 1989, the MDA reported for thyroid counts for ^{131}I using a NaI counting system was "on the order of 10 nanocuries" (Miltenberger 1989).

The shadow shield was replaced by a standup counter, but the date of this transition was not found. The NaI detectors were replaced by germanium detectors and eventually by broad energy germanium detectors.

In 2003, Canberra ABACOS 2000 software was used with the WBC. The MDA for each radionuclide was calculated by:

$$MDA = \frac{3.29 \times S_b + 3}{KT} \quad (5\text{-}2)$$

where:

S_b = sample standard deviation of background counts in the region of interest
K = efficiency at the centroid energy channel as determined arithmetically from the calibration efficiency equation times the yield abundance of the energy emission
T = count time

The ABACOS 2000 software produced a printout with the MDA and decision level for each result (Michel and Sun 2003; Michel 2003, 2004, 2005; Sun, Reciniello, and Sengupta 2004).

5.6.2 Chest Counting

Chest counts were not done routinely until about 1993 for reactor workers (based on review of bioassay records). Thyroid counts were not done routinely. These counts were done as a result of incidents involving iodine (thyroid) and certain gamma emitters (^{127}Xe, lung) (Miltenberger and Lessard 1987). Only one person who did iodinations was on a thyroid counting program in 2006.

Table 5-2 lists the frequencies for *in vivo* monitoring.

Table 5-2. Internal dose control program (*in vivo*).

Routine monitoring type	Period	Frequency
Preemployment and annual WBC[a]	1962–unknown	Annual for pile operators. Other workers designated by the facility health physics staff of the various facilities.
WBC[b]	1993–2000	Start of service in Building 750, and termination of service for Reactor Division and S&EP personnel with offices in the HFBR building. Annual for reactor operators, supervisors, maintenance technicians, S&EP building safety services technicians and research coordination group technicians, BLIP personnel, facility support, others on prejob basis.

a. Brodsky (1964).
b. BNL (1993).

5.7 UNCERTAINTY

At BNL, the uncertainty for a single bioassay measurement was not reported consistently. Uncertainties associated with the analysis of MFPs, ^{90}Sr, or ^{3}H performed on the site from 1952 to 1986 were not included on the BNL Bioassay Record. From the 1980s to about 1998, uncertainties are reported on WBC sheets and are believed to be consistently reported as 2-sigma errors. Reviewed statements of work for bioassay services do not contain any specification for reporting uncertainty. Eberline reported a 1-sigma total error with its bioassay results. At present, the TARL contract laboratory reports results with both a 1-sigma counting error and a 1-sigma total error [14].

5.8 DETECTION LIMITS

Table 5-3 lists the MDAs for *in vitro* bioassay analyses, and Table 5-4 lists the MDAs for *in vivo* bioassay analyses. The MDAs are listed for periods corresponding to the bioassay methods discussed in Sections 5.5 and 5.6. The reporting levels are listed in the units quoted in the references, which are generally the units of the results. The values in Tables 5-3 and 5-4 are primarily from worker bioassay record files. When bioassay results are found for a worker, the detection limits are usually included in the report. For the period 1992 – 1999, the records may indicate the counting configuration was for a lung count. However, the actual configuration was consistent with a whole body count, and the results should be treated as a whole body count.

Table 5-3. Detection limits for *in vitro* bioassay.

Radionuclide	Method/ description	Period	MDA[a]	Reporting level[b]
Am-241 (Eberline)	Urinalysis	1999–2000	6.0×10^{-2} pCi/L	No threshold
Am-241	Urinalysis	2001–2006	8.4×10^{-8} µCi/24 hr	No threshold
Am-241 (STRL–TARL)	Fecal[c]	2001	0.16 pCi/24 hr	
Br-77	Urinalysis	1975		5 µCi/L[d]
Cd-109	Urinalysis	1978	140 pCi/L	
Co-58	Urinalysis	1994	2.0×10^{-8} µCi/mL[e]	
Co-60	Urinalysis	1978	14 pCi/L	
Co-60 (STRL–TARL)[f]	Urinalysis	2002	10 pCi/L	
Cs-132	Urinalysis	1978	26 pCi/L	

Radionuclide	Method/ description	Period	MDA[a]	Reporting level[b]
Cs-134	Urinalysis	1978	15 pCi/L	
Cs-137	Urinalysis	1964–1967		0.001 µCi/L
Cs-137	Urinalysis	1978	14 pCi/L	
Cs-137	Urinalysis	1994	25 pCi/L	
Cs-137 (STRL–TARL)	Urinalysis	2002	10 pCi/L	
H-3 (in house lab)[g]	Urinalysis	1959–1993		2.0×10^{-2} µCi/L
H-3 (in house lab)[h]	Urinalysis	1994–2000	5.0×10^{-3} µCi/L	
H-3 (required by contract)	Urinalysis	2001–11/2004	1.0×10^{-2} µCi/L	
H-3 (STRL–TARL)	Urinalysis	12/2004–9/2008	5.0×10^{-3} µCi/L	
I-126	Urinalysis	1978	42 pCi/L	
I-131	Urinalysis	1978	13 pCi/L	
MFPs[g]	Urinalysis	1952–1956		5.0 dpm/24 hr
MFPs	Urinalysis	1957–1976	2 dpm/24 hr[i]	
Na-22	Urinalysis	1978	19 pCi/L	
P-32	Urinalysis	1957		0.02 µCi/24 hr
P-32	Urinalysis	1985		2000 pCi/L
Pd-109[j]	Urinalysis	1982	1.6 pCi/mL	
Po-210	Urinalysis	1961–1962		12 dpm/d
Pu-238 (Eberline)	Urinalysis	1999–2000	6.0×10^{-2} pCi/L	No threshold
Pu-238 (STRL–TARL)[f]	Urinalysis	2001–2008	6.0×10^{-2} pCi/L	No threshold
Pu-238 (STRL–TARL)	Fecal[c]	2001	0.10 pCi/24 hr	No threshold
Pu-239 (ORNL)	Urinalysis	1950–1952		0.1 dpm/24 hr
Pu-239	Urinalysis	1963–1971		0.05 dpm/24 hr
Pu-239/240 (Eberline)	Urinalysis	1999–2000	6.0×10^{-2} pCi/L	No threshold
Pu-239 (STRL–TARL)	Urinalysis	2001–2006	6.0×10^{-2} pCi/L	No threshold
Pu-239 (STRL–TARL)	Fecal[c]	2001	0.081 pCi/24 hr	No threshold
Pu-242 (STRL–TARL)	Urinalysis	2004–2006	1.0 pCi/L	No threshold
Pu-242 (STRL–TARL)	Urinalysis	2007–2008	6.0×10^{-2} pCi/L	No threshold
Ra-226	Urinalysis	1957	1.2 dpm/d	
Ra-226 (STRL–TARL)	Urinalysis	2003–2007	1.0×10^{-1} pCi/L	No threshold
S-35	Urinalysis	1985		1800 pCi/L
Sr-90[d]	Urinalysis	1950–1998		5.0 dpm/24 hr
Sr-90 (Eberline)	Urinalysis	1999–2000	1.0 pCi/L	No threshold
Sr-90 (STRL–TARL)	Urinalysis	2001–2008	1.0 pCi/L	No threshold
Sr-90 (STRL–TARL)	Fecal[c]	2001	6.2×10^{-2} pCi/24 hr	No threshold
Th-228 (STRL–TARL)	Urinalysis	2001–2009	0.1 pCi/L	No threshold
Th-230 (STRL–TARL)	Urinalysis	2001–2009	0.1 pCi/L	No threshold
Th-232 (STRL–TARL)	Urinalysis	2001–2009	0.1 pCi/L	No threshold
Uranium (fluorometric)	Urinalysis	Up to 1955		10 µg/L
Uranium (fluorometric)	Urinalysis	1956–1958		1 µg/L
Uranium (fluorometric)	Urinalysis	1959–1969		5 µg/L
U-238 (fluorometric–ORNL)	Urinalysis	1988		1 µg/L
Uranium (radiometric)	Urinalysis	1961–1969		0.2 dpm/d
U-233 (STRL–TARL)	Urinalysis	2002	3.0×10^{-2} pCi/L	No threshold
U-234 (Eberline)	Urinalysis	1999–2000	1.0×10^{-1} pCi/L	No threshold
U-234 (STRL–TARL)	Urinalysis	2001–2009	1.0×10^{-1} pCi/L	No threshold
U-234 (STRL–TARL)	Fecal[c]	2001	7.1×10^{-2} pCi/24 hr	No threshold
U-235 (Eberline)	Urinalysis	1999–2000	1.0×10^{-1} pCi/L	No threshold
U-235 (STRL–TARL)	Urinalysis	2001–2009	1.0×10^{-1} pCi/L	No threshold
U-235 (STRL–TARL)	Fecal[c]	2001	7.3×10^{-2} pCi/24 hr	No threshold
U-238 (Eberline)	Urinalysis	1999–2000	1.0×10^{-1} pCi/L	No threshold
U-238 (STRL–TARL)	Urinalysis	2001–2009	1.0×10^{-1} pCi/L	No threshold
U-238 (STRL–TARL)	Fecal[c]	2001	7.1×10^{-2} pCi/24 hr	No threshold

a. Values from review of available bioassay records. Values could be applicable earlier or later, but documentation was not found.

b. Reporting levels shown were not documented as the MDA but can be interpreted as the MDA in lieu of other information.

c. Values based on one fecal sample analysis in 2001.
d. Value below which "no exposure was assigned" from O'Connell (1975).
e. Value from Murray (1994).
f. Values for STRL – TARL are contractual required detection limits.
g. Values from employee bioassay data records.
h. Value from Scarpitta (2001).
i. Value from BNL (1958).
j. Value from Miltenberger (1982).

Table 5-4. Detection limits for *in vivo* bioassay.

Radionuclide	Method/ description	Period	MDA (nCi)[a]	Reporting level (nCi)[b]
Am-241	WBC	2003–2009	2.33–9.47	
Ba-133	WBC	2001	23.1	
Ba-140	WBC	2001	22.1	
Be-7	WBC[c]	1992–1999	25	
	WBC	1985–1987	45	
	WBC	1973–1974		5
	WBC	2002–2009	25	
	WBC (dual detector)	1992–2001	50	
	WBC (single detector)	1993	82	
Cd-109	WBC	2001	20.1	
Ce-139	WBC	1999–2009	1.76–9.0	
Ce-144	WBC	2003–2008	13.3–27.1	
Co-57	WBC[c]	1993–1999	3.14–11.1	
	WBC	1986–2009	1.24–13.0	
Co-58	WBC[c]	1992–1999	0.95–7.81	
	WBC	1973		5
	WBC	1973–2009	2.39–5.0	
Co-60	WBC[c]	1992–1999	1.19–8.13	
	WBC	1987		2.1
	WBC	1986–2009	1.29–9.20	
Cr-51	WBC[c]	1992–1998	14.8–79.1	
	WBC	2001	203	
Cs-134	WBC[c]	1992–1999	0.851–5.41	
	WBC	1979		1
	WBC	1987		2.4
	WBC	1986–2009	2.09–10.3	
Cs-137	WBC[c]	1992–1999	1.02–7.06	
	WBC	1987		3.4
	WBC	1973–1980		1
	WBC	1986–1988	3.0–6.0	
	WBC (dual detector)	1993–2009	2.64–6.08	
	WBC (single detector)	1993	8.5–9.2	
Fe-59	WBC[c]	1993–1999	2.01–7.41	
	WBC	1986–1988	6.0–8.0	
	WBC	1999–2009	4.0–7.8	
Hg-203	WBC[c]	1998	1.85–4.10	
I-123	WBC[c]	1985–1998	4.64–8.0	
I-125	Thyroid	1984		0.3
	Thyroid	1979–1980	4.1–5.4	
	Thyroid	1984–1989	0.30–0.50	
I-131	WBC[c]	1992–1998	2.19–6.64	
	Thyroid	1989	4	
	WBC	1961		10
	WBC	1986	9	
	WBC	1987	4	

Radionuclide	Method/ description	Period	MDA (nCi)[a]	Reporting level (nCi)[b]
	WBC	1973–1974		5
	WBC	1988–1989	2.3–8.3	
	WBC	1999–2008	1.92–6.0	
	WBC (dual detector)	1993	3.8	
	WBC (dual detector)	1994	3.3	
	WBC (single detector)	1993	8.8	
I-133	WBC[c]	1992–1998	0.85–11.0	
	WBC (dual detector)	1994	3.4	
Mn-54	WBC[c]	1992–1999	0.966–8.21	
	WBC	1987		2.5
	WBC	1979–1980		1
	WBC	1986–1988	2.0–4.0	
	WBC	1999–2009	2.22–3.74	
	WBC (dual detector)	1993–1994	4.7	
	WBC (single detector)	1993–1994	7.5	
Na-22	WBC[c]	1993–1999	1.03–8.87	
	WBC	1999–2009	2.58–4.03	
Na-24	WBC[c]	1995–1996	1.29–5.18	
Os-185	WBC[c]	1995–1998	1.03–4.87	
Sc-46	WBC[c]	1997–1998	1.0–3.99	
Sn-117m	WBC[c]	1995–1998	4.53–6.06	
Sr-85	WBC[c]	1999–2008	2.21–7.0	
Tl-208	WBC[c]	1992–1993	3.60–9.21	
U-235	Lung	1973–1979	0.2	
	WBC	1973		1
Xe-127	Lung	1985–1987	4.7	
Y-88	WBC	1973		5
Zn-65	WBC[c]	1992–1999	2.29–15.6	
	WBC	1973–1974		2
	WBC	1986–1988	7.0–8.0	
	WBC	1999–2008	5.0–9.18	
	WBC (dual detector)	1993–1994	8.50	
	WBC (single detector)	1993–1994	19.1	
Zr-95/Nb-95	WBC[c]	1992–1994	4.53–16.4	
	WBC[c]	1995–1998	1.65–6.93	

a. Range of values found in bioassay records and BNL correspondence. Individual records will usually include this information for the worker.
b. Reporting level was the MDA.
c. The bioassay reports may indicate a "lung geometry" was used, however, this refers to the calibration phantom and results should be treated as a whole body count.

