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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
Bq	becquerel
cd	calendar day
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
cpm	counts per minute
d	day
DAC	derived air concentration
DOE	U.S. Department of Energy
DORMS	Dosimetry Records Management System
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EU	enriched uranium
g	gram
gal	gallon
GSD	geometric standard deviation
$H_{E,50}$	committed effective dose equivalent
HE	high explosive
HEPA	high-efficiency particulate air
HERS	Historical Exposure Records System
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
L	liter
m	meter
mCi	millicurie
MDA	minimum detectable activity
MeV	megaelectron-volt, 1 million electron-volts
min	minute
ml	milliliter
mrem	millirem
NCRP	National Council on Radiological Protection and Measurements
NDE	nondestructive examination
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH-Office of Compensation Analysis and Support Claims Tracking System

PAEC	potential alpha energy concentration
PAEE	potential alpha energy exposure
pCi	picocurie
POC	probability of causation
RAMS	Radiation Alarm Monitoring System
RST	Radiation Safety Technician
s	second
SNM	special nuclear material
Sv	sievert
TBD	technical basis document
U.S.C.	United States Code
wd	working day
WL	working level
WLM	working level-month
yr	year
µCi	microcurie
µg	microgram
µm	micrometer
§	section, sections

5.1 INTRODUCTION

Technical basis documents and site profile documents are general working documents that provide guidance concerning 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 the National Institute for Occupational Safety and Health (NIOSH) 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) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work.

As noted above, the statute 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 contains an exclusion with respect to the Naval Nuclear Propulsion Program, 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 dosimetry monitoring results are considered valid for use in 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:

- Radiation from naturally occurring radon present in conventional structures
- Radiation from diagnostic X-rays received in the treatment of work-related injuries

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

5.1.1 Purpose

This TBD addresses intakes of radionuclides associated with weapons operations as well as radon exposures, which might have been enhanced due to the unique cell design at the Pantex Plant for limiting the consequences of accidents.

5.1.2 Scope

Activities at the Pantex Plant with the potential for airborne contamination occurred in bays and cells. The principal function in the bays is the assembly and disassembly of nuclear explosives, particularly the mechanical portion of operations that includes electrical components and tritium reservoirs. Physics package assembly and disassembly, which involved bare high explosives (HE) and sealed pits, occurred in the cells. There are 13 cells for assembly and disassembly at the Pantex Plant. Operations with radioactive components began in these cells in 1956. Cell 1 is no longer in use because of an accidental tritium gas release in 1989.

Internal Dosimetry Technical Basis & Quality Assurance Document (BWXT Pantex 2001) implies that particle size measurements could have been made for specific incidents but had not been performed (at that time) for routine airborne contamination conditions. Data were found for particle size measurements assessed for uranium and thorium during the 1990s. It appears that measurements were used to determine appropriate radiation protection measures and not for dose assessment. The dose reconstructor should use the default 5-µm activity median aerodynamic diameter assumption (with the exception of tritium and radon progeny) unless data on specific particle size are available in the records and are representative of the intakes being considered.

5.1.3 History of Internal Dosimetry

There was no routine bioassay program at Pantex before 1972 for uranium, thorium, or plutonium. Bioassay was performed for specific events; for instance, bioassay was obtained from workers involved in a plutonium contamination event in 1961 and from those involved in decontamination of the facility after the event. A 1967 report describing an inspection of the radiation protection program states that Pantex used air samples and contamination surveys to indicate the need for bioassay and did not maintain a routine plutonium or uranium bioassay program (Davis 1967). The report further states that Pantex performed about 10 tritium urinalyses a month, and there was no indication of personnel exposure. There might have been a small routine tritium program, but the research for this analysis found no other information.

The 1991 procedure *Analysis of Biological Samples for Uranium, Thorium, and/or Plutonium* stated that urinalysis was to be conducted for personnel exposed to 40 derived air concentration (DAC)-hr integrated air concentrations as measured by breathing-zone monitors or was to be estimated if not specifically monitored (MHSMC 1991a). The procedure also stated that, “personnel working in potentially contaminated areas shall be entered into the routine bioassay program and shall have a routine bioassay for the suspect heavy metal radionuclide performed every 4 to 6 months.” However, the routine bioassay program for radionuclides other than tritium was short-lived, occurring mostly in 1991 and 1992.

The TBD research did not reveal the level of air concentrations or other workplace indicators that triggered special bioassays before 1991.

In 1989, Pantex contracted with Delphi Groupe to develop the Historical Exposure Records System (HERS), an electronic database that contains the best-available personnel dose data. Original

personnel dose records were reviewed, discrepancies identified and corrected, and data entered in the database. This effort reconstructed and included missing, incomplete, and invalid doses. It included records from 1957 to 1983, with the exception of March 1976 and December 1979 because data were not available for those months. HERS reports are available for several contamination events that occurred in 1989, but earlier data were not easy to extract from the files. The dose records in the worker or claimant files will contain the HERS data, but those data do not include bioassay data. A review of all worker files in the NIOSH Office of Compensation Analysis and Support Claims Tracking System (NOCTS) revealed that only 10% of the files had any bioassay data and most of the data were for tritium results. Only 3% of the files had uranium bioassay results and all but one of the results were for samples collected since 1986. No plutonium or thorium bioassay results were found in the NOCTS files. Therefore, this TBD provides other approaches for determining intakes of radionuclides that dose reconstructors can use when bioassay data are not available.

Table 5-1 provides a historical perspective of bioassay practices at Pantex from 1972 to 2002. The table lists the number of workers monitored for the radionuclides of interest for each year.

Table 5-1. Number of workers with bioassay monitoring.

Year	Workers monitored for tritium	Workers monitored for uranium	Workers monitored for thorium	Workers monitored for plutonium
1972	4	0	0	0
1973	1	0	0	0
1974	0	0	0	0
1975	0	0	0	0
1976	463	0	0	0
1977	466	0	0	0
1978	519	0	0	0
1979	712	0	0	0
1980	14	0	0	0
1981	41	0	0	0
1982	5	0	0	0
1983	0	0	0	0
1984	0	0	0	0
1985	17	0	0	0
1986	626	0	0	0
1987	481	0	0	0
1988	499	0	0	0
1989	212	0	0	0
1990	2,341	46	0	0
1991	1,115	431	0	0
1992	879	239	17	12
1993	1,078	90	0	0
1994	1,104	138	4	3
1995	971	37	90	33
1996	940	69	56	17
1997	933	89	13	18
1998	610	12	1	2
1999	554	13	16	1
2000	535	33	9	8
2001	512	65	16	1
2002	511	57	11	10
2003	441	87	25	9
2004	421	109	15	0

Knowing the job title and a brief description of duties for that title can be helpful in determining the correct information to use for assessing dose. Production Technicians (also called Assembly Operators) or Radiation Safety Technicians (RSTs) typically had the highest potential for intakes of occupational radionuclides. Other workers could have incurred intakes, but the probability of incurring an intake was smaller and the magnitude of an intake, if it occurred, would have been smaller. Claimant interview files might not state the same job title because the interviewee could have described the type of job performed rather than the job title. The job titles have changed over the years. Table 5-2 summarizes job titles, descriptions, and possibilities for intakes.

5.1.4 Current Internal Dosimetry Practices

In the late 1980s and early 1990s, several actions resulted in the current internal dosimetry program. First, there were new regulations from DOE (Order 5480.11, the RadCon Manual, and 10 C.F.R. pt. 835); second, a new contractor came to the site; and third, several workplace incidents occurred that demonstrated the need to improve the internal dosimetry program.

The Pantex Plant radiation protection program uses engineering and administrative controls to prevent intakes. However, because of the quantities of tritium, plutonium, uranium, and thorium handled at the Plant, there is the possibility of an accidental intake resulting in 100-mrem committed effective dose equivalent (CEDE or $H_{E,50}$). According to BWXT Pantex (2001), the purpose of the current internal dosimetry program is to detect intakes equal to or greater than 10 mrem $H_{E,50}$ based on International Commission on Radiological Protection (ICRP) Publication 30 dose calculation methodology (ICRP 1982). To meet the requirements of 10 C.F.R. § 835.402(c), workers who might be likely to have internal $H_{E,50}$ doses higher than 100 mrem participate in the internal dose evaluation program. Pantex maintains routine bioassay monitoring programs for tritium to demonstrate compliance with 10 C.F.R. § 835.402(c). To identify intakes of actinides in a timely manner, the internal dosimetry program is tied closely to the quantification of airborne radionuclide concentrations to which workers are exposed. Pantex does not have a routine bioassay program for actinides, but uses occurrence-based bioassay sampling to confirm intakes and calculate internal doses. Therefore, with the exception of tritium exposure, there is reliance on personal air sampling to determine the need to conduct bioassay sampling. At present, bioassay sampling occurs within 2 to 3 d of an assessment of airborne exposures exceeding 4 DAC-hr for an individual actinide.

5.2 RADIONUCLIDES WITH POTENTIAL FOR INTERNAL DOSE

Only five groups of radioactive materials are of concern for occupational intake at Pantex: tritium, uranium, thorium, plutonium, and radon progeny. BWXT Pantex (2001) discusses the first four radionuclides; the latest version of Section 2 of this Pantex Site Profile (ORAUT 2004a) discusses processes and locations where radioactive material could have been present.

5.2.1 Tritium

The principal sources of tritium at Pantex were and are the weapons components known as *reservoirs*, which first arrived at Pantex in late 1956 or early 1957. A Crockcoft Walton neutron generator in use before 1956 produced some tritium in the off-gas, but the amount would have produced negligible intakes. Tritium sealed under high pressure in the reservoir units has the potential to leak during disassembly. BWXT Pantex (2001) states that tritium could leak through reservoir materials, which presumably refers to concern for migration of molecular tritium through welds. The tritium in the reservoirs is 99% gaseous molecular hydrogen (DT, HT, or T_2) and 1% tritiated water vapor (HTO or T_2O). Tritium gas interacts over time with moisture in the air,

Table 5-2. Job titles and descriptions of work with possibility for occupational intake.

Job title	Description of work	Possibility for intake (1 highest)[a]
Production Technician, Assembler, Assembly Operator, Assembly Fabrication	Assembles, disassembles, reassembles, inspects components	1
Quality Assurance Technician I	Conducts NDE evaluations with linear accelerators, X-ray machines, etc.; conducts telemetry testing; performs confirmatory measurements on components, assemblies, containers, etc.	1
Quality Assurance Technician II	Performs NDE, electronic, destructive, telemetry, and radiation measurement testing	1
RST (entry)	Performs monitoring and sampling; collects samples; assists RST in monitoring personnel	1[a]
RST	Performs monitoring and sampling; collects samples; performs radiation and contamination surveys; conducts surveillance work	1
RST (Senior)	Responds to contamination or radiation alarms; performs surveillance, monitors radiation conditions in workplace	1
Firing Site Technician	Includes hydroshot operators, driver, anyone involved with cleanup of hydroshot contamination	1
Not known, possibly drivers or teamsters	Includes burning of HE and cleanup of ash at burning ground	1
Material Handler (pits and cans)	Operates material handling/moving equipment; transports material; loads and unloads materials and containers	2
Operations Manager, Production Supervisor	Supervises personnel engaged in manufacturing, assembly, packaging, material control, etc.	2
Quality Control Inspectors/ Auditors	Conducts special audits; different from quality assurance technicians	2
Security, protective force, guard	Performs per job title	2[b]
Engineer, engineering	Performs variety of tasks associated with design, testing, procedure development	2[c]
Machinist	Machining on DU for one weapon design only	See Section 5.2.2.4
Metrology laboratory staff	Performs nonradiological metrology calibrations	Environmental only
Fireman	Performs per job title	Environmental only
Computer Programmer, Electronic Data Processing Analyst	Performs computer programming, maintenance	Environmental only
Secretary, Administrator, Technical Writer, nonoperations management, Planner	Performs per job title	Environmental only
Tool and dye maker	Performs per job title	Environmental only
Food service	Performs tasks associated with operation of cafeteria	Environmental only
Stores Stockman, Clerk, Supervisor	Performs tasks associated with general stores	Environmental only

a. Based on actual contact with components or contamination or RSTs assisting potentially contaminated personnel.
b. In general, security personnel had little chance of intakes; however, some small intakes from contamination in cells or igloos are possible. The default assumption is to place security personnel in category 2; however, based on other information in the file, the dose reconstructors can assign environmental intakes only if they believe the Energy Employee's tasks did not involve entry into cells, Gravel Gerties, igloos, or locations with resuspendable contamination.
c. Engineering tasks cover a wide range, and most have no potential for intakes. However, some tasks might have involved observations during assembly or disassembly work or observations during hydroshots. If the engineer did not have a dosimeter or never had recordable dose, assign an environmental dose only unless there is information in the file to indicate otherwise.

hydrogenated materials (e.g., hydrocarbons, organic compounds, and concrete), and some forms of metals to form tritiated compounds and metal tritides.

Tritium gas is far less hazardous than tritiated water, organically bound tritium, or metal tritides, but it combines with water vapor in the air or body tissues to form compounds. Of particular importance is tritiated water, which the human body absorbs. Elemental tritium is not absorbed through the skin to a significant degree. Tritiated water vapor is readily absorbed through the skin and lungs and retained in the body. Tritiated water that enters the body is chemically identical to ordinary water and is distributed throughout the entire mass of body water.

5.2.1.1 Internal Assessment for Tritium During Routine Operations

Notes:

For tritium, *uptake* refers to total tritium distributed in body fluids regardless of mode of intake. Uptake can be thought of as *total intake* and includes skin absorption. Uptake is equivalent to *whole body* in the Integrated Modules for Bioassay Analysis (IMBA) computer program (James 2003), and is the product of the urine concentration in activity per liter multiplied by 42 L of body fluids.

The following discussion makes no distinction between *annual dose* and *committed dose*.

Because the tritium uptakes discussed in this section were determined from Pantex dose calculations, which in turn were determined from urine samples, dose reconstructors should consider uptakes to be considered HTO (inorganic tritium in IMBA) unless otherwise noted in a worker's records.

Pantex analyzed tritium bioassays on the site. For workers assigned to the tritium bioassay program, the frequency of monitoring was monthly. In addition, there were bioassays for new hires and terminations (although it is not clear if this was for all new hires or just those expected to be in the bioassay program). In addition, for each month one-twelfth of the worker population received an annual urinalysis (Alley 1990).

The following tritium discussion deals with four periods: 1956 to 1971, 1972 to 1982, 1983 to 1988, and 1989 to the present. The discussion explains the selection of those periods.

The extent of a routine tritium bioassay program before 1972 is unclear although there are indications of sampling of about 10 workers per month in the 1960s. Because there is no evidence that workers were monitored for tritium before 1972, dose reconstructors should interpret routine occupational records before 1972 that show "0" for internal dose to mean that no information is available rather than to indicate a dose below detectable levels. Dose records in the 1990s specifically state monitored internal emitters (and give a dose) or state not monitored as "N/M". No NOCTS files had tritium bioassay results for years before 1972, and the files included only a few post-1972 tritium results. Tritium doses in the files should be treated the same as those for 1972 to 1982 as described below.

From 1972 to the present, although tritium bioassay occurred, there are few routine monitoring data in individual worker dosimetry records. A few urinalysis records for 1972 show consistent use of 0.25 µCi/L as a detection level and 0.5 µCi/L as the minimum detectable activity (MDA). A batch of urinalysis records for 1983 shows background counts per minute, gross counts per minute, and final concentrations in microcuries per liter. It appears that Pantex recorded nonzero concentrations when

the gross counts per minute exceeded the square root of twice the background counts per minute, which would be a decision level. The smallest nonzero concentration recorded was 0.023 µCi/L, so 0.05 µCi/L would be a reasonable estimate of the MDA at that time. *Technical Basis for the Internal Dosimetry Program and the DOE Pantex Facility* (BWXT Pantex 1992) lists the tritium urinalysis "detection limit" as 14 dpm/ml (0.0063 µCi/L). Although it does not state so directly, this document implies that this is the MDA. BWXT Pantex (2001) lists the tritium urinalysis MDA as 15 dpm/ml (0.0068 µCi/L).

5.2.1.1.1 Dose to Uptake

The most complete set of tritium information consists of maximum and average doses for 1972 to 2001 (Table 5-3). Because it is likely that the dose reconstructor will find only tritium doses rather than actual bioassay results in the worker files, the following paragraphs provide methods to convert from recorded dose to uptake (for input into IMBA or the tritium tool).

Table 5-3. Tritium dose data.