5.9 EXCRETA SAMPLE KIT CODES

No codes have been found for excreta sampling kits. Routine samples for H-3 were grab samples (BNL 1993) to be collected weekly or within 4 hours of completion of a task that required bioassay. Twenty-four-hour samples could also be collected when other radionuclides were suspected (BNL 1993). In 1984, 24-hour kits consisted of three 500-ml bottles (Miltenberger and Steimers (1984). In 1999, spot samples were collected in 60-cm^3 plastic bottles (Holeman 1999).

5.10 SOLUBILITY TYPE, FRACTION ACTIVITY, AND PARTICLE SIZE BY FACILITY

In the absence of measurements or studies, NIOSH guidance requires the use of default solubility classes and particle size values from the ICRP (NIOSH 2002, pp. 15, 16). Facility-specific solubility and particle size data for BNL have not been found. Activity fractions for occupational exposure were

generally not available. For some facilities, stack emissions or other measurements could be useful for dose reconstructions. Table 5-5 lists this information. In all cases, specific information about particle size was not found; the default of 5-µm activity median aerodynamic diameter is assumed to apply. In addition, specific information about solubility type was not found; the solubility that is most favorable to the claimant should be applied.

Table 5-5. Solubility type and fraction activity by facility.

Facility[a]	Compound[b]	Radionuclide	Activity fraction
Sewage Treatment Plant (1962)	Unknown. Measured at the outfall from the Imhoff tank (Stubbings 1962).	Sr-90	0.20
		Cs-137	0.80
Sewage Treatment Plant (1973)	Unknown. Fractions based on activities measured at the input to the sand filter beds (ERDA 1977, p. 126).	H-3	9.87E-01
		Be-7	7.33E-03
		Na-22	6.01E-05
		Na-24	3.59E-03
		Cr-51	2.49E-04
		Co-58	9.02E-05
		Co-60	2.58E-05
		Zn-65	3.44E-05
		Sr-90	5.58E-05
		I-131	1.37E-03
		Cs-134	1.72E-05
		Cs-137	1.07E-04
		Ce-144	1.98E-04
BGRR Buildings 701 to 703, 704 (fanhouse), 708 (instrument house), 709 (canal house), 709A (canal water treatment facility) (1950–1957)	Natural uranium metal fuel, 1- × 4–in. cylinders in 11-ft finned aluminum cartridges. Activity fractions based on estimates of core activity at the time of 28 fuel ruptures and one irradiated uranium sample failure (Meinhold and Meinhold 2001).	Ba-140	1.70E-01
		Ce-144	1.08E-01
		Cs-137	7.19E-03
		La-140	1.95E-01
		Nb-95	1.62E-01
		Ru-103	7.28E-02
		Ru-106	6.09E-03
		Sr-89	1.17E-01
		Sr-90	7.78E-03
		Zr-95	1.53E-01
		U-235	8.35E-08
		U-238	1.81E-06
		U-234	1.81E-06
		Pu-239	2.86E-04
BGRR Buildings 701-703 (1958–1969)	Enriched uranium fuel, in the form of curved plates of uranium-aluminum alloy, clad on all surfaces by 0.5 mm of Al. Routine releases of fission products occurred due to U contamination on the fuel's surface and trapped in the cladding. Based on I-131 and gross beta measurements in the 1967 to 1969 environmental monitoring reports (Meinhold and Hull 1998).	Co-60	8.05E-02
		Zr-95	3.79E-01
		Ru-103	2.30E-01
		Ce-141	1.26E-01
		Ce-144	1.15E-01
		Cs-137	6.90E-02
		I-131	2.86E-03

Facility[a]	Compound[b]	Radionuclide	Activity fraction
BGRR Buildings 701-703 (1970–2006)	Residual contamination from fission, activation, and transuranic radionuclides produced in the core and spread primarily by contact with the cooling water (Musolino 2000). Fuel storage canal was decommissioned in 1973.	Pu-238	6.66E-06
		Pu-239/40	5.86E-04
		Pu-241	2.59E-04
		Am-241	5.05E-05
		Fe-55	9.79E-01
		Co-60	4.89E-04
		Sr-90	9.09E-03
		Cs-137	1.08E-02
		U-235	1.37E-07
		U-238	3.57E-06
BMRR Buildings 490-491 (1959–2000)	Enriched uranium fuel, in the form of curved plates of uranium-aluminum alloy, containing 12% by weight of fully enriched uranium. Estimates are for 1997 routine releases of isotopes likely to be of dosimetric importance (Meinhold and Meinhold 2001, Table 5, p. 57) 99.9% of activity release to sewer from laboratories was H-3 (ERDA 1977, p. 123).	Ce-141	2.99E-03
		Ce-144	2.33E-02
		Co-60	4.35E-02
		Fe-59	6.32E-02
		I-131	5.47E-01
		Zn-65	3.20E-01
Chemistry Hot Chemistry Polonium Facility T-137 (1954–1961)	Polonium source used in chemistry apparatus; also separated as a byproduct of the production of At-211 for medical purposes.	Po-210	
Cosmotron Building (1953–1966)	Activation of short-lived isotopes in the ventilating air in the tunnel and experimental areas. See “Accelerators & RHIC” below.	C-11	
		N-13	
		O-15	
AGS Buildings 905-912 (1960–2006)	Activation of short-lived isotopes in the ventilating air in the tunnel and experimental areas. Processing of irradiated targets (Meinhold and Meinhold 2001, p. 58; ERDA 1977, p. 310). See “Accelerators & RHIC” below.	C-11	
		N-13	
		O-15	
AGS Bubble Chamber Building 919 (1977)	Handling of tritium sources up to 100 mCi.	H-3	1
Accelerators and RHIC (1953–2006)	Posted contamination areas primarily in the vicinity of fixed targets (Lessard, Schaefer, and Karol 2001). Based on recent information, but is assumed to apply to earlier routine operations. Before 1987, beam intensities were less, but some uncontained targets (e.g., tungsten) were used.	Be-7	5.50E-01
		Na-22	1.40E-01
		Mn-54	1.40E-01
		Co-57	1.40E-01
		Co-60	3.00E-02
Hot Laboratory (central lab, fanhouse, liquid waste tank farm, liquid waste concentration plant) Building 801, 802 (1951–2006)	Acid dissolution of irradiated uranium samples to recover radioiodines (March 1952 to June 1960). In 1957, there was an accidental release of unirradiated natural uranium hexafluoride due to an explosion (Meinhold and Meinhold 2001, p. 59). Radioactive solutions up to mCi amounts, 200 Ci/yr of H-3 fabricated into accelerator targets (1977). 87% of the liquid waste in 1973 was H-3 (ERDA 1977, p. 125). Fractions based on environmental releases of isotopes of dosimetric importance, 1983–1991.	Co-58	9.33E-02
		Co-60	1.30E-03
		Cs-137	1.92E-02
		Eu-155	3.37E-03
		I-131	8.67E-02
		Mn-54	1.88E-03
		Zn-65	7.91E-01
		Zr-95	2.95E-03
		U	Unknown
		H-3	Unknown
Hot Laboratory (1955–1966)	Polonium contamination; Po-210 sources manufactured in alpha facility.	Po-210	
Nuclear Engineering T-197, 480 (1954–1962)	Unspecified polonium use requiring urinalysis (1954); and air sampling (1962).	Po-210	

Facility[a]	Compound[b]	Radionuclide	Activity fraction
Physics T-248, T-109, others before (1947–1951)	Polonium used in physics experiments.	Po-210	
Physics Building 510 (1962–present)	Processing irradiated targets, up to 1 mCi; 5,000 Ci of Co-60 in the *GIF* (1977).		
Radiation and Chemical Technology Buildings 526-527	Unknown uranium of various enrichments. Originally housed a criticality facility for reactor physics (ERDA 1977, p. 48). Monomer processing (ERDA 1977). Concrete dried and impregnated with monomers.		
Engineering Test Building 528 (1963–present)	Contained Co-60 sources up to 10,000 curies which were removed by the mid-1970s. Low activity sources for instrument testing	Co-60	
High Intensity Radiation Development Laboratory Building 830 (1951–present)	Contained million-Ci Co-60 and Cs-137 sources. The HIRDL was used for the development and testing of these sources. Operations were curtailed in 1972 due to lack of funding. In 1977, only a water-tank Co-60 GIF was still in operation (1970 – 1990s). Fabrication of solid sources up to Ci amounts;	Cs-137 Co-60	
Hot Machine Shop Building 530 (1947–1975) Replaced by Building 462	Machining radioactive solids (1977), including uranium and radioactive graphite. Repair work on “hot” fuel-handling tools.		
Health Physics & Safety, Instrumentation Building 535 (1964–present)	Radioactive solutions up to 100 µCi (1977).		
Chemistry Building 555 (1966–present)	Radioactive solutions and irradiated targets up to 1 mCi (1977).		
Reclamation and Hot Laundry Building 650 (1959–1996)	Radioactive decontamination up to mCi amounts (1977). 82% of activity released to sewer in 1973 was H-3, remainder was gross beta (ERDA 1977, p. 125).		
HFBR Building 750 (1965–1997)	“Fully” enriched uranium and aluminum alloy, heavy-water-cooled and -moderated reactor that operated up to 40 MW-thermal (60 MW planned in 1977). Fractions based on activity in heavy water when systems were opened for refueling or maintenance (1977).	H-3 Na-24 Cr-51 Co-59 Co-60 Zn-69 I-131 Cs-137	9.998E-01 2.22E-04 2.22E-06 8.89E-10 6.67E-10 8.89E-09 6.67E-10 3.33E-10
Tandem Van de Graaff Accelerator Building 901, 901A (1970–present)	Tritium gas and vapor. Fractions based on activity released from facility in 1979 (Naidu 1979).	H-3 (gas) H-3 (vapor)	8.95E-01 1.05E-01
Superconducting Test, Machine Shop, RARAF[c] Building 902 (1968–present)	Irradiated target processing, up to 100 µCi (1977).		

Facility[a]	Compound[b]	Radionuclide	Activity fraction
BLIP Facility Buildings 930 and 931B (1972–present)	Processing irradiated samples containing short-lived radionuclides (1977) for medical use; targets were sealed during normal operations but radioactive gases (N-13, O-14, O-15, N-16) and other isotopes formed in cooling water and equipment (ERDA 1977, p. 113). Fractions estimated based on activity in cooling and shield tank after 1 year of operation (ERDA 1977, p. 123).	H-3 (HTO)	6.35E-01
		Be-7	3.05E-01
		Na-22	2.54E-03
		Na-24	5.08E-02
		Cr-51	4.07E-03
		Co-58	1.52E-03
		Co-60	1.02E-03

Facility[a]	Compound[b]	Radionuclide	Activity fraction
Waste Management Incinerator Building 445 (1981–present)	BNL granted a permit by New York in March 1981. Fractions based on averages from 1981 to 1991 for isotopes of importance for internal dose.	Fe-59	9.01E-02
		I-131	1.03E-01
		Mn-54	1.02E-02
		Ru-103	2.87E-02
		Ru-106	2.68E-03
		Tc-99	5.01E-02
		Zn-65	7.15E-01

a. Facilities were combined for this analysis if they were similar and had a common list of radionuclides of concern.
b. If chemical compounds were not available, the best description found is listed.
c. RARAF = Radiological Research Accelerator Facility.

5.11 FACILITY-SPECIFIC RADIONUCLIDE CONVERSIONS

The natural uranium fuel elements for the BGRR were reportedly fabricated by the BNL metallurgy group. The enriched uranium, aluminum-clad elements that replaced the natural fuel were apparently not fabricated at BNL. These data are summarized in Table 5-6.

Table 5-6. Facility-specific radionuclide conversions.

Process description	Activity per unit mass (Bq/g uranium)			
	U-234	U-235	U-236	U-238
BNL metallurgy group, natural U fuel fabrication, late 1940s and possibly up to 1958	1.26E+04	5.59E+02	negligible	1.21E+04

5.12 WORKPLACE MONITORING DATA

If bioassay data are not adequate to evaluate an individual's internal doses, dose reconstructors can use workplace monitoring data (NIOSH 2002). The following types of workplace data might be available for BNL: breathing-zone air samples, general area air samples, and surface contamination surveys. However, these data are not likely to be in individual exposure records. Data on respirator use are not likely to be available. Fit testing records are Industrial Hygiene records maintained by the Safety and Health Services Division. In the case of surface contamination data, site- and process-specific resuspension factors are not likely to be available. During the operation of the HFBR, the airborne tritium contamination was between 1×10^{-6} µCi/cm^3 and 1×10^{-5} µCi/cm^3 on the equipment level. Since the shutdown, the concentration has been between 1×10^{-7} µCi/cm^3 and 5×10^{-7} µCi/cm^3 on the equipment level and between 1×10^{-8} µCi/cm^3 and 5×10^{-8} µCi/cm^3 on the operations level. A reasonable estimate of the maximum exposure can be made by assuming that an individual spent 4 hr/d on the operations level and 4 hr/d on the equipment level (BNL 1999b).

5.13 RADON

For dose reconstruction under EEOICPA, occupational radon exposure is exposure to radon emanating from sources other than those naturally occurring in the area. Dose reconstructors must subtract the natural background level of radon exposure from any measured values when assessing

occupational exposure (NIOSH 2002, p. 32). BNL was not a processing or storage location for large quantities of ^{226}Ra or ^{222}Rn [15]. Onsite environmental monitors occasionally detected low levels of ^{226}Ra between 1987 and 1996 (see Table 4-2). The source was likely the April 26, 1986, accident at the Chernobyl Nuclear Power Plant in Ukraine.

5.14 GUIDANCE FOR DOSE RECONSTRUCTORS

During the evaluation of information for SEC determination, NIOSH determined that bioassay data after 1993 was available from the site. Therefore, if the energy employee has no bioassay data since 1993, it should be assumed that the energy employee did not need to be monitored for internal exposure and only ambient environmental intakes need to be assigned. The following is some additional guidance for tritium dose assessments for all years of employment. If there is no bioassay and no tritium doses reported with external doses, then no tritium dose should be assigned for any given year. If there is no tritium bioassay, but there is a tritium dose with the external report, dose reconstructors should use the larger of:

- The reported dose (lognormal distribution with a GSD of 3; or
- The missed dose (triangular distribution) based on the threshold at which BNL recorded bioassay results or calculated H-3 dose. Based on a review of the records, the threshold for calculating dose was 0.3 µCi/L. The annual missed dose based on the application of the 0.3 µCi would be 11 mrem (0, 0.011, and 0.022 would be the IREP input parameters).

6.0 OCCUPATIONAL EXTERNAL DOSIMETRY

6.1 INTRODUCTION

BNL was established to provide facilities for scientific research which, because of size, complexity, or mode of operation were beyond the means of most single universities. In providing nuclear reactors and particle accelerators for its own staff and perhaps more importantly visiting scientists, emphasis was placed on a multidisciplinary approach to scientific questions. That same mode of operation is still in practice today.

6.2 PURPOSE

The purpose of this section is to describe the external dosimetry systems and practices at BNL and to assist in the evaluation of occupational external exposures from processes at BNL. An objective of this section is to provide supporting technical data to evaluate, with assumptions favorable to the claimant, occupational external doses that can reasonably be associated with radiation exposures to both the monitored and unmonitored worker.

6.3 SCOPE

This section presents historical and current practices in relation to the evaluation of external exposure data for monitored and unmonitored workers. It addresses the evaluation of worker exposure, missed dose, and the bias and uncertainty associated with the monitoring of external dose.

6.4 BNL DOSIMETRY PROGRAM

The Oak Ridge dosimeter was used and processed by BNL from startup through 1984, at which time the Laboratory switched to a vendor (Lane and Reciniello 2004, Yoder 2005). The vendor services continued through 1995. It should be noted that the dosimetry data through 1984 is available only as quarterly summaries, even though the badge exchange period was monthly.

In December 1995, BNL was DOELAP-accredited (Loesch 1995) and started its own program using the Harshaw 8814 and 8806 TLD dosimeters. At times, primarily in the accelerator facilities, these dosimeters were complemented with CR-39 as the neutron dosimeter (Sengupta 2000).

6.4.1 Early Dosimetry for High-Energy Proton Accelerators

Design and construction of the first GeV proton accelerators in the 1950s demanded an increased understanding of high-energy accelerator radiation environments. When a beam of protons in a high-energy (>500 MeV) accelerator strikes its target, a variety of charged and uncharged particles are produced, including neutrons, neutrinos, pions, kaons, muons, electrons, and positrons. The Cosmotron, which operated from 1952 to 1966, was initially only partially shielded. The shielding was improved with time as the beam current and available run times increased. For personnel dosimetry, the desired approach was to measure each component (neutron, photon, etc.) of the radiation dose and multiply by an appropriate relative biological effectiveness (RBE) value to account for the fact that some types of radiation cause a given effect at lower doses than do X-rays and gamma rays. However, initially only very limited information and measurement techniques were available. At BNL a method was devised in which NTA film was used as the primary dosimeter because the beta-gamma film was too insensitive for the weekly exchange cycle being used. Beta-gamma film was included in the badges because they might have been worn in other laboratory areas and would have been useful in case of an overexposure at the accelerator. The NTA film was calibrated with actual machine radiation and a tissue-equivalent ionization chamber. Because the calibration varied considerably with location, a conservative value of mrem per track on the film was used to interpret the results (Cowan 1969). Even though the NTA film responded primarily to neutrons, the calibration was to the total dose equivalent from all radiations. This was viable because neutrons were expected in all locations. NTA film dosimeters were required for all personnel at the facility. Administrative controls were applied to keep people away from the direct beam and areas with high levels of scattered radiation. At the Cosmotron, individuals were primarily exposed to scattered radiation that had an RBE in the range of 3 to 5, but an RBE of 10 was applied to personnel measurements. However, the weekly exchange resulted in poor statistics even for the NTA film (Cowan 1966). After the Cosmotron was fully shielded, conventional methods, using NTA film calibrated with Pu-Be neutrons and beta-gamma film calibrated with ^{60}Co, were used. This method did not underestimate the dose equivalent in the areas where most exposures occurred and considerably overestimated them in other areas. Therefore, it was judged to be satisfactory for routine use (Cowan 1969).

The AGS, which began operations in 1960, was a well-shielded 30-GeV proton accelerator. In such machines, neutrons constitute the primary hazard outside of the shielded areas of the facility (NCRP 2003). Initially the approach to personnel dosimetry used at the Cosmotron was applied. However, in the early 1960s a measurement technique was developed using a tissue-equivalent proportional counter to determine the distribution of linear energy transfer (LET) under specific measurement conditions. The term quality factor (QF), which was more broadly representative of the effects observed in biological studies, replaced RBE for radiation protection purposes. In 1954, values of QF as a function of LET had been published by the NCRP (NCRP 1971). Therefore, it became possible to determine the mean QF without a detailed knowledge of the radiation components. This served as a means of ensuring that the personnel doses as recorded on the NTA and beta-gamma films were providing conservative dose estimates. The term QF has since been replaced by the radiation weighting factors.

From the onset, BNL recognized the hazards of direct exposure to the beams of high-energy protons. Access to areas where the beams were brought out through the shielding, either scattered from a target or by ejection of a portion of the internal beam, had to be carefully controlled. Not only was it less certain that the values of QF being assumed would provide a sufficient safety factor but, in addition, exposure to a direct beam could result in a part of the body being irradiated that did not include the film badge (Cowan 1963, p. 14). Shortly after the AGS commenced operation, BNL found

that a considerable flux of μ-mesons (muons) was penetrating 16 ft of heavy concrete and emerging into the experimental area. They were then faced with determining whether dose due to this flux was being properly evaluated, particularly because high-energy muons are minimum ionizing particles and the NTA film used for personnel monitoring did not record their tracks. Muons decay into high-energy electrons plus neutrinos. The high-energy electrons interact with matter to produce high-energy bremsstrahlung radiation which, in turn, can produce photoneutrons. Alternately, the bremsstrahlung radiation can produce more electrons and positrons by pair production, and the resulting electron-photon cascade can extend through several generations depending on the initial electron energy. Because pair production is the more likely of these last two processes, the number of neutrons present was relatively small. Therefore, the radiation emerging from the concrete into the experimental area consisted of a mixture of muons, electrons, gamma rays, and some neutrons. Two BNL beta-gamma film and NTA badges were exposed near a tissue-equivalent chamber to test the response of their personal monitors to the muon-caused radiation. Evaluation of the film badges by the normal procedure, which by now included the beta-gamma film, convinced the BNL health physics personnel that the dose equivalents being assigned provided an adequate safety factor (Cowan 1963, p. 12).

It appears that the BNL health physics personnel were aware of the unique problems associated with the dosimetry for high-energy accelerators. When information was lacking, they endeavored to choose values of RBE or QF that would overestimate the dose equivalent. However, the missed doses for the weekly exchange period were high. The values of missed dose to be used in dose reconstructions are given in Section 6.5.

6.4.2 Neutron Dosimetry

The neutron doses were measured with NTA film from the beginning of operations through 1995. Particles from high-energy accelerators produced stars from spallation products on the NTA film, and doses were evaluated by counting the number of prongs on the stars. The total number of prongs were related to an effective neutron dose with the result added to the total neutron dose. It was found that the dose from these spallation products was very small. An RBE and later a QF of 10 was used from the very beginning of operations of the accelerators and cyclotrons for these products (Cowan ca. 1953). Vendor-supplied NTA and CR-39 plastic track detectors were tested at the AGS starting in 1986 (Gilmartin 1986f). Lexan was added to the badges for the AGS as of December 5, 1987, and CR-39 was used in the badges for the reactors and NSLS. All badges continued to include NTA film (Schopfer 1987) through 1995.

Due to problems with the Lexan dosimeter, its use was discontinued in 1995. BNL site experts report that the highest of credible dosimeter values were used as the doses of record during this period. Frequently this was the NTA film. Where vendor-supplied detailed records are available, it should be possible to identify which measurement is reported. Column (5) of the report shows which dosimeter was used to assign the neutron dose using the following codes: (P) NTA Film; (B) Neutrak 144 (CR-39); and (E) Neutrak 1 (Lexan) (Kahnhauser 2011). After 1995 the neutron dosimeter for all facilities consisted of a multipurpose TLD badge. A special-purpose albedo TLD badge was also available. To monitor high-energy neutrons from accelerators, a CR-39 dosimeter in a plastic bag was attached to the multipurpose dosimeter (Sengupta 2000).

For a best estimate during the period of overlapping dosimeters (1987 to 1995), dose reconstructors should consider using the recorded value because it is based on the judgment made by the site experts at the time. As stated before, this is likely to be the highest measured dose, but it might or might not include the Lexan dose because that was at times considered to be unreliable. Overestimates could include a sum of all of the measured values that are available for each monitored period, using the assumption that the two (or possibly three) dosimeters responded independently to different energy ranges when, in fact, there was significant overlap.

6.4.3 Special Dosimetry

On occasion, special dosimeters were worn to monitor nonroutine work that resulted in "significantly non-uniform doses to various areas of the whole body" (Sengupta 2000). The special dosimeters were worn without the regular dosimeter. The highest dose measured by the special dosimeters was the dose of record. All measured radiations were recorded as WB dose or dose equivalent. The correct value for the RBE (later the QF) was determined to be conservative; because a value of 10 was always used (Patterson and Thomas 1994; Cowan 1963).

6.4.4 Lost or Destroyed Dosimeters

In the cases of lost or destroyed dosimeters, results were derived from past results of similar work, coworker results, or the product of instrument measurements and time spent in the radiation zone (Reciniello 2006).

6.5 MISSED DOSE

Missed doses for monitored employees at BNL result basically from dosimeter MDLs and exchange periods. In reviewing individual dose records, the exchange period and zeros for that individual can be determined. The review of individual records is necessary because exchange periods varied before 1955 (Cowan ca. 1953). This is applicable for both Hp (0.07) and Hp (10).

Another potential source of historical missed dose is possible error in determining true doses from high-energy particles. However, it has been determined that the values of the RBEs (and later QFs) were conservative and resulted in larger than actual true doses being recorded (Cowan ca. 1953, Cowan undated). Apparently all workers were issued dosimeters from startup through 1954 (Cowan ca. 1953).

Table 6-1 lists the period of use, type of dosimeter, exchange period, MDL, and estimated annual missed dose.

Table 6-1. Estimated annual missed photon, beta, and neutron dose (mrem).

Period of use	Dosimeter	MDL[a]	Exchange frequency	Maximum annual missed dose[b]
Startup–1954	Two-element film	30	Weekly Monthly	780 180
1955–1995	Multielement film	30	Monthly	180
1996–present	Harshaw 8814 TLD	10	Monthly	60
Startup–1995	NTA film[c]	30[d]	Weekly Monthly	780 180
1987–1995	Lexan[c]	30	Monthly	180
1996–present	CR-39[c]	<20	Monthly	120

a. Estimated MDLs for each dosimeter in the workplace even though many doses were reported at less than the MDL.
b. Estimated annual missed dose calculated using MDL/2 from NIOSH (2007).
c. Processing done by RS Landauer. In addition to NTA film, Lexan was added for the AGS and CR-39 was added for the reactors and NSLS from 1987 to 1995. Testing of these plastic dosimeters began in 1986.
d. Based on the analysis of background tracks and historical calibration factors presented in Attachment A.

6.5.1 Unmonitored Worker

Unmonitored workers are those who generally did not enter radiation zones or areas where work with radioactive materials was undertaken. These workers were not issued dosimeters based on the

expectation that the probability of exceeding 10% of the allowable limit was small (Sengupta 2000). Therefore, these workers can be assigned doses as given in Table 4-1 of this document. All radiation workers were monitored from the startup of operations (Cowan ca. 1953).

Doses for unmonitored employees could be as much as 10% of the relevant standard, which was 1,500 mrem/yr (i.e., 300 mrem/wk) in the latter 1940s and then 500 mrem/yr from the early 1950s until 1987. In 1987, a 100-mrem/yr guideline was established before monitoring was required [16].

6.5.2 Annual Average Worker Dose

Radiation workers will likely have dosimetry information available in the files for a claim. For workers who were not considered to be radiation workers, but who occasionally entered facilities with radiation sources, an overestimate of external dose can be obtained from the annual average external dose values presented in this section.