Year	Workers monitored for tritium	Maximum recorded individual tritium dose (mrem)	Maximum uptake in µCi[a]	Average worker tritium dose (mrem)[b]	Average uptake in µCi[a]
1972	4	12	42	8	28
1973	1	0	0	0	0
1974	0	NA[c]	NA	NA	NA
1975	0	NA	NA	NA	NA
1976	463	0	0	0	0
1977	466	0	0	0	0
1978	519	0	0	0	0
1979	712	0	0	0	0
1980	14	114	400	43.8	160
1981	41	122	430	14.2	50
1982	5	37	130	20.2	71
1983	NA	NA	113[d]	NA	0.070[d]
1984	0	NA	NA	NA	NA
1985	17	3	21	0.6	4.3
1986	626	6	43	0.1	0.71
1987	481	2	14	0.02	0.14
1988	499	3	21	0.01	0.071
1989	212	1,180	430[e]	8.5	30
1990	2,341	3	14	0.002	0.010
1991	1,115	5	24	0.02	0.10
1992	879	5	24	0.05	0.24
1993	1,078	14	68	0.2	1.0
1994	1,104	11	53	0.1	0.48
1995	971	12	58	0.1	0.48
1996	940	7	34	0.017	0.082
1997	933	1	5	0.003	0.015
1998	610	1	5	0.0066	0.0066
1999	554	0	0	0	0
2000	535	0	0	0	0
2001	512	0	0	0	0
2002	511	0	0	0	0
2003	441	0	0	0	0
2004	421	0	0	0	0

a. Based on 3.5 µCi/mrem for 1972 to 1982, 7.1 µCi/mrem for 1983 to 88, and based on Equation 5-8b for 1990 to present.
b. Based on Pantex recorded values.
c. NA = not available
d. Obtained directly from urinalysis results.
e. Based on 1981 maximum because of the large single incident in 1989.

Tritium intake prior to 1983. To convert from tritium dose back to uptake for 1972 to 1982, dose reconstructors should use a dose conversion factor of 3.5 µCi/mrem. This conversion results from the approach in ICRP Publication 10 (ICRP 1968) explained in NUREG-0938 (Brodsky 1983; 1.5 mCi = 425 mrem). It assumes an acute intake and a quality factor of 1.7 for tritium beta particles. Dose reconstructors should use a lognormal distribution with a geometric standard deviation (GSD) of 3 and should apply this same conversion to recorded tritium doses for years before 1972 if they encounter any such doses. Because this conversion produces a dose lower than the recorded dose by nearly 44% when input into IMBA, it is permissible to use the doses as recorded for likely noncompensable, maximum internal dose cases.

Tritium intakes 1983-88. Ikenberry (1983) described the uptake to dose calculation method used at that time, which was based on ICRP Publication 2 (ICRP 1959) but used a quality factor of 1.

$$dose\ rate\ in\ rem/d = 8.12 \times 10^{-6}\ q \tag{5-1}$$

where q is the uptake in microcuries. Total dose was determined by integrating over the dose rate curve. For an acute exposure,

$$dose\ in\ rem = 1.4 \times 10^{-4}\ q \tag{5-2}$$

and for chronic exposure,

$$dose\ in\ rem = 8.12 \times 10^{-6}\ qt \tag{5-3}$$

where t is the period of chronic exposure in days.

Equation 5-2 produces a conversion factor of 7.1 µCi/mrem and, assuming a 365-d exposure, Equation 5-3 produces a conversion factor of 0.33 µCi/mrem. Ikenberry (1983) does not specify a particular intake scenario, so it is not known which of the two equations produced the reported doses. Both were probably used to fit whichever intake scenario was appropriate for each worker but, for the purpose of establishing a default intake, Equation 5-2 is claimant-favorable along with the assumption of a lognormal distribution with a GSD of 3.

Tritium intakes 1989 to present. In 1989, DOE Order 5480.11 (DOE 1988) required sites to convert to ICRP 30 internal dose methodology (ICRP 1982). Spot-checking of case files showed that Pantex used both acute and chronic assumptions for different cases. For instance, a May 1991 letter to a worker's file (case 1327) states that a 0.5-ml aliquot was analyzed by liquid scintillation and that "Doses were calculated by the use of computer algorithms incorporating an assumption of a single intake 30 d before the measurement." Two other cases (3754 and 4276), dated 1991 and 1993, respectively, showed outputs from the REMedy© internal dosimetry computer code, and both assumed a chronic intake mode to calculate the dose.

For the acute intake scenario, BWXT Pantex (1992) provided Equation 5-4 to convert from calculated dose to uptake assuming a single exponential retention curve with a 10-d retention half-time:

$$dose = 1.3 \times 10^{-3}\ C_0 \tag{5-4}$$

where C_0 is the initial body water concentration in disintegrations per minute per milliliter and the dose is in millirem. The concentration is distributed in 42,000 ml of body water, so the uptake in disintegrations per minute is $42{,}000C_0$. Therefore,

$$uptake\ in\ dpm = (42{,}000)(dose)/(1.3 \times 10^{-3}) = (3.23 \times 10^{7})(dose\ in\ mrem) \tag{5-5a}$$

$$uptake\ in\ pCi = (1.46 \times 10^{7})(dose\ in\ mrem) \quad (5\text{-}5b)$$

For the chronic intake scenario, the same document provides Equation 5-6 for calculating dose from a urine sample:

$$dose\ in\ mrem = [(8.7 \times 10^{-5}\, t) + 1.3 \times 10^{-3}]\, C_e \quad (5\text{-}6)$$

where C_e is the urine concentration in disintegrations per minute per milliliter. Monthly sampling was the normal frequency for workers potentially exposed to tritium so, with a t of 30 d:

$$dose\ in\ mrem = 3.9 \times 10^{-3}\, C_e \quad (5\text{-}7)$$

Distributing the tritium in 42,000 ml of body water gives:

$$uptake\ in\ dpm = (1.07 \times 10^{7})(dose\ in\ mrem) \quad (5\text{-}8a)$$

$$uptake\ in\ pCi = (4.82 \times 10^{6})(dose\ in\ mrem) \quad (5\text{-}8b)$$

The same equation for an intake period other than 30 d is:

$$uptake\ in\ pCi = (1.89 \times 10^{4})(dose\ in\ mrem)/[(8.7 \times 10^{-5}\, t) + 1.3 \times 10^{-3}] \quad (5\text{-}9)$$

Equation 5-5 differs from Equation 5-8 by a factor of 3, so if the doses are large it could be important to know if the original calculation of the recorded tritium dose assumed the chronic or acute scenario. It is implied, although not explicitly stated, in the Pantex procedure *Internal Dose Assessment* (MHSMC 1991b) that chronic intakes were applied to workers receiving routine monthly bioassay and an acute intake scenario was applied to workers receiving termination or infrequent bioassay. Disassemblies were occurring more often than assemblies during this period, so chronic intakes were more likely.

Equation 5-8 and IMBA produce a slightly smaller dose than originally recorded. For instance, a recorded dose of 10 mrem for a monthly sample results in a recalculated dose of 7.4 mrem. Therefore, if the only information available is the recorded tritium dose, and it is evident from the records that the worker was on a monthly sampling frequency, dose reconstructors can use the recorded dose directly for the likely noncompensable, maximum internal dose approach.

Therefore, for tritium doses in the records for 1989 to the present, the dose reconstructor should apply the following steps:

If the worker's file provides (in order of priority):

- Actual bioassay results and an acute intake date or chronic exposure period, use that information to determine dose; or
- Dose and a chronic intake period, use the recorded dose unless better accuracy is required, in which case use Equation 5-8 for monthly sampling frequency or Equation 5-9 for another exposure period; or
- Dose calculated from a termination sample or single sample when the worker was not on a monthly routine, use Equation 5-5.

Regardless of the step used to determine the dose, dose reconstructors should assume a lognormal distribution with a GSD of 3.

5.2.1.1.2 Tritium Missed Dose Before 1972

See Sections 5.2.1.1.5 and 5.2.1.1.6 for unmonitored worker discussions.

5.2.1.1.3 Tritium Missed Dose, 1972 to 1988

The apparent urinalysis MDA in 1972 of 0.5 µCi/L would have resulted from a chronic intake for 30 d at 1.6×10^6 pCi/d, and the resultant dose to all organs would have been about 3 mrem. If the annual tritium dose is recorded as zero but it appears that some bioassay monitoring occurred, the dose reconstructor should assign the dose as a triangular distribution with a minimum of 0 mrem, a mode of 18 mrem, and a maximum of 36 mrem. These values are small enough that they apply to the entire period; however, if more refinement is desired, the values below for 1983 to 1988 could be more appropriate.

The urinalysis MDA of 0.05 µCi/L applicable in 1983 would have resulted from a chronic intake for 30 d at 1.6×10^5 pCi/d, and the resultant dose would have been far less than 1 mrem. It is unlikely that Pantex tracked doses below 1 mrem, so an assignment of a 1-mrem dose for each month of potential missed dose is reasonable for the period from 1983 to 1989. If the annual tritium dose is recorded as 0 mrem but it appears that bioassay occurred, the calculation should use a triangular distribution with a minimum of 0 mrem, a mode of 6 mrem, and a maximum of 12 mrem.

5.2.1.1.4 Tritium Missed Dose, 1989 to Present

The 1991 internal dose assessment procedure (MHSMC 1991b) lists urinalysis results above which a dose assessment is necessary (Table 5-4). By inference, results below the values in the table did not need dose assessment because, as the procedure states, “The activities cited below have been calculated to result in 1 mrem of exposure based on methods described in ANSI N13.14” (HPS 1983). This TBD analysis has not established how far back in time these screening values were in place, but it is plausible that Pantex started using them in 1989 with the implementation of DOE Order 5480.11 (DOE 1988). Assuming a chronic intake for the monthly sample period and a urinalysis result of 0.135 µCi/L at the end of the period, the IMBA Version 3.1.99 calculates a daily uptake of 4.28×10^5 pCi/d (or 0.87 mrem to all organs). The daily uptake rate and the total potentially missed dose change dependent on the number of monitoring periods, as listed in Table 5-5. However, the potentially missed dose is reasonably close to 1 mrem/30 d, so it is claimant-favorable and efficient to use 1 mrem for each monitoring period. Therefore, if a worker’s record shows, for example, 2 mrem for three monitoring periods in a year and zero dose for the remaining nine periods, the unrecorded dose would be 9 mrem and the recorded dose would be 6 mrem (or could be adjusted using Equation 5-8). This dose would apply equally to all organs. [Using the formula from ANSI Standard N13.14-1983 (HPS 1983), the acute intake in Table 5-4 results in a 1-mrem dose if the time after intake is 7 d, so Pantex must have used the 0.357-µCi/L value for incident follow-up samples. It would not be appropriate for potentially missed dose estimation.)

Table 5-4. Tritium urinalysis screening levels, 1991.[a]

Analysis period	Urine tritium concentration (µCi/L)
Termination	1.35 E-2
Monthly	1.35 E-1
Acute	3.57 E-1

a. From MHSMC (1991b); assumed to apply to 1989 to present.

Table 5-5. Potentially missed intake and dose from monthly sampling for chronic intake of tritium, 1989 to present.[a]

Monthly periods missed	Daily intake ($\times 10^5$ pCi)	Total missed dose (mrem) (all organs)
1	4.28	0.869
2	3.73	1.51
3	3.62	2.23
4	3.58	2.93
5	3.56	3.64
6	3.55	4.37
7	3.54	5.08
8	3.54	5.82
9	3.54	6.55
10	3.54	7.27
11	3.54	7.98
12	3.54	8.73

a. Based on 0.135 µCi/L excretion at end of total period.

If the annual tritium dose is recorded as zero but it appears that bioassay occurred, the dose should be assigned as a triangular distribution with a minimum of 0 mrem, a mode of 6 mrem, and a maximum of 12 mrem.

5.2.1.1.5 Unmonitored Workers, 1956 to 1971

During weapons assembly, there was little chance that tritium could leak because workers did not manipulate the valves on the tritium reservoir. A very small amount of tritium migration through reservoir welds occasionally occurred, which is why workers surveyed the reservoirs on arrival. However, weapons brought in for inspection, repair, or disassembly provided a possibility for a small release of tritium and subsequent intake for Production Technicians (Assembly Operators), RSTs, and Quality Assurance Technicians. Around 1980, disassembly of weapons became more frequent than assembly, and releases were more likely to occur. Table 5-3 shows this pattern and lists the maximum and average recorded tritium doses from 1972 to 2001. Because tritium uptakes before 1972 were unknown but unlikely to be greater than the post-1972 uptakes, this analysis assumed that twice the highest uptake from the 1970s was to apply to all the years from 1957 to 1971 (Table 5-6.) Dose reconstructors should consider the resultant dose a constant upper bound.

5.2.1.1.6 Unmonitored Workers, 1972 to Present

It is unlikely that unmonitored workers had higher intakes than monitored workers. However, for the period from 1972 to 1976 and for 1984, there was little or no tritium bioassay monitoring. During the years in which monitoring occurred, there is no guarantee that everyone exposed to tritium was monitored. Nevertheless, for Production Technicians (Assembly Operators), RSTs, and Quality Assurance Technicians, tritium uptakes were possible. This analysis estimated uptakes for

unmonitored workers in these job categories using Table 5-3 values; the average uptake was not allowed to become less than that which would result in a dose of 6 mrem/yr. Table 5-6 lists those uptakes. Dose reconstructors should consider tritium uptakes for unmonitored Production Technicians (Assembly Operators), RSTs, or Quality Assurance Technicians to be a triangular distribution with the modes and maximums in Table 5-6 and minimums of zero.

Table 5-6. Tritium uptakes for unmonitored workers.

Year	Maximum recorded individual tritium dose (mrem)	Maximum uptake (μCi)[a]	Average worker tritium dose (mrem)[b]	Average uptake (mode of distribution) (μCi)[a]
1956-71	24[c]	84[c]	16[c]	56[c]
1972	12	42	8	28
1973	0	5.8[d]	0	2.9[e]
1974	NM[f]	5.8[d]	NM	2.9[e]
1975	NM	5.8[d]	NM	2.9[e]
1976	0	5.8[d]	0	2.9[e]
1977	0	5.8[d]	0	2.9[e]
1978	0	5.8[d]	0	2.9[e]
1979	0	5.8[d]	0	2.9[e]
1980	114	400	43.8	160
1981	122	430	14.2	50
1982	37	130	20.2	71
1983	NA	113	NA	2.9[e]
1984	NM	5.8[d]	NM	2.9[e]
1985	3	21	0.6	4.3
1986	6	43	0.1	2.9[e]
1987	2	14	0.02	2.9[e]
1988	3	21	0.01	2.9[e]
1989	1,180	430[g]	8.5	30
1990	3	14	0.002	2.9[e]
1991	5	24	0.02	2.9[e]
1992	5	24	0.05	2.9[e]
1993	14	68	0.2	2.9[e]
1994	11	53	0.1	2.9[e]
1995	12	58	0.1	2.9[e]
1996	7	34	0.017	2.9[e]
1997	1	5.8[d]	0.003	2.9[e]
1998	1	5.8[d]	0.0066	2.9[e]
1999	0	5.8[d]	0	2.9[e]
2000	0	5.8[d]	0	2.9[e]
2001	0	5.8[d]	0	2.9[e]
2002	0	5.8[d]	0	2.9[e]
2003	0	5.8[d]	0	2.9[e]
2004	0	5.8[d]	0	2.9[e]

a. Based on 3.5 μCi/mrem for 1972 to 1982, 7.1 μCi/mrem for 1983 to 1988, and Equation 5-8b for 1989 to present.
b. Based on Pantex recorded doses.
c. Assumed values based on twice highest values in 1970s.
d. Based on (0.135 μCi/L)(42 L).
e. Based on (0.5)(0.135 μCi/L)(42 L).
f. NM = not monitored
g. Based on 1981 maximum because of the large single incident in 1989.

There is no reason to expect workers other than Production Technicians (Assembly Operators), RSTs, and Quality Assurance Technicians to incur uptakes of tritium from other than environmental sources.

5.2.1.2 Tritium Release Event in 1989

During a release event in 1989, significant amounts of tritium were released and four workers were exposed to tritium. This event is reasonably well documented, and data are available for assessment. Data on this event are from two primary sources: *Radiation Safety Department Incident Records, Date: 5/17/89, Description: Cell 1 Incident* (Pantex 1989) and a classified document. An individual had an acute tritium exposure at 2:30 p.m. (1430) on May 17, 1989. No alpha contamination was found on the individual. Urine specimens were taken 2 and 4 hr after the incident. According to unclassified information from the classified report, the first bioassay occurred at 1630 using a 0.5-ml aliquot specimen pipetted into 10-ml Biofluor. This sample was analyzed using a Tri-Carb Model 2250 CA liquid scintillation counter. The tritium activity was 291,000 dpm, which was equivalent to 262 µCi/L of tritium.