Summary information is available on annual doses received by BNL workers and visitors from 1958 through 2009. This information was gathered and summarized in the SEC Evaluation Report for the first BNL SEC (NIOSH 2009). The following discussion presents a summary of the information provided in that report and additional information used to determine average annual doses for employees and visitors of the Brookhaven National Laboratory. The dose values in the SEC Evaluation Report were supplemented for 1971 (Moser ca. 1972), 2008 (DOE 2010), and 2009 (DOE 2011).

The dose values available are the annual average dose to BNL workers and visitors based only on the individuals with positive measured dose. Because only the individuals with positive measured dose are included, the average values represent a favorable to claimant estimate of dose for individuals who were not likely to be routinely exposed to external radiation fields but might have had intermittent potential for exposure. The term “visitor” as used here represents a non-Brookhaven employee and includes visiting scientists, construction workers, and others who visited the site that were required to be monitored for entry into radiological facilities. The average dose was available for most years, except for 1958 through 1966, 1974, and 1975. For these years, the average available value represented an average over all employees and visitors. The employee and visitor doses were determined from the given overall average using ratios of doses for 1967 to 1973 and 1976 to 1979. For 1967 to 1970, only the employee dose was given. For these years the visitor dose was evaluated based on the binned dose values and the average measured employee dose. The binned data was represented by the number of measured badge readings within each dose bin of 0 to 1 rem, 1 to 2 rem, 2 to 3 rem, 3 to 4 rem, 4 to 5 rem, 5 to 6 rem, 6 to 7 rem, 7 to 8 rem, and 8 to 9 rem. First, an estimate of the average dose was evaluated using the midpoints of the bins multiplied by the number of badges recorded in each bin, summed over all bins. This provided an overestimate of the average dose in comparison with the reported average values. The ratio of the reported average to the estimated average for the employee category was used to adjust the estimated average for the visitors to obtain an average for the visitors.

For 2009 and 2010, the employee average was the only value available. The visitor average was estimated using the 2008 ratio of visitor-to-employee average dose. This provided a higher estimate of visitor dose in comparison with the ratio for other recent years.

No dosimetry information was available for 1947 through 1957. The values for 1958 represent a reasonable estimate for these years because the potential for external dose was less in the early years as facilities and programs were being developed.

In addition to the doses based on reported annual average values, the potential missed dose has also been evaluated based on guidance in ORAUT-OTIB-0020, *Use of Coworker Dosimetry Data for*

External Dose Assignment (ORAUT 2011c). A potential missed dose was evaluated for 11 zero badge readings per year (assuming a monthly badge exchange frequency) to maximize the potential external doses. The annual average dose values and the total dose (including the missed dose) are presented in Table 6-2. The values in this table can be used when the available information indicates the worker might have been exposed intermittently but there are no dosimetry records in the files. Because the records for radiation workers appear always to include external dosimetry information, it is unlikely a radiation worker would not have external dosimetry available. Because the values in Table 6-2 are annual average values they should be used only for unmonitored workers with an indication in their files that they intermittently entered radiation zones. The dose values should be assigned as a constant because they represent an overestimate for unmonitored workers.

Table 6-2. Annual average workplace external dose.

	Employees		Visitors		Year	Employees		Visitors	
Year[a]	Measured	Total[b]	Recorded	Total	Year	Measured	Total	Recorded	Total
1958	0.312	0.477	0.223	0.388	1985	0.215	0.380	0.107	0.272
1959	0.317	0.482	0.227	0.392	1986	0.224	0.389	0.086	0.251
1960	0.297	0.462	0.213	0.378	1987	0.138	0.303	0.056	0.221
1961	0.289	0.454	0.207	0.372	1988	0.145	0.310	0.051	0.216
1962	0.272	0.437	0.195	0.360	1989	0.115	0.280	0.040	0.205
1963	0.271	0.436	0.194	0.359	1990	0.099	0.264	0.047	0.212
1964	0.276	0.441	0.198	0.363	1991	0.115	0.280	0.048	0.213
1965	0.276	0.441	0.198	0.363	1992	0.098	0.263	0.029	0.194
1966	0.315	0.480	0.225	0.390	1993	0.108	0.273	0.039	0.204
1967	0.277	0.442	0.225	0.390	1994	0.150	0.315	0.124	0.289
1968	0.279	0.444	0.219	0.384	1995	0.175	0.340	0.121	0.286
1969	0.192	0.357	0.172	0.337	1996	0.129	0.184	0.033	0.088
1970	0.201	0.366	0.171	0.336	1997	0.086	0.141	0.018	0.073
1971[b]	0.353	0.518	0.280	0.445	1998	0.099	0.154	0.019	0.074
1972	0.385	0.550	0.279	0.444	1999	0.052	0.107	0.024	0.079
1973	0.412	0.577	0.280	0.445	2000	0.058	0.113	0.038	0.093
1974	0.428	0.593	0.306	0.471	2001	0.043	0.098	0.028	0.083
1975	0.455	0.620	0.325	0.490	2002	0.070	0.125	0.018	0.073
1976	0.363	0.528	0.186	0.351	2003	0.046	0.101	0.019	0.074
1977	0.387	0.552	0.245	0.410	2004	0.080	0.135	0.070	0.125
1978	0.271	0.436	0.148	0.313	2005	0.048	0.103	0.013	0.068
1979	0.231	0.396	0.149	0.314	2006	0.041	0.096	0.046	0.101
1980	0.107	0.272	0.023	0.188	2007	0.033	0.088	0.039	0.094
1981	0.185	0.350	0.129	0.294	2008	0.036	0.091	0.043	0.098
1982	0.230	0.395	0.136	0.301	2009	0.029	0.084	0.034	0.089
1983	0.280	0.445	0.122	0.287	2010	0.054	0.109	0.063	0.118
1984	0.253	0.418	0.135	0.300					

a. For 1947 to 1957, use the 1958 values.
b. Total dose includes the annual average plus missed dose.

6.6 RADIATION ENERGIES AND PERCENTAGES AT SELECTED BNL FACILITIES

Section 2.0 describes the different types of facilities at BNL, including reactors, accelerators, and support facilities and target areas. Table 6-3 lists the reactor and accelerator facilities and data related to radiation types and energies. Table 6-4 provides this information for support facilities.

The photon radiation experienced by workers in the accelerator facilities occurs from radiation penetrating the shielding, which is designed primarily for stopping neutron radiation. This radiation tends to be high-energy photons that are able to penetrate the shielding material. A small component of Compton-scattered photons would also be expected in the radiation. Exposure to photons also

occurs when workers enter the inner areas of the accelerator to perform maintenance, target changes, and experiment setup while the facility is shut down. This exposure results from activation products generated during operation. This photon energy is typically of high energy, >250 keV. Therefore, the photon energy range for accelerators that involve proton, neutron, and nuclei beams has been set as indicated in Table 6-3. This photon energy distribution is consistent with the distribution applied to accelerator facilities at other sites.

Table 6-3. Electron and photon energies and percentages for reactors and accelerators.

Process/building	Description	Operations		Radiation type	Energy (keV)	Percentage
		Begin	End			
BGRR	Reactor	1950	1969	Beta Photon	>15 30–250 >250	100 25 75
HFBR/750	Reactor	1965	1999			
BMRR/491	Reactor	1959	2000			
LMFR/820	Reactor	1957	1959			
MRC490/490A	Radioisotopes	1958	Present			
RTF/490	Accelerators	1991	<2006	Electron Photon	>15 >250	100 100
BLIP/931	Accelerator	1973	Present	Electron Photon	>15 30–250 >250	100 10 90
PET/490	Accelerator	1977	Present	Electron Photon	>15 >250	100 100
Cosmotron/same	Cyclotron	1952	1966	Photon	30–250 >250	10 90
RHIC/same	Accelerator	1999	Present			
AGS/same	Synchrotron	1960	Present	Photon	30–250 >250	10 90
AGS experimental area/912	Accelerator	1960	Present			
AGS ring/913		1960	Present			
AGS booster/942		1992	Present			
200 MeV LINAC/930	Accelerator	1972	Present			
Tandem Van de Graaff/901	Accelerator	1970	Present			
Tandem-to-Booster Beam Line/Heavy Ion Beam tunnel	Accelerator	1991	Present			
Accelerator Test Facility/820	Accelerator	1989	Present	Electron Photon	>15 30–250 >250	100 10 90
NSLS/725	Electron storage	1982	Present	Electron Photon	>15 30–250	100 100
Chemistry Dept./490A	Accelerator			Electron Photon	>15 30–250 >250	100 10 90

Table 6-4. Electron and photon energies and percentages for (support facilities).[a]

Process/building	Description	Operations		Radiation type	Energy (keV)	Percentage
		Begin	End			
NSRL[b]	Space Laboratory	2003	Present	Unknown	Unknown	Unknown
Tritium evaporator/802B	Tritium processing	1995	Present	Beta	<15 avg.	100
WCF/811	Waste processing	1947	1987	Beta Photon	>15 30–250 >250	100 25 75
TPL/801	Hot laboratory	1951	Present			
HWMF/444	Incinerator	1970 s	Early 1990s			
Chemistry Dept/555 & 490A	Chemistry R&D	1966	Present			
LEAF/555	Laser electron accelerator	1998	Present			
Instrument Division/356	Material irradiation	1960	Present			
S&EP/348	Instrument calibration	1959	Present			
Biology/463	Biological R&D	1950	Present			
Oceanographic Sciences/318	Oceanographic R&D					
Radiation Sciences/ 703, 703W	Health physics & environmental support					
Nuclear Waste Mgt./830	Waste handling					

a. The isotopes involved ranged from H-3 to Au-198 resulting in beta, electron, and photon energies ranging from a few 10s to several 1000 keV with wide-ranging percentages.
b. NSRL = National Air and Space Administration Space Radiation Laboratory.

6.7 NEUTRON RADIATIONS AND PERCENTAGES

Table 6-5 lists facilities with neutron radiations. They are reactors, accelerators, general laboratories, and medical facilities. The reactors are no longer in operation; the last was shut down in 2000.

Table 6-5. Facilities, neutron energies, percentages, and correction factors.

Facilities	Source	Neutron energy (MeV)	Dose fraction (%)	ICRP 60[a] correction factor
BGRR, HFBR, BMRR	Reactors	0.1–2.0	100	1.91
Cosmotron, AGS, RHIC, NSLS	Accelerators: particle interactions[b]	0.1–2.0	75	1.91
		>2	25	1.32
Medical, calibration, chemistry sources	^{252}Cf	0.1–2	100	1.91
	AmBe, PuBe,	2–10	100	1.32
	Weighted average[c]	0.1–2	57	1.91
		>2	43	1.32

a. ICRP (1991).
b. Select energy appropriate for facility and work activities.
c. Weighted average from alpha-neutron and spontaneous fission neutrons (ORAUT 2009).

Figures 6-1 and 6-2 show the neutron energy information on which Table 6-5 is based. For Figure 6-1, measurements in the thermal column were normalized such that the flux in the bare thermal column was 7×10^8 n/cm^2/s. The thermal flux in the instrument tunnel was 2×10^9 n/cm^2/s. The converter plate was a ^{235}U plate used to convert the thermal beam into a fission spectrum.

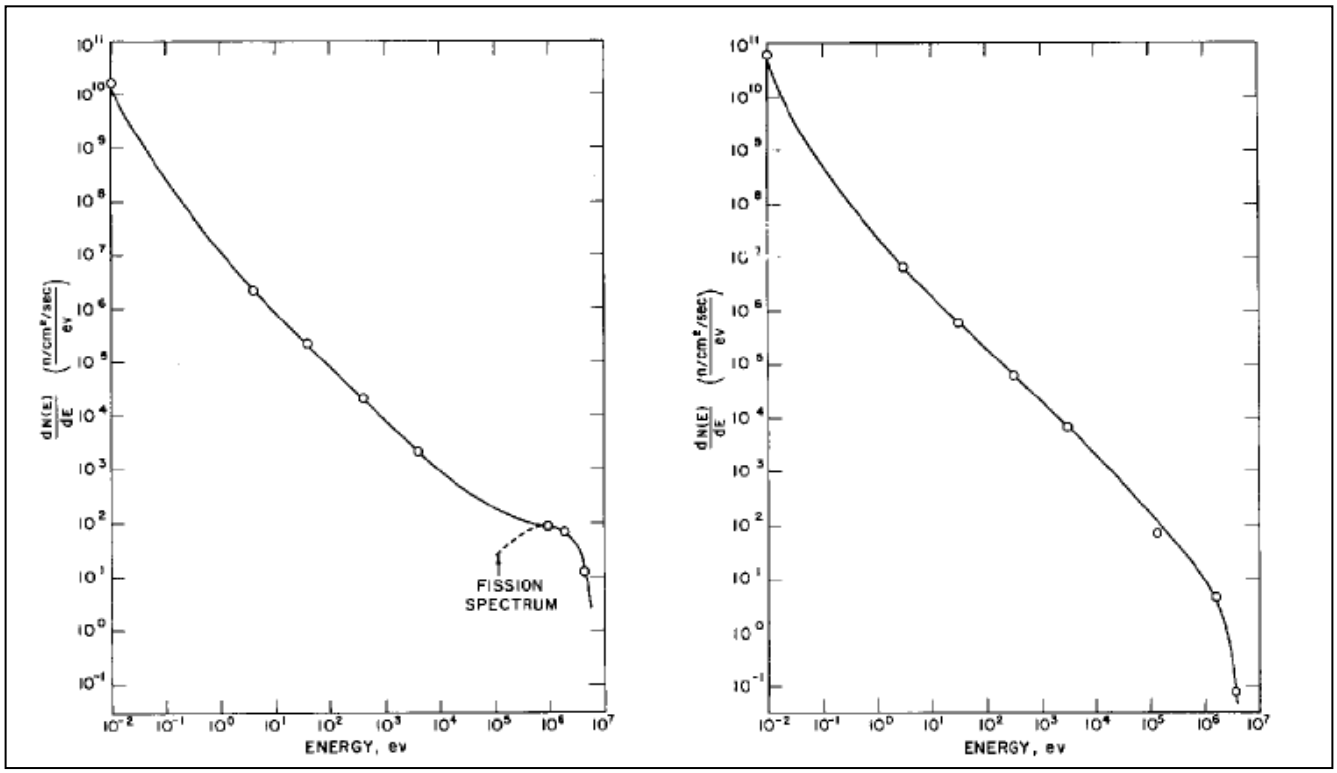

Figure 6-1. Neutron spectrum in the thermal column with the converter plate (left) and in the instrument tunnel (right).

All potential neutron exposures of >100 mrem annually used NTA film through 1995. From start up to 1984, the NTA film was only processed if the density or darkening behind the cadmium (Cd) filter was greater than some predetermined value behind a photon equivalent, but neutron insensitive, filter

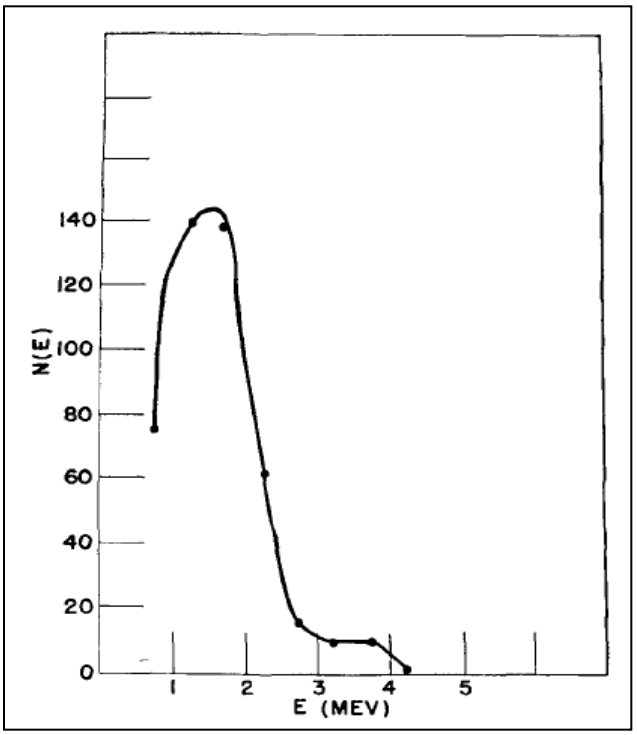

Figure 6-2. Neutron energy spectrum at the Cosmotron (Sanna and O'Brian 1963).

(Faust 2010). This data could also be used to determine thermal neutron doses. Documentation of BNL's method has not been found, but it is thought to be consistent with the practices of other sites (Caruthers and Story 1964; Strom 1982). Caruthers and Story (1964) indicates that a difference in exposure behind the Cd and neutron insensitive filters of 20 mR would be equivalent to about 10 mrem from thermal neutrons. BNL included the dose from thermal and slow neutrons in their fast neutron results. Starting in 1996, the Harshaw 8806 TLD along with CR-39 was used at the accelerators (Lane and Reciniello 2004; Sengupta 2000). Recorded neutron doses include both thermal and fast neutrons generated by the accelerators with energies >100 MeV (Sengupta 2000). The actual "dose of record" always included a QF of 10 starting from the first day of operation (Cowan ca. 1953). DOE is in the process of implementing ICRP Publication 60 (ICRP 1991) neutron weighting factors into the routine determination of the recorded neutron dose. For BNL, the date this was January 1, 2010 (ORAUT 2011d). Therefore, no adjustment in the recorded neutron dose (as described in the last column of Table 6-5) is necessary for records after 2009.

Three available studies of neutron energy spectra provide information for the 60" cyclotron (Xie 1986) and the AGS Experimental Area (Presig 1995 and Presig 1996). The neutron spectra provided in these studies were evaluated based on the five IREP input neutron energy ranges. The neutron energy distribution indicated in Table 6-5 for accelerator facilities was found to provide a favorable to claimant dose assignment compared to the neutron spectra given in the three studies.

6.8 RECORDED DOSE PRACTICES

Recorded dose practices at BNL are given in Table 6-6 and include those provided by both the site and its vendor and any special dosimeter results (Sengupta 2000). Special dosimeters were worn if needed due to abnormal conditions, such as highly directional beams. If more than one dosimeter was required, the highest value was recorded. The regular dosimeter was not worn on these occasions (Sengupta 2000).

6.9 INTERPRETATION OF REPORTED DOSES

Early personnel doses were reported in millirep for both penetrating (photon) and nonpenetrating (beta) if beta or low-energy electrons were present. High-energy beta or electrons were recorded as penetrating dose. The recorded total doses include the results from any special dosimeters worn for that exchange period. If it is necessary to obtain organ doses, dose reconstructors should use

Table 6-6. Recorded dose practices.

Year	Dosimeter measured quantities	Compliance dose quantities
1949–1984	Beta=Open window, mrem[a] Photon (P), mR Fast Neutron (FN), mrem	Skin = OW+P WB = P+ FN+ tritium
1985–1995	Beta or nonpenetrating, mrem Photon (P), mrem Fast Neutron (FN),[b] mrem	Skin = NP + P WB = P + FN + tritium
1996–present	Beta or nonpenetrating, mrem Photon (P), mrem Fast Neutron (FN), mrem	Skin = NP + WB WB = P + FN + tritium

a. At startup, mrep was used interchangeably with mR, mrad, and later mrem.
b. Attached to either dosimeter when neutrons might be present.

OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007). For the period from startup through 1995, the "Exposure (R) to Organ Dose (HT)" dose conversion factors should be applied to the recorded and missed dose. From 1996 and later years, the "Deep Dose Equivalent (Hp(10)) to Organ Dose (HT)" dose conversion factors should be applied.

Table 6-7 lists the interpretation of the reported data by period. Reported doses have been corrected for background using site controls. The controls are dosimeters kept on the site in locations used for the storage of personnel dosimeters. All issued dosimeters were stored on storage racks at positions throughout the site. Dosimeters were not to be taken home at the end of the shift, a practice beginning at the startup of the site and continuing at present (Sengupta 2000).

Table 6-7. Interpretation of reported data.

Period	Reported quantity	Description	Interpretation of zeros	Interpretation of blanks	Rollup of individual and annual data	Monitored/ unmonitored
Startup–1985	Skin = mrad WB = mR Neutrons = mrem	mrep (about startup) mR, mrad, and mrem used interchangeably	MDL/2 times number of zeros	If no dosimeter for that period, treat as unmonitored.	If special dosimeters were used, include results.	Only those >10% of standards were monitored. Those entering controlled areas were issued visitor dosimeters.
1986–present	Skin = mrem WB = mrem Neutrons= mrem	mrem used for all	MDL/2 times number of zeros	If no dosimeter for that period, treat as unmonitored.	If special dosimeters were used, include results.	All personnel expected to be exposed to >100 mrem in an exchange period were monitored.

6.10 ADJUSTMENTS TO RECORDED DOSE

The site used NTA film as the primary means to monitor workers for exposure to fast neutrons from the site startup through 1995. Two issues related to use of NTA film to monitored neutrons are discussed here: NTA neutron track fading and angular dependence.

6.10.1 NTA Track Fading

The problem of NTA neutron track fading was known at the BNL site for many years with the first study of the effect described in monthly reports for November and December of 1950 (BNL 1950). The effect was also reported in 1967 (BNL 1967), 1972 (Distenfeld and Klemish 1972), 1973 (Phillips, Swezey, and Gilmartin 1973), and 1975 (Philips 1975) using standard BNL NTA dosimeters. A summary of these studies is presented in Table 6-8.

Table 6-8. Summary of NTA fading studies at BNL.

Year reported	Exposure	Temperature (°F)	Humidity (%)	Loss/week (%)[a]
1950	Not specified	Ambient	Ambient	9[b]
1967[c]	PuF_4	Ambient	Ambient	12
1972[c]	PuBe	76	43	11.8
1972	PuBe	78	50	18.0
1972	PuBe	72	61	18.3
1972	PuBe	76	Desiccated	4
1972	PuBe	90	71	42
1973	PuBe	Ambient	Desiccated	6
1975	PuBe	Ambient	Desiccated	3–6

a. The weekly loss values are evaluated based on a power function of time.
b. Based on one-third of tracks lost in 1 month.
c. Data for the 1967 and 1972 studies are averaged over a 4-week period.

Results of the 1972 study indicate the loss rate is dependent on temperature and humidity. The majority of neutron exposures at BNL were associated with work at the AGS in well-ventilated aircraft type hanger buildings with inside conditions similar to those outside (Distenfeld and Klemish 1972 p 6). The 1967 study indicated that during the summer months the temperature and humidity would be higher than other periods of the year and fading could be of concern, although, a 2-week wear period was considered to be permissible (BNL 1967). Desiccation of the film in badges enclosed in thin plastic sealers was being investigated at that time.

For the period from site startup through 1984, a fading correction factor of 18% per week provides a reasonable overestimate of fading for NTA film dosimeters on an annual basis. For a monthly wear period with 1 week for reading of the dosimeter, an average time for fading would be 3 weeks. This corresponds to a loss of about 55% of the tracks. The recorded neutron doses should be corrected by applying a multiplication factor of 1.81 (inverse of 55%).

For 1985 through 1995, the NTA film dosimeters were provided by and read by Landauer. The calibration methods used by Landauer involved exposure of calibration dosimeters at the same time the batch of worker dosimeters was prepared. The calibration dosimeters were also read at the same time as the worker dosimeters (Faust 2011). This method ensured that track fading effects were inherently included in the calibration process. Therefore, no correction for track fading is necessary for this period.

6.10.2 NTA Angular Dependence

Kathren, Prevo, and Block (1965) studied the response of NTA film dosimeters as a function of neutron incidence angle for neutrons of energy 0.87 to 6.19 MeV. The data presented in that paper

are based on the response of the film expressed as “tracks per field dose” and indicate that for neutrons above about 2 MeV the film response is low. This implies that for NTA film dosimeters calibrated with the beam at 90 degrees to the badge, the calibration curve results in an underestimate of dose for fast neutrons if the worker is not exposed entirely in the AP geometry. For neutron energies above about 3.5 MeV, the response was a maximum at 90 degrees and lower at lower angles. For lower neutron energies, a correction is not needed because there was an over-response of the film for some angles. For fast neutrons, this corresponds to a correction factor of 1.33 applied to recorded doses. The calibration angle of 90 degrees corresponds to an exposure geometry of 100% AP. However, the application of such a correction factor is only appropriate if the worker was likely exposed in a field of neutrons impacting the worker from many directions. The use of the correction factor would not be appropriate for workers who were exposed in an AP geometry because the calibration matches the exposure conditions. Most dose reconstructions use the DCFs for AP geometry because these DCFs are the highest except for a few organs (red bone marrow, bone surface, esophagus, lung, and remainder organs) where the PA DCFs are greater.

Calculations were performed to compare exposures in the AP geometry (without angular dependence correction) and in the rotational geometry (with angular dependence correction). The calculations were performed for the neutron energy range from 2 to 20 MeV. The results indicate that for a few organs, use of the corrected rotational geometry results in a higher estimate of recorded dose. Table 6-9 indicates the ratio of corrected dose (based on rotational geometry) to the dose for AP geometry without correction. A value greater than 1 indicates the rotational geometry DCFs with angular correction would give a higher neutron dose for cases where the DCF values from OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007) are used. As discussed above, the angular correction factor is applicable only to the energies above 2 MeV and should not be applied to lower energy neutrons. The factors in Table 6-9 can be applied to the recorded neutron dose calculated using the AP DCFs for the appropriate external organ. Figure 6-3 is a logic diagram for application of the angular correction factors in Table 6-9. A favorable to claimant approach would be to assign the correction only to organs having a correction factor greater than 1. For application of the best estimate methods (ORAUT 2005), the factor must be applied as part of the Monte Carlo analysis and not as a correction factor for all geometries.

Table 6-9. Ratio of NTA film neutron dose for rotational geometry with angular correction to AP geometry.

External organ of reference	Ratio: corrected rotational/AP
Bladder	0.84
Bone (red marrow)	1.31
Bone (surface)	1.28
Colon	0.96
Esophagus	1.13
Breast (female)	0.87
Liver	1.02
Lung	1.12
Gonads (female–ovaries)	1.06
Remainder organs	1.17
Skin	1.33
Stomach	0.86
Gonads (male–testes)	0.80
Thyroid	1.05
Lymphoid tissue	0.96

As an example of application of the NTA film correction factor, consider the neutron energy distribution for accelerator facilities from Table 6-5. Assume there are positive neutron doses reported in the dosimetry records for the worker prior to 1996. First, the neutron dose for each energy range is evaluated using the AP geometry dose factors for the organ of reference. The calculated dose in the

0.1 – 2 MeV neutron energy range is assigned without application of an angular correction factor. If the geometry is not AP, the calculated dose is in the 2 – 20 MeV neutron energy range is multiplied by the correction factor from Table 6-9 for the reference organ. If the dose is being evaluated as an overestimate, then only angular correction factors greater than 1.00 need be applied.