The individual received medical care for the intake that included many special urine and blood samples and mandatory forced fluids. Urinalysis results will be in the worker's records and are summarized in a letter from the Medical Director (Lang 1990).

The latest version of Section 4 of this Site Profile addresses the environmental release from this incident (ORAUT 2004b).

5.2.2 Uranium

5.2.2.1 Background

Uranium at Pantex was enriched, natural, or depleted. Natural uranium was in a form referred to as tuballoy. Enriched uranium (EU) was in a sealed component with little likelihood of release. No data are available to indicate that EU was ever a contaminant in the workplace. The internal dosimetry technical basis document (BWXT Pantex 1992) stated, "All of the unsealed uranium used at the Pantex facility is either depleted uranium or natural uranium." DU manufactured after 1952 could have contained contaminants from movement of recycled U and DU throughout the Portsmouth, Paducah, K-25, Y-12 complex. Exact levels of contaminants in Pantex DU have not been discovered and probably varied from batch to batch. As an upper bound, dose reconstructors should add the following intakes of contaminants to any intakes of DU: 307 pCi ^{239}Pu/g DU; 3.53 pCi ^{237}Np/g DU; 509 pCi ^{99}Tc/g DU.

According to BWXT Pantex (2001) and interviews with workers, uranium contamination at Pantex is either uranium metal or air-oxidized uranium. Exceptions would be the burning of DU-contaminated HE components at the burn pads and explosion of DU during hydro tests, which would have produced some thermally oxidized DU. BWXT Pantex (2001) states that uranium compounds at Pantex are assumed to exhibit class Y inhalation behavior. However, an earlier internal dosimetry technical basis document (BWXT Pantex 1992) used an assumption of 80% class Y and 20% class W. Neither assumption was based on solubility studies of actual Pantex contamination. Because oxides of uranium can exist over a range of solubility, dose reconstructors should assume either absorption type M or S to maximize the dose to the specific organ of concern. Exposure to significant quantities of type F uranium at Pantex is not credible.

Because components are new during assembly operations, there is little likelihood that significant removable DU oxide would have been on them. During disassembly, aged uranium components from certain weapons programs had a coating of oxide in the form of *black dust* that was potentially present as airborne contamination. Uranium oxide became most noticeable beginning in the early 1980s and was present on eight of the 31 weapons types disassembled at Pantex to date, with types 28 and 55 apparently having the highest contamination from black dust according to worker interviews. Following a contamination event in 1989, consideration for preventing contamination by uranium oxide resulted in modifications to disassembly operations such as the use of downdraft tables.

Some DU was released at the burning grounds from burning of contaminated HE and at the hydro test firing sites when hydro tests involved DU components (Firing Sites 4, 5, and 10 only). In addition, machining of DU-contaminated metal occurred associated with one weapon design.

Because there is no evidence that workers were routinely monitored for uranium before 1991, unless claimant records clearly indicate that uranium bioassay occurred, dose reconstructors should interpret routine occupational records before 1991 that show "0" for internal dose as no information available rather than as a dose below detectable levels. Dose records in the 1990s specifically state whether internal emitters were monitored (and give a dose) or not monitored (N/M). Pantex provided routine urinalysis of uranium in 1991 and 1992. The technical basis document at the time (BWXT Pantex 1992) stated that the uranium urinalysis method was isotopic analysis using alpha spectrometry that "can detect 0.03 pCi/isotope/sample." The document reported an environmental background urinary excretion rate of 0.15 dpm/d of ^{238}U based on studies of potentially exposed and unexposed Pantex workers. This environmental screening level was carried over to the internal dose assessment procedure (MHSMC 1991b), which indicated that dose assessment was to occur for any uranium result with a net activity greater than or equal to 0.15 dpm. BWXT Pantex (1992) stated that the 0.15-dpm screening level would not apply if isotopic ratios implied that the uranium was not derived from DU.

Since 1993, monitoring of uranium exposures has been event-driven and is identified by air-monitoring data. Since the middle 1990s, Pantex has used lapel air samplers to monitor for intakes and trigger bioassay measurements. Because Pantex has performed bioassays on more than 300 workers since 1993, the implication is that there must have been workplace indications of potential uranium intakes. *Analysis of Biological Samples for Uranium, Thorium, and Plutonium* (MHSMC 1991a) provided the following workplace indicators that would trigger bioassay:

- *all personnel … not wearing … respiratory protection whose tracked internal annual exposure is equal to 40 DAC-hours*
- *all personnel whose breathing zone monitor indicates that they have been exposed to 40 DAC-hours* [also lists the DAC for ^{238}U as 6×10^{-11} µCi/ml]
- *all personnel found to have skin contamination equal to or greater than … 1000 dpm/100 cm^2 ^{238}U.*

BWXT Pantex (2001) decreased the trigger value for special bioassay: "Special bioassay samples are collected (usually within 2 to 3 days) when airborne exposures exceed 4 DAC-hours for any single actinide (i.e., >4 DAC-hours for ^{239}Pu, ^{232}Th, or ^{238}U creates an occurrence)." This analysis has not determined exactly when the change occurred between 1991 and 2001.

Table 5-7 lists recorded doses (CEDE) from uranium exposures from 1991 to 2004 as recorded in the facility's dosimetry records management system (DORMS). Although these doses are not directly relevant to dose reconstruction, the overall trend is indicative of reduced uranium intakes after 1993.

The weight percent and activity fractions of radionuclides of DU and even natural uranium can be variable. Values were listed in the 1992 version of the internal dosimetry technical basis document (BWXT Pantex 1992), but their origin was not stated and they were not mentioned in BWXT Pantex (2001). The 1992 values are not significantly different from the default values in the IMBA code (James 2003), and dose reconstructors should use the IMBA values for consistency.

Table 5-7. Uranium dose to workers.

Year	Workers monitored for uranium	Total worker uranium dose (person-mrem)	Maximum individual uranium CEDE (mrem)	Average worker uranium CEDE (mrem)
1990	46	0	0	0
1991	431	109	109	0.25
1992	239	778	502	3.3
1993	90	76	15	0.84
1994	138	0	0	0
1995	37	0	0	0
1996	69	0	0	0
1997	89	0	0	0
1998	12	0	0	0
1999	13	0	0	0
2000	33	0	0	0
2001	65	0	0	0
2002	57	0	0	0
2003	87	10	7	0.11
2004	109	0	0	0

5.2.2.2 Uranium Reporting Levels or Minimum Detectable Activities

For most of its history, Pantex followed an event-driven approach to uranium bioassay and used many laboratories, so the records for bioassay results are spotty. Table 5-8 summarizes some information

Table 5-8. History of uranium urinalysis.

Year	Laboratory	Sensitivity value	Description
1959	Los Alamos Scientific Laboratory	0.5 µg[a]	
1960	Tracer Laboratory	10 µg/L[a]	Fluorometry sensitivity
1963	Controls for Radiation	0.10 µg/L[a]	Less-than value
1965	Controls for Radiation	0.10 µg/L[a]	Less-than value
1967	Controls for Radiation	0.15 µg/L[a]	Less-than value
1968	Isotopes, Inc.	0.10 µg/L[a]	
1978	Control for Environ. Pollution	5 µg/sample[a]	Less-than value
1983	Camp Dresser & McKee	1.4 pCi/L[a]	2σ counting error only; use 3.3 for MDA
1983	Los Alamos Scientific Laboratory	5 µg/L	Less-than value
1990-92	Y-12 Bioassay Lab	0.03 pCi/sample	MDA[b]
1994	Y-12 Bioassay Lab	Approx. 0.15 dpm/ sample, U-238, U-234, U-235. 0.06 dpm/sample U-236	MDAs[a]
2001	Y-12 Bioassay Lab	U-238, U-234, U-235: 0.03 pCi/L	MDA[c]

a. From reports from the laboratories.
b. From BWXT Pantex (1992) and Ealy (1990).
c. From BWXT Pantex (2001).

found in the records. In most cases, the sensitivity parameter in the table was based on observed less-than values in the records. The equations used to determine the less-than values are not known; however, it is likely that the less-than value was more of a decision level than an MDA. Assume the MDA is twice the less-than value.

Table 5-8 has temporal gaps. Because uranium bioassays were generally not obtained routinely but usually from special bioassay samples obtained after events with potential for intake, it is not known if the gaps in the table occur because no bioassay was obtained in those years or if the bioassay sensitivities have not been found. If necessary, dose reconstructors should use the last previous MDA for years not covered in Table 5-8.

Most documentation of uranium exposure at Pantex focuses on DU, but BWXT Pantex (2001) mentions the possibility of exposure to natural uranium. When interpreting bioassay data, if the type of uranium exposure is not known, it is claimant-favorable to assume the intake was natural uranium (BWXT Pantex 2001). However, dose reconstructors should assume DU for intakes associated with hydroshots and burning pads.

5.2.2.3 Uranium Intakes from Assembly/Disassembly of Weapons for Unmonitored Workers

Dose reconstructors should use bioassay data for uranium in worker files to calculate intakes. However, uranium bioassays in the files are scarce. If they were not involved in an incident with potential for intakes, some production line workers might not have had uranium bioassay; therefore, a missed dose calculation for possible frequent intermittent intakes from disassemblies might not be possible. The following discussion provides an approach for assigning intakes for workers associated with assembly and disassembly when no bioassay data are available or when bioassay data are insufficient to determine possible missed dose.

A good set of DU intake data found in the Pantex records is related to a contamination incident in February 1989. Approximately 6 months after identification of the incident, workers were given *in vivo* counts by Helgeson Scientific Services. The data from these counts were subsequently determined to be “flawed with an apparent positive bias” (Brake 1989). Because of this and a recommendation from a DOE investigation team, Pantex management committed to perform urinalysis on all workers.

Note: Dose reconstructors should <u>not</u> use data from the Helgeson *in vivo* counts.

The bioassay samples were taken in late 1989 and early 1990, approximately 1 yr after the contamination incident was identified. *Internal Dosimetry Urinalysis Studies* (BWXT Pantex 2000) contains bioassay data from urinalyses. This is the oldest set of data that provides isotopic determination of uranium alpha activity in urine samples and has sufficient data to perform statistical analysis.

Martin Marietta Energy Systems, Inc., processed the samples at the Y-12 Plant Laboratory. The MDA, 0.03 pCi per isotope per sample, was calculated from the formula in *Performance Criteria for Radiobioassay* (HPS 1996). The calculation included a 1,000-min count, a detector efficiency of 0.0985, and an average recovery of 75%. The average chemical recovery for the sample sets for this incident was 70%. Recoveries less than 25% were considered not accurate due to counting statistics associated with low recoveries. No dose assessment was provided with the data set. Data with a negative value indicated the background was higher than the activity in the sample.

5.2.2.3.1 Analysis of 1990s Uranium Urinalysis Data

From February 1 to April 1, 1990, 305 workers provided urine samples for analysis. The results are documented in the HERS microfilm archive, Roll 9030, identified by badge numbers beginning with "P." In addition, these data contain 23 spiked samples (badge number C999999) and 11 "R999999" samples. This data set was analyzed to establish a baseline for unmonitored workers (or workers whose bioassay results are not found).

The 305 "P" sample measurements, associated with real individuals, averaged 1,635 ±384 g urine and 0.229 ±0.168 dpm total uranium per sample. These measurements are reportedly 24-hr urine samples.

When four individuals for whom radiochemical recovery was listed as zero were removed from the data set, the measurements averaged 1,623 ±385 g urine and 0.227 ±0.163 dpm/d total uranium. Fitting a lognormal distribution to the non-negative values (299 of 301 observations) with two values treated as missing (Strom 1986) gave a median (geometric mean) of 0.188 dpm/d with a GSD of 2. Assuming an acute inhalation intake of absorption class S DU 14 months before the excretion results in a median intake of 75,000 dpm. Although the urine samples were taken in response to a specific event, the measured uranium excretion could have been the result of years of accumulation of DU in the body from smaller, frequent, intermittent intakes. An assumption of 10 yr of chronic intake produces 50-yr committed doses about twice the acute intake doses; however, the chronic intake scenario produces smaller doses until about 7 to 8 yr after the start of intakes. The measured excretion probably resulted from a combination of contributions from chronic intakes since 1980 and the acute intake in 1989. The differences are not large and, during this period when disassembly activity was intense, the chronic intake scenario is the more plausible for most workers.

Using the IMBA internal dosimetry software, the median inhalation intake for absorption type M uranium was determined to be 2.8 dpm/d based on the midpoint (5 yr) of the chronic intake period (that is, 0.188 dpm/d excretion after 1,826 d of chronic intake of type M uranium). The range of intake from 1 yr of chronic intake to 10 yr of chronic intake (2.99 dpm/d to 2.77 dpm/d) was contained within the standard GSD of 3. The median inhalation intake for absorption type S uranium was determined to be 41.5 dpm/d based on the midpoint (5 yr) of the chronic intake period, and the range of intake from 1 yr of chronic intake to 10 yr of chronic intake (81 dpm/d to 34.1 dpm/d) was contained within the standard GSD of 3.

Similar statistical analyses were performed on the ^{238}U results for 1991 through 1994, using coworker analysis methods presented in *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005). The median results and GSDs were 0.041 (2.7), 0.061 (2.7), 0.061 (3.0), and 0.033 (3.0) dpm/d for 1991, 1992, 1993, and 1994, respectively. Assuming ^{238}U represents 83% of the total alpha activity in DU, the total uranium median activities would be 0.049, 0.074, 0.074, and 0.040 dpm/d for 1991 through 1994, respectively. These values are less than the 0.188 dpm/d determined from the 1990 data, and they provide some confidence that the 1990 data are claimant-favorable.

Because ingestion of uranium cannot be ruled out, this TBD analysis estimated chronic ingestion intakes. The median ingestion intake of soluble uranium (f_1 = 0.02) was calculated to be 9.75 dpm/d, and the median ingestion intake of insoluble uranium (f_1 = 0.002) was calculated to be 97.5 dpm/d. Because equilibrium is quickly established for ingestion, the ranges from 1 yr to 10 yr of chronic intake were not significantly different from the median intakes.

5.2.2.3.2 Depleted Uranium Intakes for Assembly/Disassembly Unmonitored Workers, 1980 to 1993

When bioassay data are not available for Production Technicians (Assembly Operators), RSTs, and Quality Assurance Technicians for 1980 to 1993, dose reconstructors should assume a median chronic DU inhalation intake of one of the following: Type M inhalation - 1.3 pCi/d, type S inhalation – 19 pCi/d, soluble ingestion – 4.4 pCi/d, or insoluble ingestion – 44 pCi/d. The distribution should be lognormal with a GSD of 3.

When bioassay data are not available, to assess the intake of a worker whose job had a lower potential for intake but might have had incidental exposure to contamination from disassembly activities (such as category 2 in Table 5-2 or the "some potential" category in ORAUT [2004c]), dose reconstructors should assign 10% of the intake.

5.2.2.3.3 Depleted Uranium Intakes for Assembly/Disassembly Unmonitored Workers, 1994 to the Present

As indicated in Table 5-7, doses to workers were reduced considerably from 1994 to the present due to better procedural controls and barriers. All of the 1995 ^{238}U bioassay results, for instance, were below detection. To account for potential unmonitored dose, dose reconstructors should assign a claimant who worked in a job with a high potential for intake with an intake that is 0.2 times the intake from 1980 to 1993 to account for improved radiation protection barriers and procedures (lognormal, median, GSD = 3) (based on the ratio of urine excretion in the 1994 dataset to the 1989 dataset.)