6.11 VOLUNTARY NEUTRON RADIATION EXPERIMENTS

From the early 1970s to the mid-1980s studies were conducted in which voluntary subjects were exposed to neutron radiation to measure body content of various elements.

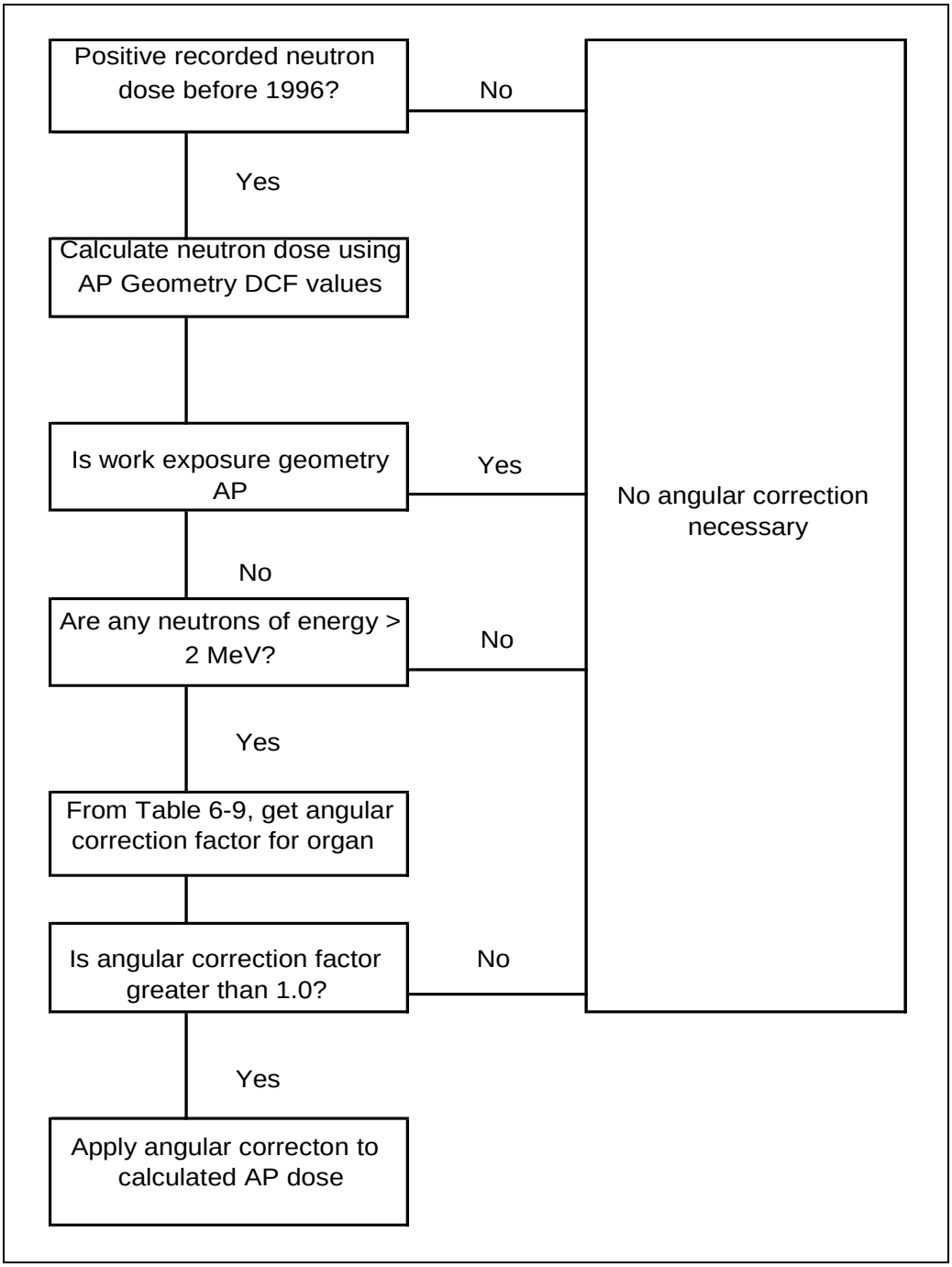

Figure 6-3. Logic diagram for application of NTA film angular correction factor.

Prompt-gamma WB neutron activation analysis was performed as part of a research program to investigate the interrelationship between diet, nutrition, and body composition. An essential requirement in the study was the measurement of the protein muscle mass or specifically total body nitrogen. The activation analysis was performed by placing the subject on a cot that moved over a neutron source for 30 minutes. The subject was exposed twice, once in the prone position and once in the supine position. The neutron source was a ^{238}PuBe neutron generator that produced fast neutrons of about 5 MeV. The patient was exposed to the fast neutron beam with a premoderator of deuterium oxide in a tank between the source and the patient. The neutrons reaching the subject were still considered to be fast neutrons (Vartsky, Ellis, and Cohn 1979). During the exposure period the entire body received a neutron irradiation that activated the body nitrogen and hydrogen for a short period. While the subject was being irradiated, the induced isotope gamma radiations were counted by appropriate detectors. The received dose was reported to be 95.4% (for a QF of 10) from fast neutrons and 4.6% from high-energy gamma radiation (Vartsky, Ellis, and Cohn 1979). The total reported dose for the procedure varied with time due to modifications and upgrades to the system. Dose assignment should be based on the dose to the employee from the medical records for the claim. If no dose is given, then the reported value of 26 mrem should be assigned (Vartsky et al. 1979), using the above neutron/photon distribution.

Delayed-gamma WB neutron activation analysis was used to evaluate calcium content of the body. To perform the analysis, the subject was first counted in a WBC to determine the background radiation level in the body. The subject was then moved to the neutron activation facility and positioned on a moveable cot with the polyethylene moderator material placed in position around the subject. The cot was then positioned in the neutron irradiation facility and the subject was irradiated for 5 minutes. A PuBe neutron source was used for the irradiations. After irradiation, the subject was returned rapidly to the WBC to count the induced radioactivity produced. The site calculated a WB dose of 0.277 rem, 92.6% (for a QF of 10) from fast neutrons and 7.4 percent from high-energy gamma radiation (Cohn et al. 1972). If a different dose is indicated in the medical records, the indicated dose should be used with the above neutron/photon distribution. If no dose is given, then the 0.277 value should be assigned using the above neutron/photon distribution.

The neutron doses should be assigned to the 2- to 20-MeV energy range with no correction for angular dependence because the beam irradiates the body in the AP and PA geometries rather than a rotational geometry. An ICRP Publication 60 correction factor of 1.32 should be applied to the neutron dose evaluation (ICRP 1982). The gamma dose component should be assigned to the >250-keV energy range.

The medical records for the energy employee should be reviewed for indications of participation in the voluntary neutron exposure studies. If information in the telephone interview record, DOL records, or DOE records indicates the employee participated in these studies but the actual exposures are not included in the records, the dose reconstructor should request the voluntary study medical records from the MRC at BNL.

6.12 BIAS AND UNCERTAINTY

No site-specific data has been found about the bias and uncertainty for 1947 to 1984 when BNL used in-house film badges with NTA film. BNL did participate in some film badge performance tests in the mid-1960s, which can be used as an indication of the performance that was achieved.

In one test, 12 commercial film badges suppliers and two national laboratories were included in a blind intercomparison study of the accuracy and consistency of film badge readings that were exposed in March 1963. BNL was one of the national laboratories according to the acknowledgements. The national laboratories were not tested for consistency (precision), but their performance was consistent with the better-performing group of commercial processors. The study

concluded that the accuracy that could be reasonably expected in practice, with a confidence limit of 90%, was in the range of –50% to +200% for groups of badges exposed to X-rays, gamma rays, and mixtures thereof, over the exposure range of approximately 20 mR to 8,000 mR (Gorson, Suntharalingam, and Thomas 1965).

A 1967 study to determine film dosimeter performance criteria included participants from 15 AEC laboratories (Unruh et al. 1967). Although the results did not identify the laboratories, it is logical that BNL would have participated. Averaged over all radiation categories, about 90% of the tests were passed by AEC contractors. The bias performance criterion was 90% to 110% (within 10%).

In 1985, BNL contracted with R. S. Landauer Company for film badge services. During this period the DOE published criterion under the DOELAP for dosimeters (DOE 1986) that required 30% accuracy with 95% confidence based on use of four quarterly badges to evaluate the annual dose. The criterion applied to beta, photon, and neutron dosimeters. Table 6-10 summarizes the specifications from Landauer in 1988 (Schopfer 1988).

Table 6-10. Landauer gamma film, NTA film, and CR-39 specifications.

Radiation category	Source used	Range maximum (mrem)	Mean accuracy[a] (mrem)	Precision (1 σ)
High-energy photon	Cs-137	50	20	20 mrem
		500	50	10%
		5000	500	10%
High-energy X-ray		50	20	20 mrem
		500	75	15%
		5000	750	15%
Low-energy X-ray	Am-241 15-20 keV 55-65 keV	50	20	20 mrem
		500	75	25%
		5000	750	25%
Fast neutron NTA	AmBe	50	20	20 mrem
		500	100	20%
		5000	1000	15%
Fast neutron CR-39	AmBe	50	20	20 mrem
		500	100	20%
		5000	1000	15%

a. Mean accuracy is the limit of the average difference of result of a group of dosimeters from the delivered exposure or dose.

A few reports of quality control checks on Landauer's performance were found. A report from 1986 showed that Landauer was generally able to meet the "mean accuracy" specification (Schopfer 1986). In a 1988 report, the bias (average relative error) over the whole test range (up to 5,000 mrem) for beta-gamma, CR-39, NTA, and Lexan dosimeters was –0.09, –0.02, –0.17, and –0.03, respectively. It was noted, however, that the precision in this test failed to meet expectations (Schopfer 1988).

Based on the above information, an uncertainty of 30% is assumed for the period from 1985 to 1995 for all dosimeters.

In December 1995, BNL received DOELAP accreditation in external dosimetry (Loesch 1995). In 1996, the Landauer film badge service was replaced by TLD system processing on the site. The Landauer-supplied CR-39 dosimeters were continued. Bias and uncertainty from 1996 on are expected to be consistent with DOELAP standards. To be granted DOELAP accreditation, BNL was required to demonstrate compliance with dosimeter system requirements. This compliance was demonstrated in performance testing of the BNL dosimetry system (Cummings 1997). The tests involved exposing sets of 15 dosimeters to various radiation fields including beta, photon, neutron,

and some combinations of these. The tested dosimeters included Harshaw 8814 (TLD-600, -700), Harshaw 8806 (TLD-600, -700), and CR-39. Of the 360 dosimeter tests, 97.8% were within the 30% overall uncertainty requirement. The average uncertainty for each set of 15 dosimeters was under 5% including uncertainties due to source standardization, dosimeter position, and scattered radiation not stemming from the phantom, but excluding uncertainty due to the dose equivalent conversion factors and the photon component of the neutron irradiations (DOE 1986). The overall bias of the test was centered about 0.0, but some of the tests showed bias as high as about 14% for the Harshaw 8814 dosimeters (overestimate of dose) to as low as about 20% for the CR-39 neutron dosimeter (underestimate of dose).

The results were similar for all types of radiations and tested dosimeters. Based on this information, an uncertainty of 30% is assumed for the period from 1996 to the present. Table 6-11 summarizes the uncertainty factors for BNL dosimetry.

Table 6-11. Uncertainty factors.

Dosimeter	Uncertainty (%)[a]
Two-element film dosimeter (1947–1963) beta/gamma	40
Multi-element Film (1964–1984) beta/gamma	30
NTA Film (1954–1984) (2–14 MeV) neutron	30
NTA Film (1954–1984) (0.1–2 MeV) neutron	30
NTA Film (1985–1995) Landauer, beta/gamma/neutron	30
Harshaw 8814 (1996–present) beta/gamma/neutron	30
Harshaw 8806 (1996–present) beta/gamma/neutron	30
CR-39 (1996–1997)	30

a. Estimated variability of measured dose based on dosimeter technology as used in site radiation fields for long-term workers.

7.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database (SRDB).

[1] Potter, Eugene. Oak Ridge Associated Universities (ORAU) Team. Principal Health Physicist. November 2009.
Figure 2-2 was taken from a BNL document that has since been removed from the BNL web site. It was annotated by the section's author with names and arrows that point out the major radioactive material and radiation-producing facilities. This figure is similar to figures that appear in a number of BNL site environmental reports, which may be consulted for comparison.

[2] Elyse Thomas. ORAU Team. Principal Medical Dosimetrist. December 2009.
Review of claim file records from BNL.

[3] Elyse Thomas. ORAU Team. Principal Medical Dosimetrist. December 2009.
Review of claim file records from BNL.

[4] Elyse Thomas. ORAU Team. Principal Medical Dosimetrist. December 2009.
Review of claim file records from BNL.

[5] Robert Morris. ORAU Team. Principal Health Physicist. August 2006.
Review of actual films at BNL.

[6] Robert Morris. ORAU Team. Principal Health Physicist. August 2006.
Review of actual films at BNL.

[7] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
Conclusions were based on a review of the environmental reports listed in the References at the end of this revision.

[8] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
The factor of 1.5 is suggested in ORAUT-PROC-0031 and is not accounted for separately in the tool used for dose reconstruction.

[9] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
A reduction factor of 0.01 was selected as a conservative value after considering a number of different scenarios and typical values as input to atmospheric transport models.

[10] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
This statement is based on professional judgment after reviewing the available data in a number of references.

[11] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
The uncertainty range values quoted are based on professional judgment and experience.