5.2.2.3.4 Depleted Uranium Intakes for Assembly/Disassembly Unmonitored Workers, 1961 to 1979

The first disassembly of a weapon with DU oxide at Pantex occurred in 1961 (BWXT Pantex 2004, p. 2-10). The number of disassemblies at Pantex is classified. Prior to 1980, approximately 10 disassemblies were performed for maintenance or quality control per year per weapons program at Pantex. Table 5-9 lists the number of weapons retired from use throughout the DOE complex. Weapons can be retired without being disassembled but the *trends* over time should be similar, although what is not shown in or inferable from the table are the numbers of partial disassemblies that resulted in refurbishing of weapons. From 1961 to 1965, most disassemblies were performed at the Medina and Clarksville plants (BWXT Pantex 2004, p 2-19), and from 1966 to 1975, disassemblies were performed at Pantex and the Iowa Army Ammunition Plant. One can infer from the table that the annual number of disassemblies at Pantex for the years before 1980 was probably somewhat less than the number for the period from 1980 to 1989. In addition, before about 1980 nuclear weapon disassembly operations were generally free of contamination. However, because the documentation of the number of disassemblies and partial disassemblies and the contamination levels are not specifically available, it is claimant-favorable to assign unmonitored workers an intake that is the same as that from 1980 to 1993 in accordance with the worker's risk potential from Table 5-2 (lognormal, median, GSD = 3).

A few small sets of uranium urinalysis results were found with analysis dates ranging from February 1963 to March 1967. Although the dates and circumstances for the samples (e.g., routine or post-accident) were not found, the 34 results were analyzed as a group assuming a lognormal distribution and coworker analysis methods presented in *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005). The median excretion rate was 0.375 dpm/d with a GSD of 4.1. Considering that the circumstances of the sampling are not known, this excretion rate compares

reasonably well with the median excretion of 0.188 dpm/d assuming chronic intake for the large group of samples collected in 1990.

5.2.2.4 Depleted Uranium Intakes from Machining

Unlike other sites that performed hydro tests, Pantex did not machine DU as part of the hydro test program. However, machining was performed on DU for a period in the early 1960s associated with one particular weapon design. A 1960 report describing these operations stated that the workers were required to wear respirators during all machining operations and that breathing-zone air sampling was performed during the machining with all samples showing no detectable activity. Judging from the air sample monitoring log at the firing sites at the same time, the air-sampling detection limit at Pantex at that time was about 2 pCi/m^3 (2 × 10^{-12} µCi/ml). This is similar to the detection limit stated for air sampling at the Hanford uranium fabrication facilities in the same time

Table 5-9. Retirements in the DOE complex.[a]

Year	Retirements	Fraction relative to 1980-89 yearly average assuming 100% at Pantex	Fraction relative to 1980-89 yearly average assuming 50% at other plants
1961	1,571	1.25	0.326
1962	766	0.611	0.305
1963	830	0.662	0.331
1964	2,534	2.02	1.01
1965	1,936	1.54	0.772
1966	2,357	1.88	0.940
1967	1,649	1.32	0.658
1968	2,194	1.75	0.875
1969	3,045	2.43	1.21
1970	1,936	1.54	0.772
1971	1,347	1.07	0.537
1972	1,541	1.23	0.614
1973	544	0.434	0.217
1974	807	0.644	0.322
1975	2,240	1.79	0.893
1976	2,181	1.74	NA[b]
1977	998	0.796	NA
1978	1,148	0.915	NA
1979	730	0.582	NA
1980	904	NA	NA
1981	1,887	NA	NA
1982	1,537	NA	NA
1983	749	NA	NA
1984	1,143	NA	NA
1985	1,322	NA	NA
1986	1,224	NA	NA
1987	958	NA	NA
1988	1,023	NA	NA
1989	1,794	NA	NA

a. From DOE (2001)
b. NA = not applicable.

frame (Wilson 1958). Assuming exposure for 8 hr/d just at the detection limit, the daily intake would have been:

working day (wd) intake = (8 hr/d)(1.2-m^3/hr breathing rate)(2 pCi/m^3) = 19 pCi

calendar day (cd) intake = (19 pCi/wd)(250 wd/yr/365 cd/yr) = 13 pCi.

Because the air sample results were below this level and the machinists wore respirators, the 13-pCi/d inhalation intake rate should be considered a constant upper bound. Ingestion of DU from this work cannot be ruled out. Following guidance in *Estimation of Ingestion Intakes* (NIOSH 2004), the estimated ingestion intake would have been:

ingestion daily intake = (0.2)(2 pCi/m^3) = 0.4 pCi/cd (constant).

5.2.2.5 Depleted Uranium Intakes from Burning of Contaminated High Explosives

A 1971 letter from the Plant Manager to the manager of the DOE Albuquerque Operations Office on the burning of HE at the burning grounds stated that, "during the past years, uranium (^{238}U) parts have been burned in conjunction with the waste HE …" (Drummond 1971). Dose from intakes by operators of the burning pad was considered occupational internal dose; exposure of other workers from plumes was considered environmental exposure. During a telephone interview, Herman Phillips, who has worked as a safety engineer at Pantex since 1952, provided details on the burning operations (Bihl and Martin 2004). Burning HE was a part-time task for Transportation Department workers involving a few hours approximately once a week. Only a few workers would have been involved during any one burn. The ignition operators were about 100 yards from the burn pad during a burning. Access for all other workers was restricted to about 300 yards or more. Ash from the burn was collected and placed in 10- to 20-gal cans for burial. Operators wore half-face respirators with high-efficiency particulate air filters during the ash collection task.

The burning grounds have operated since 1952 (DOE 1997). Air sample results from the burning grounds cover 1960 to 1967, with no results for 1963 (MHSMC c. 1967). Two categories of results are listed as *during burning* and *during cleanup*. Some results are recorded as disintegrations per minute per cubic meter and others as counts per minute.

In relation to the air samples taken during burning, 24 were listed as zero or background, 9 had results that ranged from 4 to 112 dpm/m^3, and 17 had nonzero results in counts per minute. (In the logs, the results are written as d/m/m^3.) No information was available for converting the counts-per-minute results to concentrations, but if one assumes a 25% counting efficiency, the nonzero results would represent a range from 2 to 77 dpm/m^3. Therefore, the air sample result of 112 dpm/m^3 is probably the highest concentration measured. Considering that there were 50 total samples, it can reasonably be assumed that the highest result represents the 95th percentile or higher. The exact locations of the samplers were not shown, but reference was frequently made to the pad, implying that the samplers were close to the burning material and not the worker locations.

To estimate the concentration at the location of the workers, this analysis used the following equations from National Council on Radiation Protection and Measurements (NCRP) Report 123 I (NCRP 1996), assuming the air sampler was about 2 m and the workers about 100 m from the burning material:

$$C = fQ/\pi u\sigma_y\sigma_z \qquad (5\text{-}10)$$

where C is the concentration from a ground-level release in activity per cubic meter, f is the fraction of time the wind blows in the direction of interest, Q is the release rate in activity per second, u is the wind speed in meters per second, and the sigma values are the horizontal and vertical atmospheric diffusion parameters.

In this application, the change in concentration from 2 to 100 m for the same burn is being calculated, so $fQ/\pi u$ is constant. The equations then become:

$$C_{(air\ sample)} = K/\sigma_y\sigma_z \text{ at 2 m} \quad (5\text{-}11)$$

$$C_{(workers)} = K/\sigma_y\sigma_z \text{ at 100 m} \quad (5\text{-}12)$$

The diffusion parameters are determined by:

$$\sigma_y = (0.08x)(1+0.0001x)^{-1/2} \quad (5\text{-}13)$$

$$\sigma_z = (0.06x)(1+0.0015x)^{-1/2} \quad (5\text{-}14)$$

where x is the distance from the burning point to the air sampler or workers. When expressed as a ratio of concentrations, the K term cancels, so:

$$C_{(workers)} = C_{(air\ sample)}\ \sigma_{ya}\sigma_{za}/\sigma_{yw}\sigma_{zw} \quad (5\text{-}15)$$

using the subscript w to refer to the worker's location and a to refer to the air sample location.

Using Equations 5-13, -14, and -15 at distances of 2 and 100 m and the 95th-percentile air sample concentration of 112 dpm/m^3, the estimated concentration of DU for workers during burning was 4.8×10^{-2} dpm/m^3. The intake was determined as the product of air concentration, breathing rate, and exposure time. Sampling times of some air sample results were logged; the times were usually from 1 to 2 hr. [ERDA (1976) states that, "Explosives burn rapidly and amounts of 500 lbs or more are usually consumed in less than 15 minutes. However, occasionally (about once a month) smoke can be observed for up to 30 minutes." Nevertheless, worker exposure was assumed to be 2 hr to be consistent with the air sample run times.] Therefore:

95th-percentile intake in pCi
$= (4.8 \times 10^{-2}\ \text{dpm/m}^3)(1.2\ \text{m}^3/\text{hr})(2\ \text{hr})/2.22\ \text{dpm/pCi} = 0.052\ \text{pCi DU per burn}$ (5-16)

According to Mr. Phillips, burns might have been performed about once a week (Bihl and Martin 2004). The air sample log shows far fewer entries than weekly in the years sampled; this analysis could not determine if air samples were taken for every burn. It is reasonable that different workers were involved with different burns. Therefore, the once-a-week estimate seems to be a reasonable upper bound for each worker. Under the assumption that the same workers incurred intakes when the activity referred to in the logs as "pickup" or "cleanup" occurred, the daily intake was:

$$(\text{weekly intake})(50\ \text{weeks/yr})/365\ \text{d/yr} = 7.1 \times 10^{-3}\ \text{pCi/d} \quad (5\text{-}17)$$

Twenty-one air samples were taken during cleanup activities. Only 1 result was listed as background; 6 were nonzero results recorded in disintegrations per minute per cubic meter; and 14 were nonzero results listed in counts per minute. When plotted as a lognormal distribution, the results in disintegrations per minute per meter indicated a median value of about 22 dpm/m^3 (1×10^{-11} µCi/ml) and a 95th percentile of 900 dpm/m^3. The results in counts per minute fell within this distribution with an exception of one at 1,040 dpm/m^3 under a counting efficiency assumption of 25%. No information, such as sampler location, was available on the representativeness of the air sample concentration in relation to the air breathed by the workers, but the large GSD is adequate to cover uncertainties in the representativeness of the samples. Mr. Phillips stated that the workers wore half-face respirators during cleanup activities (Bihl and Martin 2004), but credit was not taken for this use of respirators. On the few occasions that the time of sampling was logged, the times ranged from 40 to 90 min. A 2-hr exposure time for cleanup of the ash was assumed. There was no definitive information about the

frequency of cleanup activities; the air sample logs show cleanup at a less frequent rate than burning. This analysis made the claimant-favorable assumption that cleanup occurred once a week. Therefore, the 95th-percentile default intake from cleanup of DU-contaminated ash is:

95th percentile intake in pCi/d
= (900 dpm/m^3)(1.2 m^3/hr)(2 hr)(50/365)/2.22 dpm/pCi = 130 pCi/d (5-18)

Assuming the same workers performed the cleanup as those who performed the burn, the intake from the burn itself is negligible in relation to the cleanup activities. Therefore, the default intake rate of DU for burning ground workers is 130 pCi/d for 1952 to the present. This intake would represent the upper bound so should be considered a constant distribution. Dose reconstructors should assume either absorption type M or S.

5.2.2.6 Intakes from Hydroshots

5.2.2.6.1 DU

Pantex has used firing sites for HE quality control and research since 1952. Some of the test firings at Firing Sites 4, 5, and 10 involved DU through 1985 (DOE 1997; MHSMC 1990, Chapter 7). It is not known when hydroshots first involved DU; the Environmental Assessment implies, but does not directly state, that DU hydroshots started in 1967 (ERDA 1976); however, because firm documentation on the exact data was not found, exposure was assumed to begin in 1952. Dose from intakes by operators of the hydroshots was considered occupational internal dose; exposure of other workers from plumes was considered environmental exposure. Mr. Phillips described the hydroshot operations (Bihl and Martin 2004). Operators were in a bunker a few tens of feet from the detonation site. The bunker was fully enclosed with a ventilation system that was shut down during the detonation. After the detonation, the operators walked to ground zero to retrieve their instruments. A driver, who was outside the fenced area (approximately 2,000 ft from ground zero), drove to the detonation site to retrieve the operators. The total exposure time for the operators was less than 30 min. The cloud from the detonations was clearly visible and the operators and driver avoided direct exposure to the cloud.

The prevailing winds carried the cloud basically northward away from most Pantex buildings and worker locations. The water treatment plant and sewage treatment plants are about 1 and 2 km, respectively, northeast of Firing Site 5; they are a bit closer but nearly due east from Firing Site 4.

This analysis found data providing air concentrations inside and outside the bunker at Firing Site 4 for October 1959 to January 1962 (MHSMC c. 1967). The data list 94 results for inside the bunker and 79 results for outside. Eighty-five percent of the results are recorded as 0 dpm/m^3 with the lowest nonzero value recorded as 1 dpm/m^3 (only one significant figure was recorded). Figure 5-1 shows a log-probability plot of the inside-bunker data, and Figure 5-2 shows that for the outside-bunker data with the air concentrations expressed in picocuries per cubic meter. The inside concentrations fit a log-probability curve with a median concentration of 0.18 pCi/m^3 and a 95th-percentile value of 6.2 pCi/m^3. The outside concentrations had a median concentration of 0.82 pCi/m^3 and a 95th-percentile value of 24 pCi/m^3. The following conditions could have caused the large spread in the air concentrations:

- Different hydroshots could have contained different masses of DU or HE.

- The cloud could have been sharply defined close to the detonation spot and shortly after the detonation time, such that under rare weather conditions the samplers were within the boundaries of the initial cloud, whereas most of the time they were not.

- A sample filter could have occasionally picked up a large particle of DU.

Pantex obtained air concentrations from hydroshots in the 1970s using a radio-controlled drone mounted with air samplers on each wing. The drone flew through the cloud within a minute or two of the test at a location expected to sample the highest concentration. These results were used to predict offsite concentrations rather than exposure of the test operators. Mr. Phillips expressed an intuitive estimate that ground-level concentrations for the operators would have been at least 100 times lower (Bihl and Martin 2004).

This analysis found results from the drone samplers for March 1971, various dates in 1972, and two dates in the summer of 1974 (Gidley 1971; Alexander 1972a,b,c, 1974). It is not known if these data

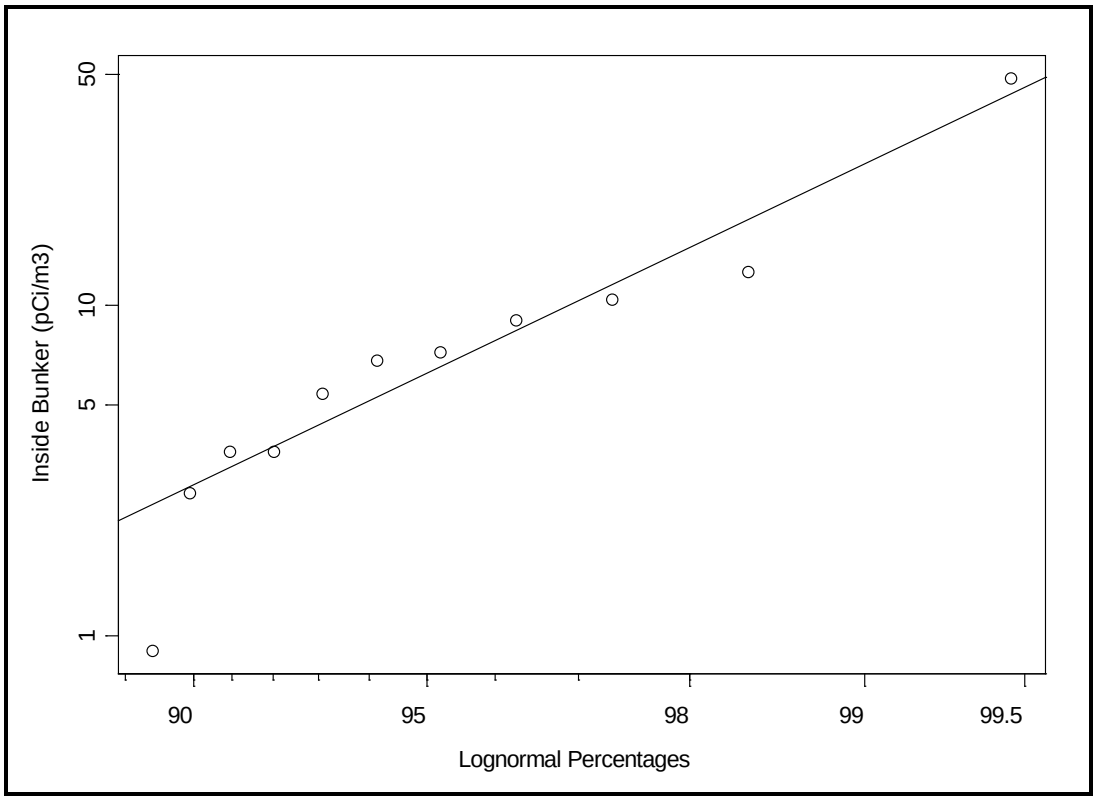

Figure 5-1. Air sample log-probability fit inside hydroshot bunkers.