[12] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
These conclusions are based on the review of a number of WB count results in individual dosimetry files.

[13] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
These conclusions are based on the review of a number of WB count results in individual dosimetry files, as well as other records and interviews.

[14] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
These conclusions are based on the review of a number of bioassay data packages and WB count records.

[15] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
Based on a review of records and site interviews.

[16] Potter, Eugene. ORAU Team. Principal Health Physicist. November 2009.
This statement is based on the historical monitoring thresholds.

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GLOSSARY

absorbed dose
Amount of energy (ergs or joules) deposited in a substance by ionizing radiation per unit mass (grams or kilograms) of the substance and measured in units of rads or grays. See *dose*.

absorption type
Categories for materials according to their rate of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively in the respiratory tract (slow solubilization). Also called solubility type. See *inhalation class*.

accelerator
See *particle accelerator*.

accreditation
For external dosimetry, the assessment of whether or not a personnel dosimetry system meets specific criteria. The assessment includes dosimeter performance and the associated quality assurance and calibration programs.

accuracy
The characteristics of an analysis or determination that ensures that both the bias and precision of the resultant quantity will remain within the specified limits.

activity median aerodynamic diameter (AMAD)
Diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

alpha particle (α)
See *alpha radiation*.

alpha radiation
Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2.

ampere (A)
International System unit of electrical current equal to 1 coulomb per second.

anterior–posterior (AP)
Physical orientation of the body relative to a penetrating directional radiation such that the radiation passes through the body from the front to the back. See *exposure geometry*.

backscatter
Reflection or refraction of radiation at angles over 90 degrees from its original direction.

beam quality
Empirical measure of the ability of a polyenergetic X-ray beam to penetrate matter affected by the kilovoltage, anode material, voltage waveform, and filtration of an X-ray tube. The half-

value layer in millimeters of aluminum is a typical measure of X-ray beam quality for the energy range used in radiography. Also called beam hardness. See *filtration*.

becquerel (Bq)
International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion (3.7×10^{10}) Bq.

beta particle (β)
See *beta radiation*.

beta radiation
Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

bioassay
Measurement of amount or concentration of radionuclide material in the body (*in vivo* measurement) or in biological material excreted or removed from the body (*in vitro* measurement) and analyzed for purposes of estimating the quantity of radioactive material in the body.

cladding
The outer layer of metal that encases a reactor fuel element or fissile material of the pit of a nuclear weapon, often made with aluminum or zirconium. In a reactor, cladding promotes the transfer of heat from the fuel to the coolant, and it builds up fission and activation products over time from the fission of the fuel.

Columbia Resin Number 39 (CR-39)
Radiosensitive material used in track-etch neutron dosimeters.

contamination
Radioactive material in an undesired location including air, soil, buildings, animals, and persons.

core
Central region of a nuclear reactor where fission of the fuel takes place.

Cosmotron (Synchrotron)
Accelerator in which charged particles (electrons, protons, ions, etc.) are accelerated in a circular path achieving very high energies.

criticality
State of a radioactive mass (e.g., the core of a nuclear reactor) when the fission reaction becomes self-sustaining.

curie (Ci)
Traditional unit of radioactivity equal to 37 billion (3.7×10^{10}) becquerels, which is approximately equal to the activity of 1 gram of pure ^{226}Ra.

cyclotron
Particle accelerator capable of large beam currents where the beam is injected in the center of a circular magnet. A fixed radio frequency signal applied to two D-shaped electrodes

accelerates the beam as it passes from one electrode to the other as the potential alternates. The radius of the beam increases as the energy increases.

decommissioning

Removal of a facility from service, usually involving decontamination of radioactivity to specified levels and often involving demolition of the facility.

decontamination

Reduction or removal of radioactive material from a structure, area, object, or person. Decontamination can occur through (1) treating the surface to remove or decrease the contamination or (2) allowing natural radioactive decay to occur over a period of time.

deep dose equivalent [Hp(10)]

Dose equivalent in units of rem or sievert for a 1-centimeter depth in tissue (1,000 milligrams per square centimeter). See *dose*.

disintegrations per minute (dpm, d/m)

Measure of radioactivity equal to the number of nuclear disintegrations in a mass per minute; 1 dpm equals 1/60 becquerel.

DOE Laboratory Accreditation Program (DOELAP)

Program for accreditation by DOE of DOE site personnel dosimetry and radiobioassay programs based on performance testing and the evaluation of associated quality assurance, records, and calibration programs.

dose

In general, the specific amount of energy from ionizing radiation that is absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays.

dose equivalent (H)

In units of rem or sievert, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

dosimeter

Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual.

dosimetry

Measurement and calculation of internal and external radiation doses.

dosimetry system

System for assessment of received radiation dose. This includes the fabrication, assignment, and processing of external dosimeters, and/or the collection and analysis of bioassay samples, and the interpretation and documentation of the results.

electron

Basic atomic particle with negative charge and a mass 1/1,837 that of a proton. Electrons surround the positively charged nucleus of the atom.

electron-volt (eV)

Unit equal to the energy of one electron moving through a potential difference of 1 volt (1.602×10^{-19} joules). The common units in nuclear physics and radiology are kiloelectron-volts (thousands) and megaelectron-volts (millions).

element

One of the known chemical substances in which the atoms have the same number of protons. Elements cannot be broken down further without changing their chemical properties. Chemical symbols for the elements consist of either a single letter or a combination of letters, some of which descend from the Latin names [e.g., Au from *aurum* (gold), Fe from *ferrum* (iron)]. This glossary indicates *elements* by their names. Specific *isotopes* appear as their standard chemical symbols with the number of protons and neutrons in the nucleus.

Energy Employees Occupational Illness Compensation Program Act of 2000, as amended (EEOICPA; 42 U.S.C. § 7384 *et seq.*)

Law that provides for evaluation of cause and potential compensation for energy employees who have certain types of cancer.

entrance skin exposure (ESE)

Exposure in air, measured in units of roentgens, at the point of entry into the body (without backscatter).

exchange period (frequency)

Period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

exposure

(1) In general, the act of being exposed to ionizing radiation. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

exposure geometry

Orientation (physical positioning) of a person or object in relation to a radiation source. This geometry is a factor in the radiation dose to various parts of the body. See *anterior–posterior*, *posterior–anterior*, and *lateral* in relation to radiology.

favorable to claimants

In relation to dose reconstruction for probability of causation analysis, having the property of ensuring that there is no underestimation of potential dose, which often means the assumption of a value that indicates a higher dose than is likely to have actually occurred in the absence of more accurate information. See *probability of causation*.

film

(1) In the context of external dosimetry, radiation-sensitive photographic film in a light-tight wrapping. See *film dosimeter*. (2) X-ray film.

film dosimeter

Package of film for measurement of ionizing radiation exposure for personnel monitoring purposes. A film dosimeter can contain two or three films of different sensitivities, and it can contain one or more filters that shield parts of the film from certain types of radiation. When developed, the film has an image caused by radiation measurable with an optical densitometer. Also called film badge.

filter

Material used (1) in a dosimeter to adjust radiation response to provide an improved tissue equivalent or dose response and (2) in an X-ray machine to selectively absorb photons from the beam to reduce unnecessary exposure of individuals or to improve radiographic quality.

filtration

The process of filtering an X-ray beam, usually with millimeter thicknesses of aluminum material between the X-ray source and the film that preferentially absorbs photons from the beam. Usually measured in equivalent millimeters of aluminum. See *beam quality* and *half-value layer*.

fission

Splitting of the nucleus of an atom (usually of a heavy element) into at least two other nuclei and the release of a relatively large amount of energy. This transformation usually releases two or three neutrons.

fission product

Radionuclides produced by fission or by the subsequent radioactive decay of radionuclides.

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

gamma ray

See *gamma radiation*.

gray (Gy)

International System unit of absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium; 1 Gy equals 1 joule per kilogram or 100 rads.

half-value layer (HVL)

Thickness of a specified substance, usually specified in equivalent millimeters of aluminum, which, when introduced in the path of a given beam of radiation, reduces the kerma rate by one-half. See *filtration*.

hot cell

Shielded laboratory for handling of radioactive materials with the aid of remotely operated manipulators. The walls and windows are made of materials that protect workers from radiation.

***in vitro* bioassay**

Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

***in vivo* bioassay**

The measurements of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

inhalation class

Former respiratory tract inhalation classification scheme developed by the International Council on Radiological Protection for inhaled material according to its rate of clearance from the pulmonary region of the lung. Materials were classified as D (days, half-life less than 10 days), W (weeks, 10 to 100 days), or Y (years, more than 100 days). See *absorption type*, which superseded this concept.

internal dose

Dose received from radioactive material in the body.

ionizing radiation

Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons. See *alpha radiation*, *beta radiation*, *gamma radiation, neutron radiation*, *photon radiation*, and *X-ray radiation*.

isotope

One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U, ^{235}U, and ^{238}U). Isotopes have very nearly the same chemical properties. See *element*.

kerma

Measure in units of absorbed dose (usually grays but sometimes rads) of the energy released by radiation from a given amount of a substance. Kerma is the sum of the initial kinetic energies of all the charged ionizing particles liberated by uncharged ionizing particles (neutrons and photons) per unit mass of a specified material. Free-in-air kerma refers to the amount of radiation at a location before adjustment for any external shielding from structures or terrain. The word derives from kinetic energy released per unit mass.

lateral (LAT)

Orientation of the body during an X-ray procedure in which the X-rays pass from one side of the body to the other. See *exposure geometry*.

linear accelerator (LINAC)

Straight single-pass particle accelerator in which radio frequencies accelerate the beam over the length of the accelerator.

lumbar spine

Region of the spine including the five lumbar vertebrae (lower back).

megaelectron-volt (MeV)

Unit of particle energy equal to 1 million (1×10^6) electron-volts.

milliampere-second (mAs)

In relation to radiography, product of the average X-ray beam current in milliamperes and the time of the exposure in seconds. These factors are selectable on the control panel of most medical X-ray equipment.

natural uranium
Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by mass. The specific activity of this mixture is 2.6×10^7 becquerel per kilogram (0.7 microcuries per gram). See *uranium*.

neutron
Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

neutron film dosimeter
Film dosimeter with a nuclear track emulsion, type A, film packet.

neutron radiation
Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons. See *neutron*.

nonpenetrating dose (NP, NPEN)
Dose from beta and lower energy photon (X-ray and gamma) radiation that does not penetrate the skin. It is often determined from the open window dose minus the shielded window dose. See *dose*.

nuclear emulsion
Thick photographic coating in which the tracks of various fundamental particles show as black traces after development. The number of tracks in a given area is a measure of the dose from that radiation. See *nuclear track emulsion, type A*.

nuclear energy
Energy released by nuclear reaction, some of which can be ionizing radiation. Of particular importance is the energy released when a neutron initiates fission or when two nuclei join together under millions of degrees of heat (fusion). Also called atomic energy.

nuclear track emulsion, type A (NTA)
Film sensitive to fast neutrons made by the Eastman Kodak. The developed image has tracks caused by neutrons that become visible under oil immersion with about 1,000-power magnification.

nuclide
Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

occupational dose
Internal and external ionizing radiation dose from exposure during employment. Occupational dose does not include that from background radiation or medical diagnostics, research, or treatment, but does include dose from occupationally required radiographic examinations that were part of medical screening.

occupational environmental dose
Dose received while on the grounds of a site but not inside a building or other facility.

occupational medical dose

Dose from X-ray procedures performed for medical screening of workers as part of an occupational health program. Doses from X-rays used to diagnose diseases or injuries, even if incurred on the job, are not considered occupational and are therefore not eligible to be included in dose reconstruction under EEOICPA.

open window (OW)

Area of a film dosimeter that has little to no radiation shielding (e.g., only a holder and visible light protection). See *film dosimeter*.

particle accelerator

Device that accelerates ions using magnetic and/or electrostatic fields for focusing and redirecting ion beams. The main purposes of accelerators are the investigation of high-energy particle behavior and production of synthetic isotopes.

penetrating dose (PEN)

Dose from moderate to higher energy photons and neutrons that penetrates the outer layers of the skin. See *dose*.

personal dose equivalent [$H_p(d)$]

Dose equivalent in units of rem or sievert in soft tissue below a specified point on the body at an appropriate depth *d*. The depths selected for personal dosimetry are 0.07 millimeters (7 milligrams per square centimeter) and 10 millimeters (1,000 milligrams per square centimeter), respectively, for the skin (shallow) and whole-body (deep) doses. These are noted as $H_p(0.07)$ and $H_p(10)$, respectively. The International Commission on Radiological Measurement and Units recommended $H_p(d)$ in 1993 as dose quantity for radiological protection.

phantom

Any structure that contains one or more tissue substitutes (any material that simulates a body of tissue in its interaction with ionizing radiation) and is used to simulate radiation interactions in the human body. Phantoms are primarily used in the calibration of *in vivo* counters and dosimeters.

photofluorography (PFG)

Historical radiographic technique used for chest images for screening a large number of people in a short period of time. The X-ray image produced on a fluorescent screen was photographed on 4- by 5-inch film. PFG was the primary method of screening large populations for tuberculosis before the advent of nonradiographic screening methods. Also called fluorography or mass miniature radiography. Not to be confused with *fluoroscopy*.

photon

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

photon radiation

Electromagnetic radiation that consists of quanta of energy (photons) from radiofrequency waves to gamma rays.