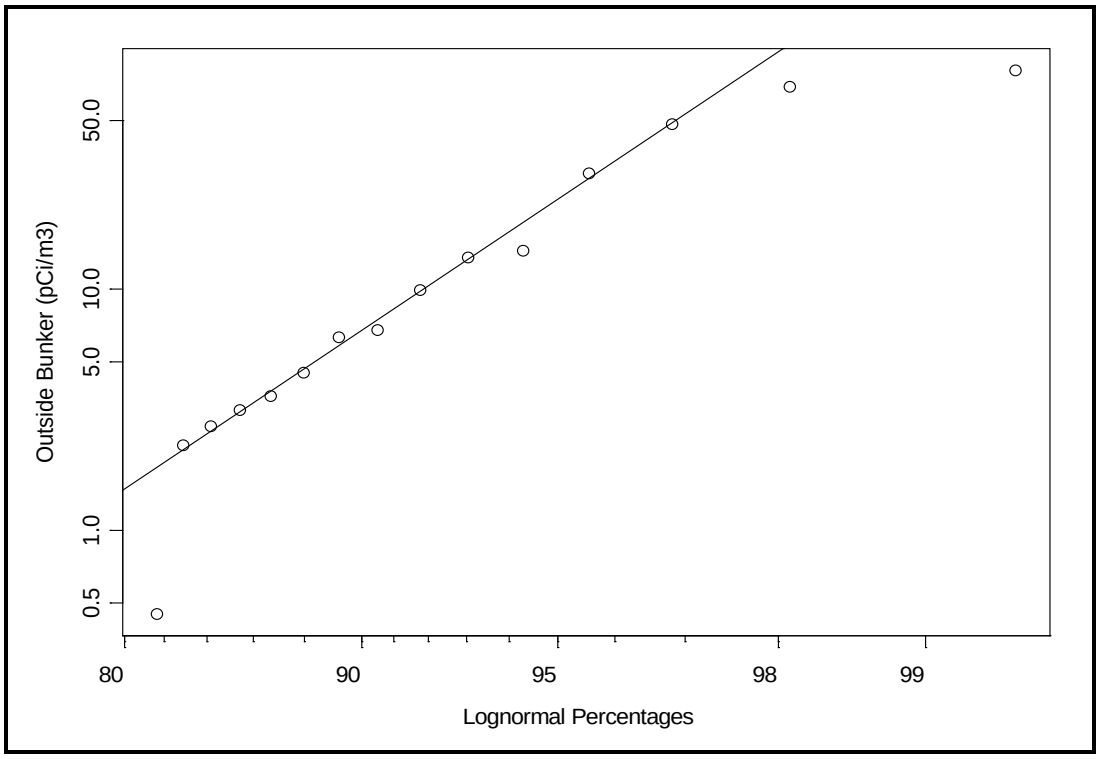

Figure 5-2. Air sample log-probability fit outside hydroshot bunkers.

comprise the complete set of data from drone flights or merely what has been found to date. The design of the firing pad at Firing Site 5 changed from a flat pad to a sand-filled pit in the summer of 1971, but the drone results do not show an appreciable change in cloud concentration before and after the change. The log-probability fit to the drone sampler results had a median concentration of 70 pCi/m^3 and a 95th-percentile concentration of 944 pCi/m^3, so the median ground concentration outside the bunker in the 1960s was about 85 times less than the median cloud concentration measured in the 1970s. Considering the large GSDs and the dynamics of cloud concentrations measured very near to the cloud origin, the two sets of measurements are reasonably consistent; that is, they do not indicate that the 1970s shots produced significantly higher or lower ground-level concentrations than the 1960s shots.

The March 1971 drone results were from a study that compared the cloud concentration of DU from a hydroshot that contained DU and two detonations from Firing Site 5 that contained only HE (Gidley 1971). The cloud concentration of DU was similar for all three shots, and the DU from the HE-only shots was resuspended contamination from the firing site area. The effect of the sandy pit on resuspension of DU is not known, but by 1977 there was a policy to not use Firing Site 5 for any non-DU detonations because, “this could cause the resuspension of some of the residue of shots fired at that bunker in the past” (Boettner 1977; Wilcox 1984; Alexander 1984). Therefore, intakes of DU by firing site personnel were possible after every shot at Firing Site 5, not just the hydroshots.

Table 5-10 lists what was discovered about the number of DU-related detonations. The number of shots in 1960 and 1961 was assumed to equal the number of air samples in the air sample log (often more than one sample per day), namely 30 and 58, respectively. These shots occurred at Firing Site 4. The shots from 1971 to 1986 occurred at Firing Site 5. Test firing using DU ended in 1986 (MHSMC 1990, Chapter 7). As shown, there are gaps in the data. This analysis used annual environmental reports to fill some of the gaps (Alexander 1975, 1976; Alexander, Cornelius, and Horton 1978; Laseter 1986).

Table 5-10. Recorded numbers of hydro test shots and other detonations at Firing Site 5 or other sites.

Year	Hydroshots	Other shots at FS-5	Total
1967	67[a]	ND[b]	ND
1968	3[a]	ND	ND
1969	37[a]	ND	ND
1970	54[a]	ND	ND
1971	49c	ND	ND
1972	45c	ND	ND
1973	28c	52[c]	80
1974	ND	ND	ND
1975	ND	ND	ND
1976	7[c]	17[c]	24
1977	ND	ND	ND
1978	ND	ND	ND
1979	3[c]	8[c]	11
1980	5[c]	0	5
1981–86	ND	ND	ND

a. From ERDA (1976). Specific site was not identified.
b. Not discovered.
c. From Johnson (1973, 1974, 1980, 1981, 1982) and Alexander (1977).

Some of the annual environmental reports listed release activities of DU based on estimates of the amount of DU that went airborne from the firing site. Table 5-11 summarizes that data. The ratio of 1973 release activities over those from 1976 was 5, whereas the ratio of the number of hydroshots was 4 and the ratio of total shots at Firing Site 5 was 3.3. Therefore, it is claimant-favorable to use total shots in 1973 as the base and the ratio of release activities to estimate total shots for years where the numbers of shots are missing. For instance, the estimated number of shots at Firing Site 5 for 1977 becomes (0.001)(24)/0.00004 = 600. However, a limit was set at 500 shots per year based on a reasonable limit of two shots per working day to which any one worker could have been exposed. Table 5-11 lists the recorded or estimated total number of shots potentially involving dispersion of DU.

The default intake for firing site operators and drivers, assuming a 30-min exposure to each shot, would be:

daily chronic intake in pCi
= (*air concentration pCi DU/m*^{3})(1.2-m^3/hr breathing rate)(0.5 hr)(*no. of shots/yr*)/(365 d/yr) (5-19)

Because of the uncertainty as to whether an operator was or was not in the cloud for any particular shot and because the operators and driver spent part of their time outside of the bunker retrieving instruments, the 95th percentile outside air concentration was used to determine the intake per shot (24 pCi/m^3). Substituting this air concentration and simplifying the math, the formula for the default intake becomes:

$$\textit{daily chronic intake in pCi} = (3.9 \times 10^{-2})(\textit{no. of shots/yr}) \quad (5\text{-}20)$$

Table 5-11. Estimated number of detonations potentially involving DU dispersion and intakes.

Period	Total shots	Intake (pCi/d)
1952-1959	500/yr[a]	20
1960	30	1.2
1961	58	2.3

1962-72	500/yr[a]	20
1973	80	3.2
1974	60	2.4
1975	120	4.7
1976	24	0.95
1977	500[a]	20
1978	500[a]	20
1979	11	0.43
1980	5	0.20
1981	6	0.24
1982	6[b]	0.24
1983	6[b]	0.24
1984	0	0
1985	6[b]	0.24
1986	6	0.24

a. Assumed 2 shots per working day.
b. Assumed equal to highest year in 1980s.

Because the 95th percentile was used, which certainly would not apply to all shots, dose reconstructors should consider the result to be a constant upper bound and assume either absorption type M or S.

5.2.2.6.2 Other Possible Radioactive Sources

The Pantex Plant Radiological Investigation Report (BWXT Pantex 2004) states that contaminated soil from the firing sites potentially contained low levels of beryllium, DU, strontium, and thorium. The more detailed discussion of the firing sites does not mention use or residual contamination of radioactive strontium, and when questioned directly if there had been a hydroshot involving radioactive strontium, Mr. Phillips said there had not been. Mr. Phillips said there had been one hydroshot involving thorium (Traub 2006). [The year of this event is still being investigated, and an update will be made when the year is determined. Until then, dose reconstructors should assign the intake below, once, to anyone involved with hydroshots.] Assuming that the air concentration from the thorium hydroshot was no worse than the 95th-percentile concentration outside the bunker from the DU shots (24 pCi/m^3), and assuming a 30-min exposure, the one-time acute intake is:

$$\begin{aligned} &\textit{acute intake in pCi} \\ &\quad = (\textit{air concentration pCi Th/m}^3)(1.2\text{-m}^3\text{/hr breathing rate})(0.5\text{ hr}) \qquad (5\text{-}21) \\ &\qquad\qquad \text{acute intake in pCi} = 14\text{ pCi.} \end{aligned}$$

As discussed in Section 5.2.3, the state of equilibrium of ^{232}Th with its progeny is not known. It is claimant-favorable to assume that the 14 pCi is for ^{232}Th and the progeny are in equilibrium. Either absorption type M or S may be assumed.

5.2.2.6.3 Clean Up of Firing Site 5

Firing Site 5 was decontaminated in the late 1990s. According to the Radiation Safety Section Manager, the work was done by a subcontractor under Radiation Work Permits with monitoring by site Radiological Control Technicians and bioassays were collected. This work was conducted during the time Pantex was using lapel air samplers and using 40 DAC-hr as the trigger for initiating bioassay. Therefore, if it is determined that an energy employee was an unmonitored worker associated with the Firing Site 5 cleanup, an assumption of chronic intake of 50 DAC-hr per year of DU can be made. [This assumes that intakes below 1 DAC-hr on any given air sampler are disregarded and there might

be 50 such results per year.] Fifty DAC-hr of type S uranium equates to an intake of 600 pCi, or 1.6 pCi/d.

5.2.2.6.4 Clean Up of Firing Site 23

In the 1980s a device was constructed at Firing Site 23 for destruction of small, prototype-sized weapons with HE and DU that had been tested off the site and returned to Pantex. The device, referred to as the *Silver Bullet*, was a contained system with air effluents vented through a high-efficiency particulate air (HEPA)-filtered system. Within the Silver Bullet, a device was surrounded with lead plates and bags of gravel. The Silver Bullet was used for one weapons design only and only about 10 shots were conducted. After the explosions, the contaminated gravel was removed by shoveling onto a conveyor belt that dumped the gravel into a wooden box for disposal. Workers wore full anti-contamination clothing and respirators during both setup of the shot and cleanup afterwards. Cleanup of the gravel was conducted after each shot.

Urinalysis data associated with one Firing Site 23 cleanup was found (Copeland 1983a,b). The exact period of the cleanup activities is not known; a batch of urine samples dated October 4, 1983 are labeled as background samples; the other sample dates range from November 9 to December 5, 1983. The November and December DU urinalysis results were analyzed as a group using the coworker analysis methods in *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005). The results ranged from <5 to 50 µg/L, with all but one exceeding the analysis detection level. The median urinalysis result was 16 µg/L with a GSD of 2.1.

The intake scenario was difficult to determine based on the pattern of the bioassay. Tracy Ikenberry, who was a radiological engineer at Pantex at the time, believed that actual removal of the contaminated gravel would have been a 1- or 2-d task at the most for a small crew of workers, although he did not witness any of the cleanups. Several of the workers had samples taken on November 9, 14, and 21, and the latest samples usually had the higher concentrations. Therefore, two intake scenarios were investigated: 1) The median result was modeled as a sample obtained four days after the start of two days of exposure, and 2) the median result was modeled as a sample obtained at the end of 12 d of chronic exposure (November 10-21). The first assumption produced the largest total intake (1.6×10^5 pCi/d DU for 2 d), which was about 3 times greater than the intake from the second assumption. Therefore, the uncertainty from the intake assumption was greater than the uncertainty in the distribution of the urinalysis results; and a GSD of 3 is reasonable.

If urinalysis results are not available after subsequent shots and clean-up activities, the following default intakes can be assigned. Personal recollections of persons associated with use of the Silver Bullet indicate that there were 10 shots with the last shot in 1988. Because these were rare, short-term events spread over many years, an assumption of acute intake is best. Assuming the first shot occurred in October or November 1983, the number of shots with cleanups can be distributed as one in 1983, two in each year from 1984 to 1987, and one in 1988. They should be modeled as acute intakes of 3.2×10^5 pCi of type S DU (GSD of 3) with intakes dates of November 9, 1983, January 2, 1984, July 1, 1984, January 2, 1985, July 1, 1985, January 2, 1986, July 1, 1986, January 2, 1987, July 1, 1987, and January 2, 1988. These intakes should be applied to workers identified as being involved with hydroshots or firing sites. The workers involved in the 1983 cleanup are identified in the records, and it is claimant-favorable to assume they were involved in subsequent cleanups if consistent with their employment history.

5.2.3 Thorium

Thorium at Pantex exists as thorium metal, thorium alloys, or materials impregnated with a thorium compound. Workers handle these forms during assembly and disassembly of certain weapons. Because of the relative hazard of thorium, Pantex uses strict workplace monitoring practices, such as smear checks of components, to verify the integrity of the thorium encapsulation. It is assumed that workers could have encountered oxidized thorium components during disassembly of weapons. Pantex has never conducted machining of components containing thorium.

Information on source terms of weapons containing thorium is classified, as is the number or percentage of weapons that contain thorium. However, there is strong indication that controls for contamination have always been in place, as has workplace monitoring for thorium.

Natural sources of thorium can exist in rocks and soils (see the latest version of Pantex TBD on occupational environmental dose, ORAUT 2004b). Thorium can be present in measurable amounts in biological materials in the environment; ingestion of these materials can result in measurable quantities of thorium in bioassay samples collected from workers. Baseline bioassay measurements have shown this to be true. Pantex determined the amount of thorium that is naturally present in baseline bioassay samples for its workers.

Thorium-232 and ^{228}Th levels were analyzed in baseline fecal samples. Environmental levels of ^{232}Th and ^{228}Th were determined using a lognormal probability analysis on the bioassay data in accordance with *A Guide for Environmental Radiological Surveillance at U.S. Department of Energy Installations* (Corley et al. 1981). The 95th-percentile results for background levels of thorium in fecal samples were 0.4 dpm/sample (0.18 pCi/sample) of ^{232}Th and 0.39 for the ^{232}Th:^{228}Th ratio. Results that exceed both screening levels are assumed to represent occupational exposure and are adjusted to a net occupational excretion by subtracting the arithmetic mean background excretion rate of 6.7×10^{-2} dpm/sample (3.0×10^{-2} pCi/sample) from ^{232}Th fecal results.

BWXT Pantex (2001) states that thorium at the plant is inhalation class Y, which would be essentially equivalent to absorption type S. This is consistent with the ICRP Publication 68 recommendation that thorium oxides are type S (ICRP 1995). Although processing of thorium at Y-12 and Hanford created disequilibrium between ^{232}Th and ^{228}Th, material handled at Pantex would have aged long enough that a significant amount of ^{228}Th would have grown back, especially for weapons being disassembled. The dose reconstructor should assume equilibrium without evidence to the contrary.

Because there is no evidence that workers were routinely monitored for thorium before 1991, unless the worker records clearly indicate bioassay for thorium, dose reconstructors should interpret routine occupational records before 1991 that show "0" for internal dose as no information available rather than necessarily as a dose below detectable levels. Dose records in the 1990s specifically state whether internal emitters were monitored (and a dose is given) or not monitored (N/M). Monitoring of thorium exposures has been event-driven since at least 1991. Table 5-12 lists the number of individuals monitored and the dose results. The procedure *Analysis of Biological Samples for Uranium, Thorium and/or Plutonium* provided criteria for when thorium bioassay monitoring was required (MHSMC 1991a). To summarize, the criteria were exposure to 40 DAC-hr of thorium in the workplace air after accounting for use of respiratory protection, if applicable, or skin contamination equal to or exceeding 200 dpm/100 cm^2. The only reported doses have occurred since 1999. This analysis found no bioassay data before 1983.

Table 5-12. Thorium dose (CEDE) to workers.

Year	Number monitored	Total worker thorium	Maximum individual	Average worker

	for thorium	dose (person-rem)	thorium dose (mrem)	thorium dose (mrem)
1991	0	0	0	0
1992	17	0	0	0
1993	0	0	0	0
1994	0	0	0	0
1995	67	0	0	0
1996	56	0	0	0
1997	13	0	0	0
1998	1	0	0	0
1999	16	25	25	1.6
2000	9	14	14	1.6
2001	16	221	150	13.8
2002	11	48	48	4.4

In early 1992, several workers were given baseline thorium urine bioassays. The analyses were performed by Controls for Environmental Pollution, Inc. The results were less than 0.05 dpm/L. Therefore, dose reconstructors should assume that the MDA was about 0.1 dpm/L (0.045 pCi/L). All the follow-up results were less than the detection limit.

It is not certain that NOCTS records will include bioassay results even for the monitored workers listed in Table 5-12. Considering the extent of workplace controls used during disassembly of thorium

weapons, the very limited amount of available data in relation to thorium intakes, and the inability to determine the frequency of handling weapons containing thorium, dose reconstructors should:

- For workers who had the highest possibility of intake (from Table 5-2), for each year from 1980 to 2000, assume a single acute intake of 40 DAC-hr (48 pCi) of ^{232}Th (in equilibrium with progeny) [$(1 \times 10^{-12}$ µCi/ml$)(10^6$ ml/m$^3)(1.2$ m^3/hr$)(40$ hr$)(10^6$ pCi/µCi$)$]. Assign 0.1 times this intake for category 2 workers in Table 5-2. These intakes are modes of triangular distributions with a minimum of zero and a maximum of 10 times the mode to account for the possibility of more than one intake per year and possible unrepresentativeness of the air-sampling system. There is no record of disassembly of thorium weapons before 1980.

- For each year from 2001 to the present, for workers with the highest possibility of intake, assume a single acute intake of 4 DAC-hr (4.8 pCi) unless the recorded dose for the worker exceeds 5 mrem CEDE. If the recorded dose exceeds 5 mrem, convert the recorded dose to intake using 0.86 pCi/mrem [$(3.2 \times 10^3$ Bq/Sv from ICRP 1989$)(10^{-5}$ Sv/mrem$)(27$ pCi/Bq$)$]. Assigning 0.1 times this intake for category 2 workers in Table 5-2. Because Pantex has used lapel sampling during this period, these intakes are the maximum of a triangular distribution with a minimum of zero and a mode of one-half the maximum.

A check on the reasonableness of the above estimates was made by analyzing ^{232}Th bioassay results. Two hundred fifty-eight worker urine samples were analyzed between 1992 and 1996. Only one result arguably exceeded the detection level; the median of the distribution was 0.000 pCi/L and the 95th percentile was 0.004 pCi/L (less than detectable). One hundred fifty-one worker fecal samples were analyzed between 1996 and 2000. About half were above the analytical detection level, but only four exceeded the expected natural excretion of about 0.32 pCi/d (2.9 µg/d × 1.1×10^{-7} µCi/µg ^{232}Th $\times 10^6$ pCi/µCi [ICRP 1975]. An acute intake of 48 pCi would result in less than 0.32 pCi/d excretion over about 6 d after the intake, so the intake estimate above and the fecal data agree reasonably well.

5.2.4 Plutonium and Plutonium Incidents

5.2.4.1 Plutonium General

Plutonium at Pantex is in the encapsulated pits of nuclear weapons. Workers handle the pits during weapons assembly and disassembly. Strict workplace monitoring practices ensure the integrity of the encapsulation including contamination smear checks during assembly and disassembly. If contamination occurred, exposure to plutonium would be acute rather than chronic. Table 5-13 lists the number of workers given plutonium bioassay by year. There were no recorded internal doses associated with these 1991 to 2002 bioassays.

Table 5-13. Number of workers on plutonium bioassay, 1991 to 2002.

Year	Number of workers monitored for plutonium
1991	0
1992	12
1993	0
1994	0
1995	28
1996	17
1997	18
1998	2
1999	1
2000	8
2001	1
2002	10
2003	9
2004	0

Although exposure to plutonium has been strictly controlled at Pantex, there is indication of past concern about potential plutonium intakes. Based on two sets of bioassay data, it appears that in 1963 and 1966 monthly urine sample collection and analysis occurred for workers who could have been exposed to plutonium. This analysis found additional bioassay data for 1961, 1968, the early 1980s, and 1994 to present. A 1961 incident that resulted in the release of plutonium is discussed below.

Dose estimates from these bioassays were not found and, because they are not listed in the records, doses were probably determined to be less than whatever recording level was used at that time. For 1961 data, the results were not detectable and the lower limit of detection was 0.2 cpm/sample or about 0.8 dpm/sample. Although not specified in these data, other bioassay data indicated that 24-hr samples were taken. The results for the 1968 samples were all less than 0.3 dpm/L. MDAs were not provided for 1963 and 1966 or for data from the 1980s. Bioassay data from March 30, 1994, had isotopic MDAs ranging from 0.01 to 0.04 dpm/sample for ^{238}Pu and $^{239/240}$Pu. A number of different vendors analyzed plutonium bioassay samples over the years.

Several individuals who had urine bioassays for plutonium in the 1960s were identified as claimants, and their claimant files were reviewed. The analysis found no indication of the bioassay history or doses from these bioassays in the records. Therefore, claimant files probably will not contain reliable bioassay data. If bioassay data are not available, dose reconstructors should use the following approach for assigning potentially missed doses to workers.

Because the plutonium was encapsulated, it was assumed that the potential for intake was rare (i.e., intakes would have been acute rather than chronic). BWXT Pantex (2001) states that plutonium at the Plant should be considered an aged weapons-grade mixture. For the following discussion, the intake activities are for the total alpha activity of the mixture. Dose reconstructors should assume the 20-yr aged mixture. Table 5-14 lists the composition of weapons-grade plutonium mixtures. Because the source of intake would have been plutonium oxides, dose reconstructors should assume inhalation type S.

In addition, dose reconstructors should use the following methods based on exposure period:

- For workers who had the highest possibility of intake (from Table 5-2), for each year of possible exposure from 1980 to 2000, when the number of disassemblies was highest, assume a single acute intake of 40 DAC-hr (290 pCi) [$(6 \times 10^{-12}$ µCi/ml)$(10^6$ ml/m^3)(1.2 m^3/hr)(40 hr)$(10^6$ pCi/µCi)]. Assign 0.1 times this intake for category 2 workers in Table 5-2. These intakes are modes of triangular distributions with a minimum of zero and a maximum of 10 times the mode to account for the possibility of more than one intake per year and possible unrepresentativeness of the air-sampling system.

Table 5-14. Activity composition of weapons-grade plutonium mixtures from Hanford.[a]

Mixture designation:	**Fresh**	**5-yr**	**10-yr**	**15-yr**	**20-yr**	**25-yr**	**30-yr**
Years of aging[b]:	**0**	**5**	**10**	**15**	**20**	**25**	**30**
Specific activity in mixture (Ci/g)							
Pu-238	8.56E-03	8.23E-03	7.91E-03	7.60E-03	7.31E-03	7.03E-03	6.75E-03
Pu-239	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02
Pu-240	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02
Pu-241	8.24E-01	6.48E-01	5.09E-01	4.00E-01	3.15E-01	2.48E-01	1.95E-01
Pu-242	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06
Am-241	0	5.83E-03	1.04E-02	1.39E-02	1.66E-02	1.87E-02	2.03E-02
Pu-239+240	7.13E-02	7.13E-02	7.13E-02	7.13E-02	7.12E-02	7.12E-02	7.12E-02
Pu-alpha	7.99E-02	7.95E-02	7.92E-02	7.89E-02	7.85E-02	7.83E-02	7.80E-02
Total alpha	7.99E-02	8.53E-02	8.96E-02	9.28E-02	9.52E-02	9.70E-02	9.83E-02
Activity ratios							
Pu-239+240: total alpha	1.00	0.836	0.796	0.768	0.749	0.735	0.725
Pu-238: total alpha	0.107	0.0965	0.0883	0.0819	0.0768	0.0725	0.0687
Pu-241: total alpha	10.3	7.60	5.68	4.31	3.31	2.56	1.98
Am-241: total	0	0.0684	0.116	0.150	0.174	0.193	0.207

a. BWXT Pantex (2001) did not provide a table of isotopic mixtures. The Hanford mixtures should be close enough for the default assumptions. The total alpha specific activity changes only about 10% from 10 to 30 yr of aging.
b. Time since separation of ^{241}Am from the Pu mix.

- For each year from 2001 to the present, for workers with the highest possibility of intake, assume a single acute intake of 4 DAC-hr (29 pCi). Assign 0.1 times this intake for category 2 workers in Table 5-2. Because Pantex used lapel sampling results as criteria for when to conduct bioassay during this period, these intakes are the maximum of a triangular distribution with a minimum of zero and a mode of one-half the maximum.

- For the period from 1958 to 1979 (except 1961, as discussed below), it is unknown what air sample levels would have triggered bioassay; however, it is expected that fewer disassemblies occurred. In addition, there would have been less oxidation of plutonium metal during the earlier years as well. However, because the documentation of the number of disassemblies and the contamination levels is not available, it is claimant-favorable to assign unmonitored

workers an intake that is the same as the intake from the 1980 to 2000 period in accordance with their risk potential from Table 5-2.

Twenty-three non-incident (reason codes annual, routine, or termination) plutonium urinalysis results have been recorded in the Pantex DORMS since 1989; these samples were obtained in 1995 and 1996. All results were below the detection limit (median = 0.00026 pCi/d). Incident-related urine sample results since 1989 were also analyzed as a group and all results except possibly one were below the detection limit (median = 0.0000 pCi/d). Although daily urine excretion of type S plutonium is a small fraction of the intake, these excretion results are consistent with the 40-DAC-hr intake assumed for the 1980 to 2000 period above.

5.2.4.2 1961 Cell Incident

An incident of plutonium exposure occurred in 1961. The details of the event are classified, but some data were available that dose reconstructors can use to estimate intakes. The appropriate intake should be applied to all workers known either to have been involved in this event or to have worked in the cells or bays in 1961 even if it is not clear that they were involved.

Three people were in the cell at the time of the incident, and all were contaminated. The problem was recognized as soon as the incident happened, and the three individuals immediately left the cell. The two operators were wearing respirators, but the foreman standing approximately 6 ft from the release point was not. Initial contamination readings were as high as 450,000 dpm, but there was no contamination outside the cell. There was a statement that an initial urinalysis performed immediately after the incident showed no internal deposition of plutonium, but there is no information on exactly how long after the incident the urinalysis occurred. A urinalysis within a few hours would be unlikely to find anything. In addition, because the material was most likely type S, urinalysis would not have been a particularly sensitive indicator of intake. The air sample data in Table 5-15 were obtained from reports.

Table 5-15. Plutonium incident air sample counts (dpm/m^3), 1961.

Location	0-2 hr	+ 3 d	+ 4 d
Assembly cell	1,900	1.03	0.43
A cubicle	880	0.55	0.156
B cubicle	---	28	0.185
Equipment room	1,030	Filter paper damage	0.156

Analysis of this incident made the following assumptions:

- The breathing rate of the three workers was 3 m^3/hr (ICRP 1994) or 0.05 m^3/min (heavy exercise, adult male).
- The workers were exposed for 5 min (assumes time in the cell and immediately following while they removed contaminated clothing).
- The workers closest to the pit were exposed to the highest air concentration (1,900 dpm/m^3).
- The foreman was not as close to the release, so the second highest air concentration (1,030 dpm/m^3) was used for him.

- The air samples were removed for counting 2 hr after the incident, but essentially all activity on the filters was obtained in the first 5 min. Therefore, the air concentration breathed by the workers in those 5 min was 120/5 min, or 24 times greater than as reported.

- The material was aged weapons-grade plutonium mixture, absorption type S.

- Assuming the plutonium was produced 10 yr before the incident, isotopic ratios can be obtained from Table 5-14.

- A protection factor of 5 (0.2) was allowed for the two workers in respirators. This factor underestimates the protection of most respirators when worn properly with a good fit. However, the quality of the respiratory protection program at that time was not known, and some intake could have occurred during undressing.

Using these assumptions, an estimation of the intake can be made as follows:

- For the workers wearing respirators:

  Total alpha = 0.05 m^3/min × 5 min × 1,900 dpm/m^3 × 24 × 0.2 = 2,280 dpm = 1,027 pCi

- For the foreman without a respirator:

  Total alpha = 0.05 m^3/min × 5 min × 1,030 dpm/m^3 × 24 = 6,180 dpm = 2,784 pCi

Both of these intakes produce expected daily urine excretion far below the likely detection level for the urinalysis. The method of analysis is noted as "N.T.A." The MDA for the analysis was not given, but given the state of the art at that time, sample results less than 0.04 dpm/sample are unlikely to be different from background. Urinalysis of these three workers occurred 8 d after the incident on November 14, 1961. The results were 0, 0.01, and 0.03 dpm. It is claimant-favorable to use the MDA of 0.04 dpm/d as the bioassay result in IMBA (absorption type S), which results in an intake of 66,000 pCi. Because this is substantially higher than expected from the air sample data, it should be considered a constant upper bound. This is the total alpha intake. Assuming 10-yr-old weapons-grade plutonium, intakes of the specific radionuclides are:

Plutonium-239:	52,000 pCi
Plutonium-238:	5,800 pCi
Plutonium-241:	370,000 pCi
Americium-241:	7,700 pCi

In addition to the incident itself, there was decontamination of the cell following the accident. *Decontamination of Building 12-44-6 Following Radiation Accident on November 6, 1961* (MHSMC 1962) contains information on decontamination activities. Exhibit 3 of that report provides urinalysis data for the workers who participated in cleanup operations. Only three samples were 0.04 dpm or higher. These workers wore full-face respirators and full protective clothing during decontamination activities. The first three individuals listed in Exhibit 3 of the accident report are the workers involved in the incident

5.2.4.3 1978 Storage Cylinder Incident

Sometime just before November 14, 1978, a nuclear materials inventory of a Nuclear Weapons Accident Residue (NWAR) storage cylinder (a concrete cylinder about 6 ft in diameter buried vertically

and mostly belowground) was attempted. The mounded earth overcap was removed and the cylinder was opened. Heavy rainfall occurred during the time the cylinder was open, and the cylinder was flooded, which soaked various cans storing radioactive waste. The cans were removed, surveyed, and moved to Magazine 4-75 (also referred to as Igloo 75). Although no contamination was found during the initial survey, the cans were wet so the alpha survey was ineffective. A subsequent survey on November 14 found alpha contamination associated with a small hole in one 11-M can that contained mostly plutonium waste (uranium and tritium contamination was also possible). The igloo was secured. Potentially exposed workers were given bioassay on or about November 17.

A special decontamination procedure was developed which included, for example, full anti-contamination clothing and full-face respirators, continuous radiation monitoring, high-volume air sampling, and HEPA-filtered vacuum cleaners. Cleanup began on January 23, 1979, and concluded by February 8.

Details are provided in MHSMC (1979), which includes the decontamination procedure, results of surveys, pictures, notes associated with the original movement of the cans from the cylinder and with cleanup of the igloo, and results of bioassay samples. It is possible that claims files for energy employees associated with this work do not have these bioassay results; if an energy employee was employed at Pantex in 1978 or 1979, the bioassay results in this file should be checked.

5.3 ASSESSING OCCUPATIONAL INTERNAL DOSE FROM ELEVATED RADON

Uranium occurs naturally in virtually all soils, with average levels of about 1 part per million. Radium-226 is typically in secular equilibrium with ^{234}U and decays to ^{222}Rn with a half-life of 1,600 yr. Therefore, the noble gas ^{222}Rn is continuously produced in soil where it can be trapped in the crystalline structure of minerals or released to the interstices between solid materials. In the absence of buildings, ^{222}Rn produced within a meter or so of the soil surface can diffuse into the atmosphere where diffusion and advection dilute it with outdoor air. During the 1980s it was discovered that buildings with heating and air conditioning tend to operate at slightly negative pressure [a few tens of pascals, less than 1 inch (water gauge)] in comparison to outdoor air. As a result of this negative pressure, soil gas tends to flow actively into indoor air, where it can build to higher levels than outdoors due to limited air changes and relatively small dilution volumes. This phenomenon is an example of technological enhancement of natural radioactivity (NCRP 1984a,b, 1987).