PM

A procedure detailing specific actions or directions and usually limited to one service or activity.

posterior-anterior (PA)

Physical orientation of the body relative to a penetrating directional radiation field such that the radiation passes through the body from the back to the front. See *exposure geometry*.

probability of causation (POC)

For purposes of dose reconstruction for the Energy Employees Occupational Illness Compensation Program Act, the percent likelihood, at the 99th percentile, that a worker incurred a particular cancer from occupational exposure to radiation.

proton

Basic nuclear particle with a positive electrical charge and mass slightly less than that of a neutron. There are protons in the nuclei of every atom, and the number of protons is the atomic number, which determines the chemical element. See *element*.

quality factor (QF)

Principal modifying factor (which depends on the collision stopping power for charged particles) that is employed to derive dose equivalent from absorbed dose. The quality factor multiplied by the absorbed dose yields the dose equivalent. See *dose*, *relative biological effectiveness*, and *weighting factor*.

rad

Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joules per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rads = 1 gray). The word derives from radiation absorbed dose.

radiation

Subatomic particles and electromagnetic rays (photons) with kinetic energy that interact with matter through various mechanisms that involve energy transfer. See *ionizing radiation*.

radioactivity

Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei. See *radionuclide*.

radiograph

Image produced on film by gamma rays or X-rays. See *radiology*.

radiography

The process of producing images on film (or other media) with radiation.

radiology

Medical science and specialty of producing images on radiographic film or other media, which are used to identify, diagnose, and or treat diseases, injuries, or other conditions.

radionuclide

Radioactive nuclide. See *radioactive* and *nuclide*.

reactor

Device in which a fission chain reaction occurs under controlled conditions to produce heat or useful radiation for experimental purposes or to generate electrical power or nuclear fuel.

reactor elements

Fabricated fuel and target components inserted into reactors.

relative biological effectiveness

Ratio of the absorbed dose of a reference radiation to the absorbed dose of a test radiation that produces the same biological effects, other conditions being equal. A factor applied to account for differences between the amount of cancer effect produced by different forms of radiation.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

rep

Historical quantity of radiation (usually other than X-ray or gamma radiation) originally defined as 93 ergs absorbed per gram in the body and redefined in the 1940s or early 1950s as the amount that would liberate the same amount of energy (93 ergs per gram) as 1 roentgen of X- or gamma rays. Replaced by the gray in the International System of Units; 1 rep is approximately equal to 9.3 milligray. The word derives from roentgen equivalent physical.

roentgen (R)

Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58×10^{-4} coulombs per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0°C and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

shallow dose equivalent [SDE, H_s, $H_p(0.07)$]

Dose equivalent in units of rem or sievert at a depth of 0.07 millimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

sievert (Sv)

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 Sv equals 100 rem.

skin dose

See *shallow dose equivalent*.

technique

Combination of X-ray machine settings used to produce radiographs, which consists of the kilovoltage, tube current (milliamperes), and exposure time (seconds). The last two parameters are often multiplied to yield the electric charge that has crossed the X-ray tube during the exposure in units of milliampere-seconds. Any combination of time and tube current that produces a given product in milliampere-seconds produces the same exposure for a fixed peak kilovoltage.

source-to-image distance (SID)
Distance from the X-ray machine target (anode) to the plane of the image receptor (film). This distance is standardized for typical radiographic procedures. Chest X-rays, for example, are performed at a 72-inch SID.

tandem Van de Graaff accelerator
Accelerator in which charge exchange (negative to positive) occurs in the terminal on either a thin foil or in a gas stripper tube.

thermoluminescent dosimeter (TLD)
Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

tissue equivalent
Substance with response to radiation equivalent to tissue. A tissue-equivalent response is an important consideration in the design and fabrication of radiation measuring instruments and dosimeters.

type
See *absorption type*.

U.S. Atomic Energy Commission (AEC)
Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The U.S. Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

weighting factor
The ratio of the stochastic risk arising from tissue *T* to the total risk when the whole body is irradiated uniformly.

whole-body counter (WBC)
Equipment used to perform *in vivo* bioassay. Radiation emitted from radioactive material deposited throughout the body is measured.

whole-body (WB) dose
Dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder and commonly defined as the absorbed dose at a tissue depth of 10 millimeters (1,000 milligrams per square centimeter). Also called penetrating dose. See *dose*.

X-ray
(1) See *X-ray radiation*. (2) See *radiograph*.

X-ray radiation
Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

zirconium

Metallic element with atomic number 40. Zirconium is highly resistant to corrosion, and it is alloyed with aluminum to make cladding for nuclear fuel and sometimes in small amounts with the fuel itself.

ATTACHMENT A
TECHNICAL BASIS FOR THE MINIMUM DETECTABLE DOSE
FOR NEUTRON NTA FILM AT BNL

The minimum detectable dose (MDD) or MDL for NTA film was a function of the total area of the film that were read (Taulbee 2009). The total area is the product of the area of the individual field-of-view (FOV), which varied with the microscope that was used, and the number of FOV read. At BNL, 24 or 25 FOV were routinely read. The difference in read area is only about 4% and has been ignored for this analysis.

To determine the MDD it is necessary to know the distribution of the background tracks and the calibration factor (CF) in mrem/track. Some background NTA data were found for BNL. The data were summarized in the Health Physics Summary report for May 1955 (BNL 1955). Two technicians scanned 1,794 control films. Three groups of 25 FOV were scanned for each film. BNL observed that the data fit a Poisson distribution with a mean of 0.4 tracks. To verify this, this analysis performed a least-squares fit to the data and obtained a value of the mean (and variance) of 0.36. The R^2 value was greater than 0.99. The observed and fitted data are reproduced in Table A-1.

Table A-1. Data from a 1955 background study and the Poisson fit to the data (mean = variance = 0.36).

No. of tracks	Observed		Fitted Poisson		
	Frequency	Proportion	Probability	Expected frequency	Cumulative probability
0	3,686	0.6847	0.69768	3,755.5917	0.6977
1	1,153	0.2142	0.25116	1,352.013	0.9488
2	362	0.0672	0.04521	243.3623	0.9941
3	115	0.0214	0.00543	29.2035	0.9995
4	40	0.0074	0.00049	2.6283	1.000
5	14	0.0026	0.00004	0.1892	
6	8	0.0015	0	0.0114	
7	4	0.0007	0	0.0006	
8	1	0.002	0	0	

At the time of the BNL (1955) analysis, BNL observed that there were excess tracks in the range of 3 to 8 in comparison with the Poisson fit. They attributed this to a low level of exposure to the control films. The observed and expected data from the Poisson fit are plotted in Figure A-1.

The detection limit (L_C, also referred to as the "decision level" or "critical level") is the net number of tracks for reaching a decision that an exposure has been detected. The value is calculated so that there is only a 5% chance that a result above the L_C is part of the background distribution (i.e., a false positive). Taulbee (2009) discussed three methods of calculating a detection limit for NTA film. One of the three was the method that was used in the modern testing standards (NVLAP and DOELAP). However, this method requires more information than is generally available for historical data. The other two methods produced essentially the same result for the BNL data. The classic Currie formula as cited in Taulbee (2009) will be used in this analysis. The formula may be written as:

$$Lc = 2.33 \times \text{standard deviation of the background} = 2.33 \times (0.36)^{0.5} = 1.38 \quad \text{(A-1)}$$

The Poisson distribution can only have integer values and, when films are evaluated, only whole numbers of tracks are counted. BNL counted at least three groups of 25 FOV for calibrations and for significant results, and averaged them (Phillips 1974). Therefore, it is possible that a fractional number of tracks could have been recorded. However, only one fractional calibration factor was found in the site documentation.

ATTACHMENT A
TECHNICAL BASIS FOR THE MINIMUM DETECTABLE DOSE
FOR NEUTRON NTA FILM AT BNL

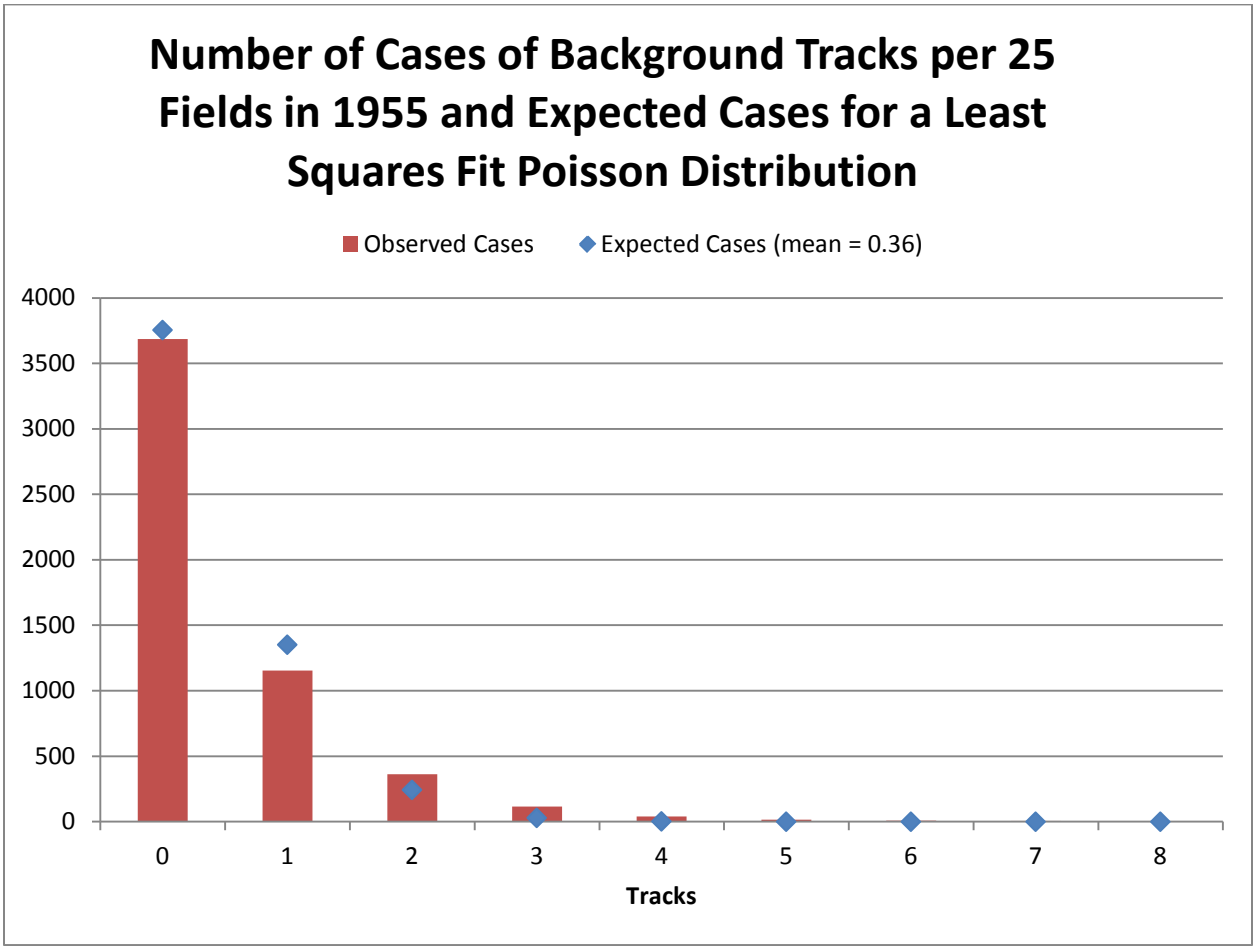

Figure A-1. Plot of observed and expected tracks for background control films.

Equation A-1 indicates that the detection limit should fall between one and two tracks. If one track was selected (i.e., 1.398 is rounded to the nearest whole number), the false positive rate would be greater than 5%, but only slightly. This is seen in the cumulative probability column of Table A-1. If a value of two tracks was selected as the detection limit, the false positive rate would be less than 1%. Given that the background rate was essentially zero, BNL would probably have recorded doses for one or more tracks per 25 fields. The actual procedures BNL used in counting have not been recovered. For purposes of this analysis, a value of 1.4 tracks was assumed.

To interpret the detection levels in terms of dose, it is necessary to select an appropriate dose CF. The CF varies with both the time (month and year) of the calibration and the energy of the neutron spectrum. For this analysis, the calibration values for 1974, after the site had 25 years of experience in evaluating NTA film, were assumed to apply for calculating the MDDs.

BNL, like most sites, calibrated the various emulsion batches with an alpha-neutron source. PoBe was used as the calibration source at BNL until about 1959 when it was replaced by PuBe (BNL 1959). These sources have similar average neutron energies (PoBe ≈ 4.2 MeV; PuBe ≈ 4.5 MeV). Differences were ignored in this analysis. Although the badges were calibrated to an intermediate energy spectrum, in the field the badges were potentially exposed to highly moderated neutrons from

reactors and to high-energy neutrons from accelerators. Very little calibration data by energy is available. However, data were found for 1974 and appear in Table A-2 (Phillips 1974). This data reflects the fact that the lower energy neutrons near the energy threshold of the NTA film are more difficult to detect than those from the calibration spectrum. The difference is a factor of 2. The table also shows the calculated MDDs assuming an L_C of 1.4 tracks. The 1-MeV value (28 mrem) is the most favorable and was rounded up to 30 mrem in Table 6-1 for use in dose reconstruction.

Table A-2. 1974 calibration factors applied to the calculated L_C in tracks.

FOV	Area (mm^2)	CF (mrem/track)	L_C (tracks)	MDD (mrem)	Calibration data from Phillips 1974 (Phillips 1974)
25	0.442	20	1.4	28	1 MeV calibration
25	0.442	10	1.4	14	4.45 MeV calibration (PuBe)
25	0.442	5	1.4	7	17 MeV calibration