While the general characteristics of areas with potential for elevated levels of indoor radon as well as construction designs that tend to enhance radon levels are known, it is rarely possible to predict indoor radon levels for a given structure. In general, structures that exhaust air to the environment without adequately engineered replacement air have higher indoor radon levels than structures that do not do this, and structures that have exposed soil (dirt floors, sumps) or exposed minerals (e.g., gravel) tend to have higher radon levels. Underground structures have a higher ratio of soil surface to building volume. All other factors being equal, an underground building would be likely to have a higher radon concentration than an aboveground building.

Thorium has ^{220}Rn progeny that is a radioactive noble gas, commonly called thoron, which has a much shorter half-life of 55.6 seconds than its parent. In general, ^{220}Rn decays before it can build up to significant levels unless there are large quantities of ^{232}Th and its decay products present. There is no reason to expect that Pantex had ^{220}Rn of significance. Work on thorium weapons was infrequent.

5.3.1 Dose from Radon-222 Progeny

Radon itself produces far less dose to the bronchial epithelium than its progeny. Because radon progeny measurements are more difficult to obtain, measurements of radon are often used as a surrogate for progeny measurements. Radon progeny concentrations are expressed as the quantity *potential alpha energy concentration* (PAEC), traditionally measured in working levels (WL). Originally, 1 WL was defined as 100 pCi/L (1×10^{-10} Ci/L = 1×10^{-7} Ci/m^3 = 1×10^{-7} µCi/cm^3) of radon in equilibrium with its short-lived decay products. At present, 1 WL is usually defined as any combination of short-lived radon decay products in 1 L of air, without regard to the degree of equilibrium, that will result in the ultimate emission of 130,000 MeV of potential alpha energy per liter of air. This is almost identical to the original definition. Time-integrated exposures to radon progeny are expressed in the quantity *potential alpha energy exposure* (PAEE), which are traditionally measured in working level-months (WLM) and defined as exposure to 1 WL for 170 hr or any equivalent concentration and time product.

5.3.2 Radon Monitoring at DOE Facilities

In 10 CFR § 835.2(a), DOE states, "*Background* means radiation from … radon and its progeny in concentrations or levels existing in buildings or the environment which have not been elevated as a result of current or prior activities…." Because background is specifically excepted from monitoring requirements, DOE and its contractors generally do not monitor for radon and its short-lived decay products. However, if radon and its progeny are elevated due to DOE activities, then DOE requires monitoring. This has been the policy of DOE and its predecessor agencies [the U.S. Energy Research and Development Administration, the U.S. Atomic Energy Commission (AEC), and the Manhattan Engineer District].

5.3.3 Underground Buildings

At Pantex, the Gravel Gertie cells are in Buildings 12-44, 12-85, 12-96, and 12-98, which are considered to be underground even though they are not below grade. Bays, which are also considered underground, are in Buildings 12-17, 12-19, 12-21, 12-56, 12-64, 12-84 East, 12-84 West, 12-99, 12-104, and 12-117. Workers in these buildings were likely to have greater exposures to radon and its decay products than workers in other buildings.

5.3.4 Radon Concentrations

A DOE-wide survey of radon levels (UNC Geotech 1990) sampled 137 locations at Pantex and made duplicate measurements at 13 locations. Table 5-16 lists complete survey data.

Eight buildings at Pantex measured above 4 pCi/L (4×10^{-9} µCi/cm^3), which is the U.S. Environmental Protection Agency reference point for considering remedial action for indoor radon (UNC Geotech 1990). As listed in Table 5-17, the average for all buildings was 1.62 ±1.24 pCi/L with a geometric mean (median) of 1.37 pCi/L and a GSD of 1.68. Values ranged from 0.8 to 8.1 pCi/L. Underground buildings had a higher average, and aboveground buildings had a lower average.

Considering the uncertainty in these measurements, the average absolute difference between duplicate measurements was 0.27 pCi/L with no obvious dependence on the average value of the measurement (Figure 5-3). The average ratio was 1.03, which indicates no significant bias.

Far more important than measurement uncertainty is the issue of representativeness (i.e., an uncertainty that cannot be quantified from available measurements). Most of these Pantex

measurements were made over a 2-month period during the winter, which is normally expected to be the time with the highest radon concentrations because buildings are closed and heated most of the time.

There is an earlier set of radon measurements. For 6 months at the beginning of 1969, Pantex monitored radon levels in Cells 1 to 6 on a twice-monthly basis using Eberline-supplied radon film badges. The raw results were reported as number of tracks in exposed and covered areas, and the integrated radon concentration (in picocurie-hours per cubic meter) was inferred from the net number of tracks (McFall 1969). The integrated radon concentration was converted to an average radon concentration in picocuries per liter by dividing by the number of hours of exposure and multiplying by 1,000 cm^3/L. Of the 66 radon film badges issued, this TBD analysis found no record of analysis for 6, and 60 had reported analyses. Of the 60 reported analyses, 6 were damaged. Of the 54 undamaged results, 33 were reported as zero. When the zeros are included, the overall mean concentration in the cells was 4.2 ±8.6 pCi/L with a range from 0 to 47.2 pCi/L. Fitting a lognormal distribution to all 54 points yielded a median of 1.2 pCi/L with a GSD of 6.7. This median is slightly lower than the median from the 1990 data, and the GSD is considerably larger. These results probably reflect (1) the time of year of the sampling and the longer period over which the sampling took place and (2) the difference in the sampling methods. Table 5-18 summarizes the statistics from the 1969 radon datasets.

Table 5-16. Results of 137 radon measurements in 1990 (UNC Geotech 1990).

RPIS Bu-Ins-Bldg Code	Building number		General description	Gross sq ft	No flrs	Radon (pCi/L)	Dupl. radon (pCi/L)	Install date	Retrieve date	Room
01001DOE BLDG	DOE Building		DOE Building (12-36) is of brick construction (with bricks from the Panhandle area).			8.1		01/10/90	02/16/90	Main Office
0100111-48	11-48		11-48	3200	1	7.8		01/10/90	02/16/90	Maintenance Shop
0100112-104	12-104 East	U	12-104 is bay building	99680	2	7.1		01/10/90	02/16/90	Bay 5
0100112-66	12-66		12-66 is SNM [special nuclear material] warehouse	25900	1	5.4		01/10/90	02/16/90	Center Of building
01001FS-01	FS-1	U	FS-01 is an earth covered storage facility for HE	5364	1	5.2	5.9	01/10/90	02/16/90	Break room
0100112-23	12-23		12-23	3200	1	4.5		01/10/90	02/16/90	North wall middle
0100112-15	12-15		12-15	16800	1	4.2		01/10/90	02/16/90	Training Room 103
0100112-60	12-60		12-60 is Mass Properties Facility	8600	1	4.1		01/10/90	02/16/90	Office - Vault
0100112-79	12-79		12-79 is warehouse/loading dock	28700	1	3.2		01/10/90	02/16/90	South Warehouse Area
0100112-104	12-104	U	12-104 is bay building	99680	2	3		01/10/90	02/16/90	106f R Collins Office
0100112-15	12-15		12-15	16800	1	2.8		01/10/90	02/16/90	Training Graphic Arts
0100112-58	12-58	U	12-58 is bay building	2600	1	2.8		01/10/90	02/16/90	East Wall Between Bay 4&5
0100112-15	12-15		12-15	16800	1	2.6		01/10/90	02/16/90	Training Office
0100112-44	12-44-4	U	12-44 is Gravel Gertie cell building	27100	1	2.6		01/10/90	02/16/90	Round Room
0100112-6	12-6		12-6	23700	1	2.6		01/10/90	02/16/90	Room 131 Quality Records
0100112-14	12-14		12-14	900	1	2.4		01/10/90	02/16/90	Office
0100112-26	12-26 Tooling Warehouse	U	12-26 is bay building	87500	1	2.4		01/10/90	02/16/90	Tooling Warehouse Office
0100112-26	12-26	U	12-26 also has pit vault	87500	1	2.3		01/10/90	02/16/90	Bay 28
0100112-35	12-35		12-35	13400	1	2.3	2.1	01/10/90	02/16/90	Area Mechanics Office
0100111-17	11-17	U	12-17is bay building	6700	1	2.2		01/10/90	02/16/90	Bay 7 Lab
0100112-1	12-1		12-1 is cafeteria/change room	27600	2	2.2		01/10/90	02/16/90	Lepor Colony
0100112-5	12-5		12-5	74400	1	2.2		01/10/90	02/16/90	Electric Shop Office
0100112-86	12-86		12-86 is an Inert Assembly and Test Facility			2.2		01/10/90	02/16/90	Electrical Testing Area
0100111-7	11-7		12-7	34100	1	2.1	2.2	01/10/90	02/16/90	Break room
0100112-61	12-61		12-61	24000	1	2.1		01/10/90	02/16/90	Warehouse Area
0100112-44	12-44	U	12-44 is Gravel Gertie cell building	27100	1	2		01/10/90	02/16/90	Cell 8
0100112-64	12-64	U	12-64 is bay building	32000	1	2		01/10/90	02/16/90	Bay 9
01001STATION 30	Zone 4 Station 30	U	4-30 is underground igloo SNM storage building			2		01/10/90	02/16/90	Control Room
0100112-28	12-28			3500	1	1.9		01/10/90	02/16/90	Quality Hallway
0100112-37	12-37			22700	1	1.9		01/10/90	02/16/90	Room 120 Control Room
0100112-42	12-42 Radiation SAFET			47400	2	1.9		01/10/90	02/16/90	12-42 Cr
0100112-84	12-84	U	Bay	1	1	1.9		01/10/90	02/16/90	Bay 13
0100111-29	11-29			4200	1	1.8		01/10/90	02/16/90	Office
0100112-37	12-37			22700	1	1.8		01/10/90	02/16/90	Room 121 Tech Doc
0100112-37	12-37			22700	1	1.8		01/10/90	02/16/90	Room 112 Mail Room
0100112-5	12-5			74400	1	1.8		01/10/90	02/16/90	Plant Eng Annette Covington
0100112-5	12-5			74400	1	1.8	1.8	01/10/90	02/16/90	Plant Design Eng
0100116-2	16-2 Courier			20072	1	1.8		01/10/90	02/16/90	114 Break room
0100110-9	10-9			15500	1	1.7		01/10/90	02/16/90	Office
0100111-27	11-27			5100	2	1.7		01/10/90	02/16/90	Room 119
0100112-50	12-50			1400	1	1.7		01/10/90	02/16/90	West Of 12-50 Door
0100112-75	12-75			21862	1	1.7		01/10/90	02/16/90	Desk Lieutenants Office
0100112-99	12-99			60716	1	1.7		01/10/90	02/16/90	Break Room
0100111-20	11-20			16600	1	1.6		01/10/90	02/16/90	Office South Wall
0100112-16	12-16			5000	1	1.6	1.1	01/10/90	02/16/90	Plastic Shop Office
0100112-2	12-2 Safety			13456	1	1.6		01/10/90	02/16/90	Dosimetry Lab Room 157
0100112-2B	12-2B			3220	1	1.6		01/10/90	02/16/90	North Wall By Clock
0100112-42	12-42			47400	2	1.6		01/10/90	02/16/90	South Vault
0100112-6	12-6			23700	1	1.6		01/10/90	02/16/90	Room 103 Cafeteria

Table 5-16 (Continued). Results of 137 radon measurements in 1990 (UNC Geotech 1990).

RPIS Bu-Ins-Bldg Code	Building number		General description	Gross sq ft	No flrs	Radon (pCi/L)	Dupl. radon (pCi/L)	Install date	Retrieve date	Room
0100112-84	12-84			1	1	1.6		01/10/90	02/16/90	125d
0100112-36	12-36			29400	1	1.5		01/10/90	02/16/90	Emergency Preparedness
0100112-52B	12-52B1					1.5		01/10/90	02/16/90	Meteorology
0100112-9	12-9			18500	3	1.5		01/10/90	02/16/90	HE Side
0100112-96	12-96	U	Gravel Gertie	7865	1	1.5		01/10/90	02/16/90	Round Room
0100112-99	12-99	U	Bay	60716	1	1.5		01/10/90	02/16/90	Bay 7
0100112-26	12-26	U	Bay	87500	1	1.4		01/10/90	02/16/90	Bay 30
0100112-107	12-107 South			10000	1	1.4		01/10/90	02/16/90	By C.L. Saban's Office
0100112-44E	12-44-E	U	12-44 is Gravel Gertie cell building	1900	1	1.4		01/10/90	02/16/90	Marion Everett's Office
0100112-6	12-6			23700	1	1.4		01/10/90	02/16/90	Room 112 Tom Folks
0100112-68	12-68			35900	1	1.4		01/10/90	02/16/90	Machine Shop Office
0100112-69	12-69			9800	1	1.4		01/10/90	02/16/90	Emmett Hallway
0100111-5	11-5	U	Bay	9000	2	1.3		01/10/90	02/16/90	Control Bay
0100112-2	12-2			13456	1	1.3	2	01/10/90	02/16/90	Medical Office
0100112-24	12-24 South	U	Bay			1.3		01/10/90	02/16/90	Bay 27
0100112-49	12-49			3900	1	1.3		01/10/90	02/16/90	Electronics Room
0100112-6	12-6			23700	1	1.3		01/10/90	02/16/90	104 Room Standards
0100116-12	16-12			28500	2	1.3		01/10/90	02/16/90	Employment
0100112-61	12-61			24000	1	1.2		01/10/90	02/15/90	Break Room
0100112-107	12-107 North			10000	1	1.2		01/10/90	02/16/90	Preventive Maint Section
0100112-11	12-11			2900	1	1.2		01/10/90	02/16/90	Data Management
0100112-42	12-42			47400	2	1.2	1	01/10/90	02/16/90	Upstairs Assembly Ops
0100112-6	12-6			23700	1	1.2		01/10/90	02/16/90	Room 121 Elaine Miller
0100112-64	12-64	U	Bay	32000	1	1.2	1.3	01/10/90	02/16/90	Bay 15
0100112-69	12-69			9800	1	1.2		01/10/90	02/16/90	DOE Office
0100112-9	12-9			18500	3	1.2		01/10/90	02/16/90	Office
0100112-97	12-97B			10000	1	1.2		01/10/90	02/16/90	By Refrigerator in Break Room
0100112-98	12-98	U	Gravel Gertie	34358	1	1.2	1.2	01/10/90	02/16/90	Cell 2
0100112-21	12-21 Gas Lab			29300	2	1.1		01/10/90	02/16/90	Break Area
0100111-18	11-18 Control Room			1500	1	1.1		01/10/90	02/16/90	Control Room
0100111-2	11-2			9600	2	1.1		01/10/90	02/16/90	110
0100111-51	11-51			11600	1	1.1		01/10/90	02/16/90	Office
0100112-32	12-32 South Side			7600	1	1.1		01/10/90	02/16/90	Above Phone on Ramp
0100112-98	12-98-3	U	Gravel Gertie	34358	1	1.1	0.8	01/10/90	02/16/90	Round Room
0100111-36	11-36			5000	2	1		01/10/90	02/16/90	Office
0100111-50	11-50			22151	1	1		01/10/90	02/16/90	Room 110 Office
0100112-100	12-100			4360	1	1		01/10/90	02/16/90	Environmental Protection
0100112-11A	12-11A			5200	1	1		01/10/90	02/16/90	Quality Hallway
0100112-5	12-5			74400	1	1		01/10/90	02/16/90	General Stores Office
0100112-52B	12-52B					1		01/10/90	02/16/90	Meteorology
0100112-52C	12-52C			3600	1	1	0.9	01/10/90	02/16/90	Meteorology
0100112-6	12-6			23700	1	1		01/10/90	02/16/90	Assy Eng Office
0100112-6	12-6			23700	1	1		01/10/90	02/16/90	Room 700 Stoddard
0100112-61	12-61			24000	1	1		01/10/90	02/16/90	Office Area
0100112-82	12-82	U	Bay	6800	1	1		01/10/90	02/16/90	E-Bay Office
01001STATION C	Station C					1		01/10/90	02/16/90	West Wall By Exit
0100112-21	12-21			29300	2	1		01/10/90	02/16/90	X-Ray Office
0100112-2B	12-2B			3220	1	0.9		01/10/90	02/16/90	South Wall - Nancy's Office
0100112-104	12-104	U	Bay	99680	2	0.9		01/10/90	02/16/90	Bay 13
0100112-104-EAS	12-104 East	U	Bay			0.9		01/10/90	02/16/90	Bay 2
0100112-106	12-106			5400	1	0.9		01/10/90	02/16/90	Across From Room 105 Janitor
0100112-24	12-24 North	U	Bay			0.9		01/10/90	02/16/90	Bay 10
0100112-31	12-31	U	Bay	7600	1	0.9	0.8	01/10/90	02/16/90	Bay 3 Outside

Table 5-16 (Continued). Results of 137 radon measurements in 1990 (UNC Geotech 1990).

RPIS Bu-Ins-Bldg Code	Building number		General description	Gross sq ft	No flrs	Radon (pCi/L)	Dupl. radon (pCi/L)	Install date	Retrieve date	Room
0100112-64	12-64			32000	1	0.9		01/10/90	02/16/90	D&I Office
0100112-99	12-99			60716	1	0.9		01/10/90	02/16/90	105-F Manufacturing Office
0100112-84	12-84-East			1	1	0.8		01/10/90	02/16/90	Break Room
0100112-84	12-84	U	Bay	1	1	0.8		01/10/90	02/16/90	Bay 4
0100112-84	12-84			1	1	0.8		01/10/90	02/16/90	Break Room
0100112-84	12-84	U	Bay	1	1	0.8	1.3	01/10/90	02/16/90	Bay 7
0100112-101	12-101 Portable Maint			5334	1	0.8		01/10/90	02/16/90	By Sign-Out Board
0100112-102	12-102			5778	1	0.8		01/10/90	02/16/90	Tech Applications
0100112-103	12-103			23608	1	0.8		01/10/90	02/16/90	Smoking Area
0100112-104	12-104 West			99680	2	0.8		01/10/90	02/16/90	128f Manufacturing Office
0100112-111	12-111			7416	1	0.8		01/10/90	02/16/90	Carpenter Shop
0100112-112	12-112			6525	1	0.8		01/10/90	02/16/90	Camera Room
0100112-17	12-17			32500	2	0.8		01/10/90	02/16/90	Break Area
0100112-19	12-19 EAST			32500	2	0.8		01/10/90	02/16/90	Break Area East Side
0100112-20	12-70					0.8		01/10/90	02/16/90	Cafeteria
0100112-3	12-3			2000	1	0.8		01/10/90	02/16/90	Transportation
0100112-35	12-35			13400	1	0.8		01/10/90	02/16/90	Utilities Console Room
0100112-39	12-39 Fire Department			8200	1	0.8		01/10/90	02/16/90	Sleeping Room
0100112-41A	12-41A			3000	1	0.8		01/10/90	02/16/90	North Wall
0100112-42A	12-42A			19900	1	0.8		01/10/90	02/16/90	Outer Wall by Sandia Sign
0100112-5	12-5			74400	1	0.8		01/10/90	02/16/90	Master Mechanics
0100112-59	12-59			8300	1	0.8		01/10/90	02/16/90	Chem Lab Office
0100112-5C	12-5C			21700	2	0.8		01/10/90	02/16/90	Sheet Metal Shop
0100112-84	12-84	U	Bay	1	1	0.8		01/10/90	02/16/90	Bay 12
0100112-86	12-86					0.8		01/10/90	02/16/90	86-2e-5
0100112-86	12-86					0.8		01/10/90	02/16/90	206s Upstairs Assembly Ops Office
0100112-86	12-86	U	Bay			0.8		01/10/90	02/16/90	Bay 10
0100112-97	12-97A			10000	1	0.8		01/10/90	02/16/90	By Clock in Hallway
0100112-97	12-97C			10000	1	0.8		01/10/90	02/16/90	Above Fire Ext by Copier
0100112-99	12-99	U	Bay	60716	1	0.8		01/10/90	02/16/90	Bay 6
0100116-1	16-1 VMF			54200	1	0.8		01/10/90	02/16/90	Office
0100116-12	16-12			28500	2	0.8		01/10/90	02/16/90	Purchasing
01001STATION B	Station B					0.8		01/10/90	02/16/90	East Wall Center
01001Trailer	Parking Lot					0.8		01/10/90	02/16/90	West Trailer from 12-2
01001Trailer	Parking Lot					0.8		01/10/90	02/16/90	East Trailer from 12-2

Table 5-17. Summary statistics of 1990 radon measurements.

Parameter	All buildings	Underground buildings	Aboveground buildings
Mean (pCi/L)	1.62	1.81	1.56
SD (pCi/L)	1.24	1.35	1.21
CV	0.77	0.75	0.77
GeoMean (pCi/L)	1.37	1.51	1.33
GSD	1.68	1.75	1.66
Min (pCi/L)	0.8	0.8	0.8
Max (pCi/L)	8.1	7.1	8.1
Max/Min	10.1	8.9	10.1
Count	137	31	106

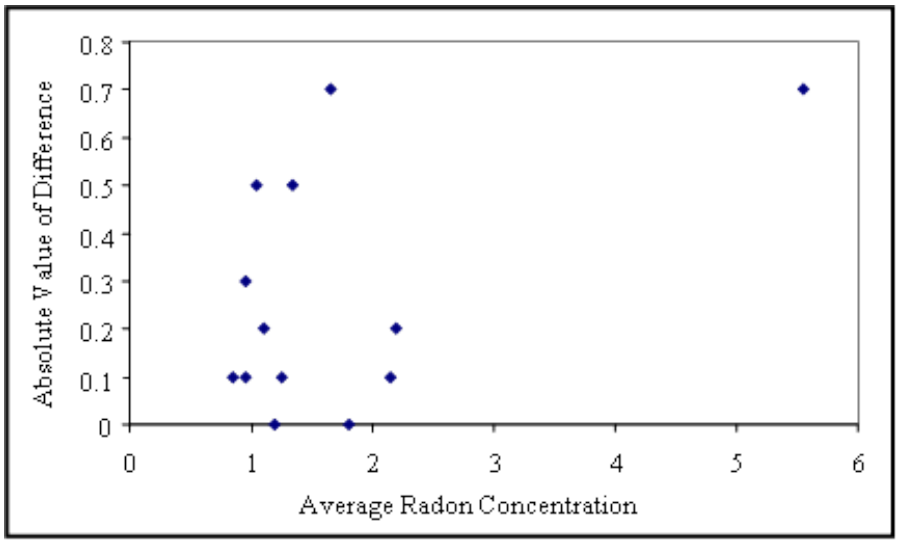

Figure 5-3. Absolute differences between duplicate radon measurements.

Table 5-18. Summary of 1969 radon measurements in Cells 1 to 6.

Parameter	pCi/L	WL
Mean	4.24	0.0170
Standard deviation	8.58	0.0343
Coefficient of Var.	202%	
Minimum	0	0
Maximum	47.22	0.1889
Count	54	
Using 54 measurements (including 33 zeroes):		
Lognormal median	1.20	0.0048
GSD	6.70	6.70
Lognormal mean	7.33	0.0293
Lognormal std. dev.	44.2	0.1767
Using 21 nonzero measurements:		
Lognormal median	7.34	0.0293
GSD	2.47	2.47
Lognormal mean	10.91	0.0437
Lognormal std. dev.	10.89	0.0436

5.3.5 Working Level-Months

The Pantex-measured radon concentrations were converted to equilibrium equivalent concentrations by multiplying the radon concentration by the equilibrium factor *F* using an assumed value of 0.4 as recommended by the ICRP (1981) and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1993). The equilibrium equivalent concentration was divided by 100 pCi/L/WL to arrive at the PAEC. These operations were combined to create:

$$PAEC = C \times F/100 \text{ pCi/L/WL} \quad (5\text{-}21)$$

where *C* is the radon concentration in picocuries per liter and PAEC is in WLs. Dose reconstructors should multiply the PAEC by the months per year of exposure to determine the WLM for input to the Interactive RadioEpidemiological Program (IREP).

For workers who spent most of their time in a facility with an earthen cover (Level 1 from Table 5-2), dose reconstructors should use the 1990 median value for underground buildings of 1.5 pCi/L for *C* and 12 months for the period (unless the person only worked for part of a year). This results in an annual average exposure of:

$$(1.5 \text{ pCi/L})(0.4)(12 \text{ months})/100 \text{ pCi/L/WL} = 0.072 \text{ WLM/yr} \quad (5\text{-}22)$$

For workers with possible occasional entries into underground buildings (e.g., those with a risk ranking of 2 in Table 5-2), dose reconstructors should assume that their WLMs are one-tenth of those from Equation 5-22.

Radon exposure applies from 1958 when the Gravel Gerties were completed to the present. The exposure distribution is lognormal. Parameter 1 is the median value in WLM from Equation 5-22. Parameter 2 is the GSD. Dose reconstructors should use a GSD of 3 to allow for uncertainties in the application of the 1990 radon measurements to a full year (rather than only winter months) and to account for possible yearly differences in radon due to frozen ground or snow cover.

5.4 INTAKE SUMMARY

Table 5-19 provides a summary of the default intakes developed in Sections 5.2 and 5.3. These default intakes should be used when bioassay data are missing or inadequate.

Table 5-19. Summary of default intakes.

	Work or worker category	Period	Material	Mode	Absorption type	Intake (pCi/d or noted)	Distribution	GSD
1	Production techs, QA techs, RSTs, assemblers/disassemblers	1956–present	Tritium	Chronic inhalation/ absorption	NA[a]	From Table 5-6	Triangular (0, mean from Table 5-6, max from Table 5-6)	NA
2	Production techs, QA techs, RSTs, assemblers/disassemblers	1961–1993	DU or U	Chronic inhalation or chronic ingestion	M or S	1.3 or 19	Lognormal	3
					Soluble or insoluble	4.4 or 44		
3	Production techs, QA techs, RSTs, assemblers/disassemblers	1994–present	DU or U	Use 20% of values in row 2				
4	Production techs, QA techs, RSTs, assemblers/disassemblers	1980–2000	Th-232 or Th-228	One acute intake per year	S	48 pCi	Triangular (0, 48, 480)	NA
5	Production techs, QA techs, RSTs, assemblers/disassemblers	2001–present	Th-232 or Th-228	One acute intake per year	S	2.4 pCi[b]	Triangular (0, 2.4, 4.8)	NA
6	Production techs, QA techs, RSTs, assemblers/disassemblers	1958–1979	Pu	One acute intake per year	S	290 pCi	Triangular (0, 290, 2900)	NA
7	Production techs, QA techs, RSTs, assemblers/disassemblers	1980–2000	Pu	One acute intake per year	S	290 pCi	Triangular (0, 290, 2900)	NA
8	Production techs, QA techs, RSTs, assemblers/disassemblers	2001–present	Pu	One acute intake per year	S	14.5 pCi	Triangular (0, 14.5, 29)	NA
9	Production techs, QA techs, RSTs, assemblers/disassemblers	1958–present	Radon	Chronic	NA	0.072 WLM/yr[c]	Lognormal	3
10	Category 2 in Table 5-2 or some risk	1961–1993	DU or U	10% of values in row 2				
11	Category 2 in Table 5-2 or some risk	1994–present	DU or U	2% of values in row 2				
12	Category 2 in Table 5-2 or some risk	1980–2000	Th-232 or Th-228	One acute intake per year	S	10% of value in row 4i	Triangular (0, 4.8, 48)	NA
13	Category 2 in Table 5-2 or some risk	2001–present	Th-232 or Th-228	One acute intake per year	S	10% of value in row 5b	Triangular (0, 0.24, 0.48)	NA
14	Category 2 in Table 5-2 or some risk	1958–1979	Pu	One acute intake per year	S	10% of value in row 6	Triangular (0, 29, 290)	NA
15	Category 2 in Table 5-2 or some risk	1980–2000	Pu	One acute intake per year	S	10% of value in row 7	Triangular (0, 29, 290)	NA
16	Category 2 in Table 5-2 or some risk	2001–present	Pu	One acute intake per year	S	10% of value in row 8	Triangular (0, 1.45, 2.9)	NA
17	Category 2 in Table 5-2 or entry into bays and Gravel Gerties	1958–present	Radon	Chronic	NA	0.0072 WLM/yr[c]	Lognormal	3
18	Machinists	1960–1965	DU	Chronic inhalation	M or S	13	Lognormal	3
19	Machinists	1960–1965	DU	Chronic ingestion	Soluble or insoluble	0.4	Lognormal	3
20	Burning ground techs. /operators	1952–present	DU	Chronic inhalation	M or S	130	Constant	NA
21	Firing site techs/operators	1952–1986	DU	Chronic inhalation	M or S	Per Table 5-11	Constant	NA
22	Firing site techs/operators	Unknown, apply once in employment history	Th-232 in equilibrium	Acute	M or S	14 pCi	Constant	NA
23	Firing Site 5 cleanup if no bioassay was obtained	1994 to 1999	DU	Chronic	S	1.6 pCi/d	Triangular (0, 1.6, 3.2)	NA
24	Firing Site 23 cleanup if no bioassay was obtained	Nov. 10, 1983; Jan. 2, 1984, 85, 86, 87, 88; July 1, 1984, 85, 86, 87	DU	Acute	S	3.2 E5 pCi	Lognormal	3
25	Involved in 1961 accident in bay	Nov. 1961	Pu	Acute	S	(all pCi) Pu-238: 5,800 , Pu-239: 52,000, Pu-241: 370,000, Am-241: 7,700	Constant	NA

a. NA = not applicable.
b. Unless there is a recorded internal dose exceeding 5 mrem, in which case use 0.86 pCi/mrem.
c. Can be prorated by month.

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GLOSSARY

aged

In the context of mixtures of plutonium isotopes, *aging* refers to the time since ^{241}Am separated from the plutonium mixture and then grew back in from decay of ^{241}Pu.

U.S. Atomic Energy Commission (AEC)

Original agency established for nuclear weapons and power production; a predecessor to the U.S. Department of Energy.

dose of record

The dose files provided by DOE to NIOSH as part of the individual worker files.

DU

Depleted uranium; uranium having less than the natural mass of ^{235}U; used as components in nuclear weapons or as a surrogate for enriched uranium or plutonium in testing.

Gertie

A facility covered with crushed gravel used to suppress potential radioactive contamination from the accidental explosion of a nuclear weapon during assembly or disassembly. Also referred to as a *Gravel Gertie*.

equilibrium factor (*F*)

In relation to the potential alpha energy of radon and its progeny in air, the ratio of the equilibrium equivalent concentration (EEC) to the actual activity concentration of radon.

hydroshot

Detonation of a mixture of explosives and DU used as a quality control technique for measuring the performance of plastic-bonded explosives.

likely noncompensable, maximum internal dose approach

An efficiency method of assigning organ (or in some cases whole body) dose whereby the organ is assigned the maximum plausible dose that could have been received. If the resulting probability of causation is 45% or less at the 99% confidence level, the dose reconstruction is considered complete. Also called the overestimate approach.

nuclear emulsion

Often referred to as "NTA" film and normally used to measure personnel dose from neutron radiation; in addition, Pantex apparently used NTA film at one time to measure alpha radiation from radon progeny in air.

potential alpha energy concentration (PAEC)

The kinetic energy potentially released in a unit volume of air by alpha particles emitted by the short-lived radioactive progeny of ^{222}Rn (i.e., ^{218}Po and ^{214}Po) or ^{220}Rn (i.e., ^{216}Po, ^{212}Bi, and ^{212}Po). PAEC is expressed in working levels (WL).

potential alpha energy exposure (PAEE)

The average potential alpha energy concentration (PAEC) to which a worker is exposed, multiplied by the time of exposure in working months of 170 hours; that is, PAEE = PAEC × time. PAEE is expressed in working level months (WLM).

Parameter 1

The column in the IREP template where the dose reconstructor will enter the calculated dose. Multiple entries based on year of employment, type of radiation, and appropriate energy ranges; internal and external exposures are possible.

Parameter 2

The column in the IREP template where the dose reconstructor will enter the lower limit of the dose distribution based on the radiation type and the dose distribution type.

progeny

Radionuclides that result from decay of a parent radionuclide.

radon

Unless otherwise specified, the isotope ^{222}Rn.

thoron

The isotope ^{220}Rn.

working level (WL)

The unit of potential alpha energy concentration (PAEC), defined as any combination of the short-lived radioactive progeny of radon or thoron in 1 liter of air without regard to the degree of equilibrium, that will result in the ultimate emission of 130,000 MeV of alpha energy (1 WL = 2.083×10^{-5} J/m^3) (10 C. F. R. pt. 835).

working level month (WLM)

The unit of potential alpha energy exposure (PAEE), defined as exposure for 1 working month (of 170 hours) to an airborne concentration of 1 WL. (1 WLM = 1 WL × 170 hours = 0.00354 J·h/m^3).